GIF Webinar Series

2016-2023

EDUCATION AND TRAINING WORKING GROUP

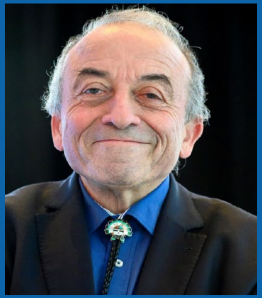
GEN IV International Forum

Expertise | Collaboration | Excellence

Wednesday, 21 June 2023 8:30 am EDT (UTC-4)

International Knowledge Management and Preservation Of Sodium Fast Reactors

Your Presenters and our host:



Joel Guidez
France

Ron Omberg
PNNL, USA

Hiroki Hayafune
JAEA, Japan

Patrick Alexander
TerraPower, USA

Cal Doucette
ARC Clean Energy, Canada

Patricia Paviet
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Meet our Moderator

Dr. Patricia Paviet, our moderator for this panel discussion, is the National Technical Director of the DOE Molten Salt Reactor Program. She is also the Chair of the GIF Education and Training Working Group. Previously, she was the Director of the Office of Materials and Chemical Technologies at DOE, Office of Nuclear Energy, responsible for the R&D activities on the back-end of the nuclear fuel cycle. She has 25+ years of innovative R&D and has worked in government, academia, industry, and national laboratories. She earned her PhD in radiochemistry from the University Paris-Orsay, France.

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Meet the Presenter

Mr. Joel Guidez graduated from “Ecole Centrale de Paris” in 1973. Over his career, he was the head of several activities on Phenix, Superphenix, and Osiris at CEA, France as well as responsible of the High Flux Reactor, Petten, Netherlands and the Nuclear representative at the French Embassy in Berlin, Germany. Since his retirement in 2020, Mr. Guidez has been a scientific advisor of several startup companies, a member of the symposium committee, honorary president of SFEN/ST7, writer of articles and scientific lecturer. His book entitled “ FAST reactors: A solution to Fight against Global Warming” will be published in September by Elsevier edition.

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European feedback experience on the sodium fast reactors and transmission for the future

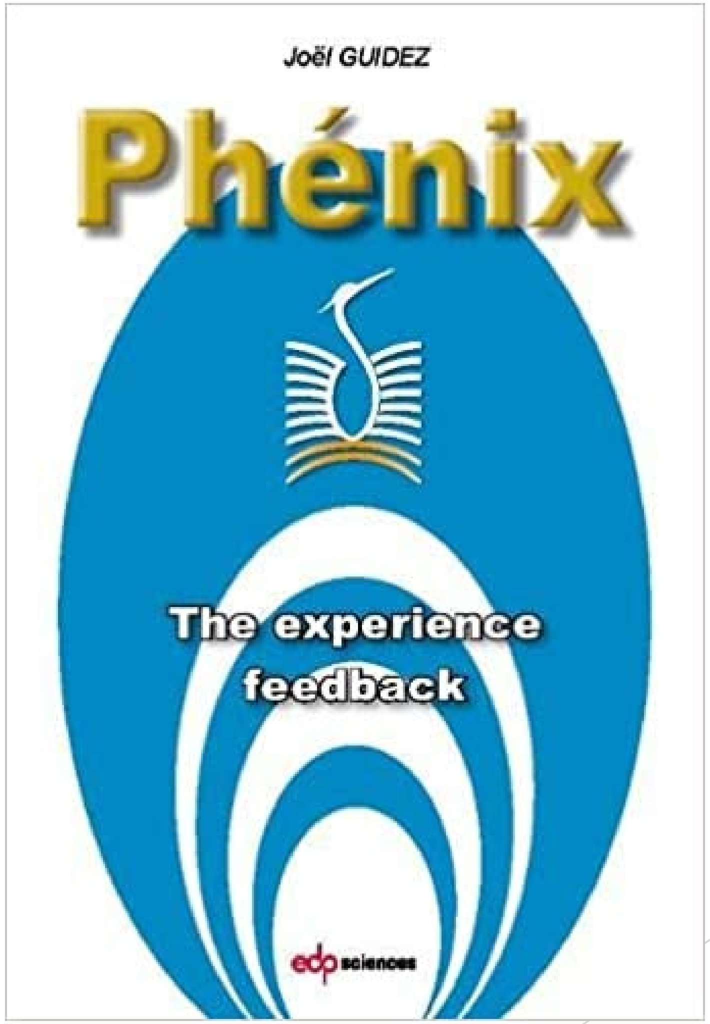
JOEL GUIDEZ

European experience on SFR

- In the 70s , several sodium fast reactors were built in Europe : France (Rapsodie, Phenix) , UK (DFR,PFR), Germany (KNK)
- In the 80s , only France continued but always with European partners
- Superphenix was an European reactor with 33 % from ENEL(Italy) , and 16% from SBK (Germany/UK/Dutch..)
- The project EFR (European Fast Reactor) began with five countries involved, during the Superphenix life but was stopped after the political shutdown of Superphenix.
- Recently (2010/2019) the ASTRID project (600 MWe) was studied in France
- Then, the European project ESFR SMART project began (2017/2022) It was the continuation of EFR , including the knowledge of last European project and of ASTRID
- Now a new European program called ESFR-SIMPLE will begin for four years to continue the ESFR SMART work

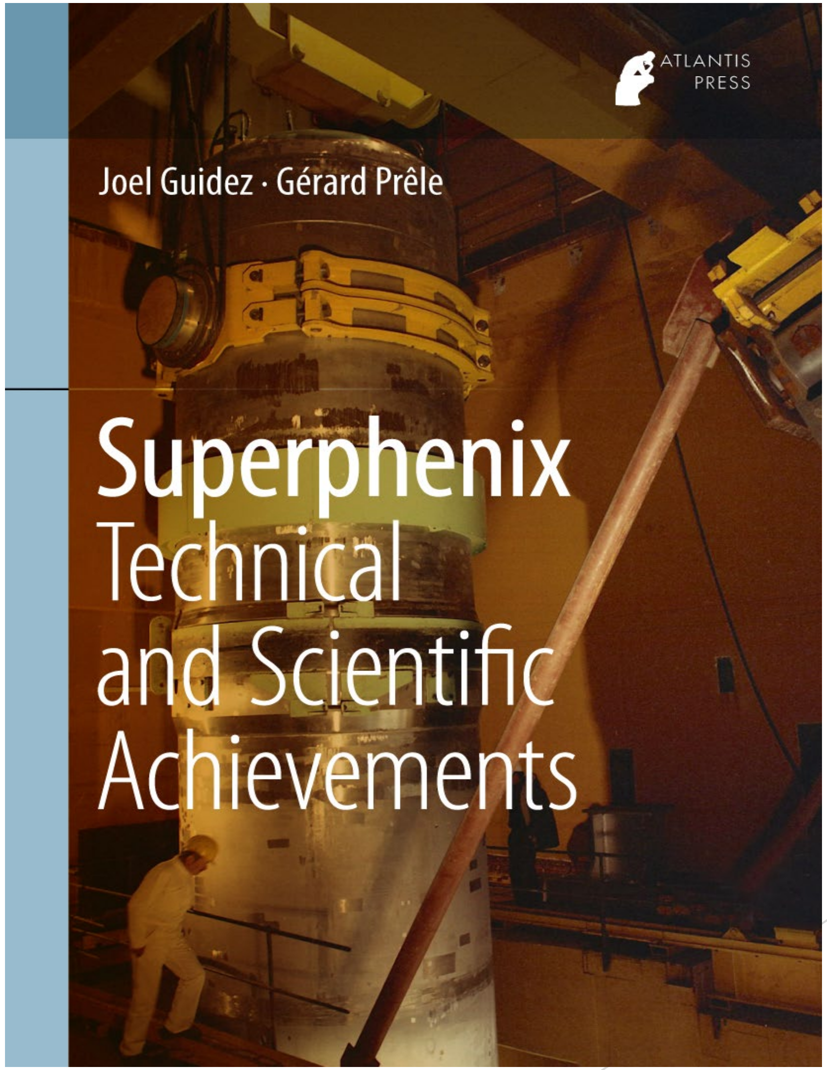
Phenix feedback experience

- The Phenix reactor (250 Mwe) was operated from 1973 to 2009
- There was an industrial demonstration of reprocessing and of manufacture of new fuel with challenges issued from this reprocessing
- The feedback experience on this reactor is resumed in this book with thematic chapters



Super Phenix feedback experience

- It is the biggest SFR never built and operated in the world
- It was operated for only eleven years(1986 to 1996) with a very strong opposition from ecologists. It was stopped on political decision after one year of final good operation and with a good rating from safety authorities
- Half of the time, the reactor was stopped waiting from administrative authorizations.
- This book summarizes the feedback experience on this reactor.



Fast Reactors are ecological in comparison of water reactors

- No need of uranium mines
- No need of enrichment factories
- Operation of the plant during several thousand years, using only waste issued from water reactors (depleted uranium and plutonium issued from reprocessing)
- Possibilities to burn all minor actinides
- Final waste from SFR cycles, are short life waste, easy to store and to manage
- Safer in operation, because without pressure in the circuits
- No releases at all (liquid or gaseous) during operation of the plant
- Better dosimetry for the workers

(see for example the chapter 23 « The environmental results » of the SPX book, or the chapter two of my book on fast reactors))

But SFRs are more expensive

- The cost of the reactor itself is estimated at about 30% more than the cost of a water reactor
- However for FOAK (first of a kind) reactors, there is a « prototype effect » and the cost will be higher . For example, for Super Phenix, the final cost was about 2.2 the cost of PWR
- The fast reactors are interesting if you have all their fuel cycle in place. So you need to have a reprocessing plant for the burned fuel and a factory able to manufacture new fuel subassemblies with the material recovered from reprocessing activities.
- So big investments are needed not only for reactors but also for reprocessing and fuel manufacturing units, during a long period of time and with a continuous political support
- The political and ecological advantages will arrive later, after several decades

What future for SFRs in the world ?

- In the United States, USA, the demonization of plutonium during the Carter administration has stopped reprocessing and fast reactors.
- Political support for nuclear energy changes often and it is difficult to obtain a continuous support for decades, for heavy investments without quick return for the politician in charge
- Few countries have reprocessing plants in operation and are able to support the SFR cycle . Storing used fuel in pools has no ecological future but is less expensive at short term.
- The cost of uranium is today low

Little hope for the future ?

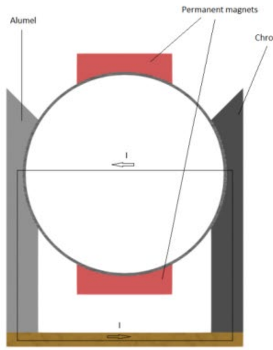
- When uranium will become rare or expensive, some countries will perhaps come back to the utilization of their waste already stored and available ?
- When some ecologists will fight against long life nuclear waste accumulation ?
- And when the passive safety , of plants without pressure and able to support by natural convection accidents as Fukushima , will be explained and give a better social acceptance ?
- It is a difficult political path , but not impossible
- It is the reason , that explains why each European country continues to have reduced teams of scientists with test loops, to maintain competencies in this field
- European projects allow to these teams to work together on common project for the future

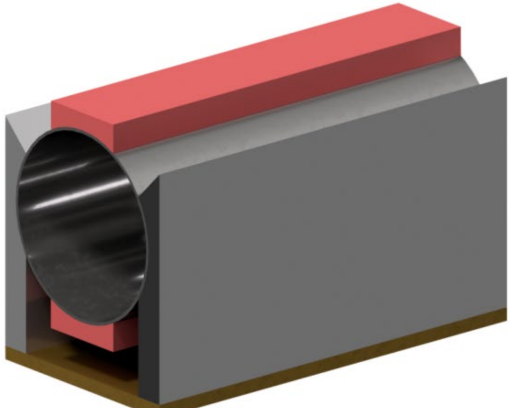
ESFR SMART safety and simplification

- ESFR SMART project was a four years European project, in continuity of the EFR project , from 2017 to 2022
- The principle of ESFR SMART project was to increase the safety of the initial EFR reactor design to be in accordance with the last safety rules issued from Fukushima experience
- All the reactor design was reviewed to be safer, but without new dedicated tools to respond to certain accidents . This type of dedicated tools make the reactor more complex, more difficult to operate and less safe
- The principle of « practical elimination » was applied to see all the possible accidents and to suppress them by dedicated design
- This work was easier due to the big feedback experience on SFR and the knowledge of all the possible accidents on this type of reactors

ESFR SMART : passive and easy to operate

- The reactor can shut down without any order with passive control rods falling only when a physical parameter as outlet sodium temperature increases
- The natural convection of the sodium in the secondary loops allows the decay heat removal by natural convection with air
- A passive system (DHRS 1) allows alone only by natural convection of air and sodium to cool the reactor in case of all the secondary loops drained
- Passive thermal pumps allow to have a good flowrate in all these circuits totaly passively and without any order
- The start of the reactor is very easy (no poison, no bore regulation, no pressure, ..) and the general operation is very simple
- In a « Fukushima situation » the response is very easy





Better mitigation of severe accident

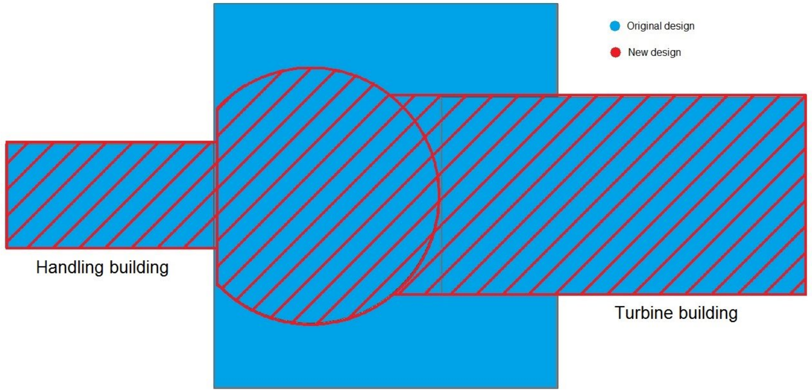
A new safety rule for the reactors is , that in case of severe accident (taken as an obligatory work hypothesis) it should be no radioactive release around the reactor in the short and long term.

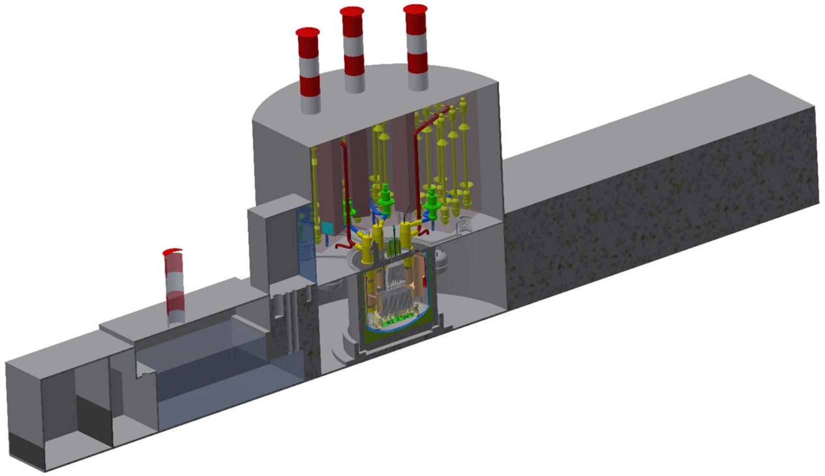
The ESFR SMART design allows to satisfy this rule:

- Even if the main vessel has a leak, the pit design allows to support the sodium leak and to assure natural convection in the core
- The decay heat removal is assured at short and long term by passive circuits out of the vessel and not damaged during the accident
- The design conception (thick metallic roof, components fixed and welded, ..) allows to avoid any leakage even in case of over pressure in the vessel.
- In case of melted core, a core catcher and special devices inside the core assure the management of the melted core
- The decay heat removal of this melted core is made by natural convection inside the pool

ESFR SMART safer but cheaper

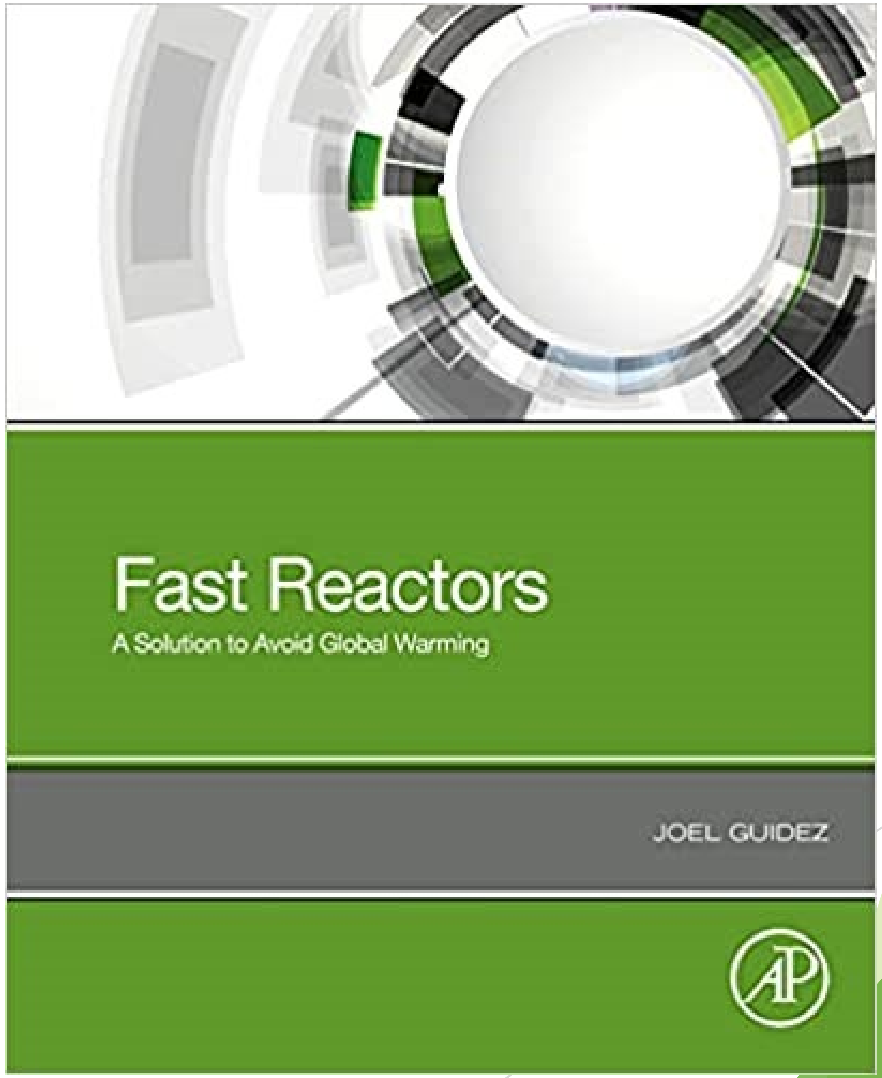
- No more safety vessel
- No more costly exchangers inside the primary vessel
- No more dome or polar table above the primary vessel
- A simple thick metallic roof with no need of cooling or neutronic protection
- With straight pipes, a gain of 50 % on the secondary loops and secondary buildings
- Same chimney for the DHRS 1 and the steam generator casing





To read more on ESFR SMART design

- ASME Journal of Nuclear Engineering and Radiation Science Special Issue: EU ESFR-SMART Project January 2022/ 25 technical papers summarizing main findings of the project
- GIF #61 Webinar J Guidez January 2022, www.gen-4.org
- My new book / ELSEVIER publication/All the chapter 6 is dedicated to ESFR SMART design and drawings



Next European program : ESFR-SIMPLE

- A new European project called ESFR-SIMPLE will begin soon for four years (kick off meeting in October 2022)
- This project will continue with the ESFR SMART options
- Some R&D initiated by ESFR SMART on bellows , thermal pumps, pit structure,.. will continue to validate these options
- The ESFR SMART design (3000 MWt) will be reviewed for lower power with the same basic options , to have several design of SMR available
- Particularly, we will see which design of SMR allows to have a reactor able to be totally manufactured in a factory (main vessel less than 9 m of diameter)
- Some options will be studied to have a better follow up of the grid by these SMR (heat storage with hot sodium, etc.)

Conclusion

- SFRs are more ecological than pressurized water reactors : no need for uranium, no need for uranium enrichment, operation using waste already available, and final nuclear waste of short life easy to manage.
- They are more expensive and need a continuous investment during several decades for the countries involved
- At short term , and with uranium at low cost, there is no hope of an industrial project before the end of this century in Europe
- The European project allows small teams of experts from different countries to remain involved with the subject, to have a global common approach and , project after project ,to improve the design and the knowledge
- A significant part of the project is dedicated to the dissemination of this knowledge for young people and young students
- All this work allows if positive political decisions are taken one day, to restart a project quickly and with the best design available

Thank You

Joel Guidez

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Meet the Presenter

Dr. Ron Omberg graduated from the University of California, Berkeley with a PhD in Nuclear Engineering in 1969 and currently serves as a Principal Technical Advisor at the Pacific Northwest National Laboratory. Since 2000, Mr. Omberg has been responsible for the Fast Flux Test Facility Knowledge Preservation Program for the DOE Office of Nuclear Energy at the Pacific Northwest National Laboratory. He worked on the design of the Fast Flux Test Facility, Westinghouse Hanford from 1970 to 1980; participated in the International Nuclear Fuel Cycle Evaluation (INFCE) from 1976 to 1980; participated in United States/Soviet Union Cooperative Threat Reduction Program, 1999 to 2009, and served as a Member of DOE/NE Nuclear Energy Advisory Committee (NEAC) Subcommittee on Infrastructure from 2000 to 2020.

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GIF Webinar Series

International Knowledge Management
and Preservation Of Sodium Fast Reactors

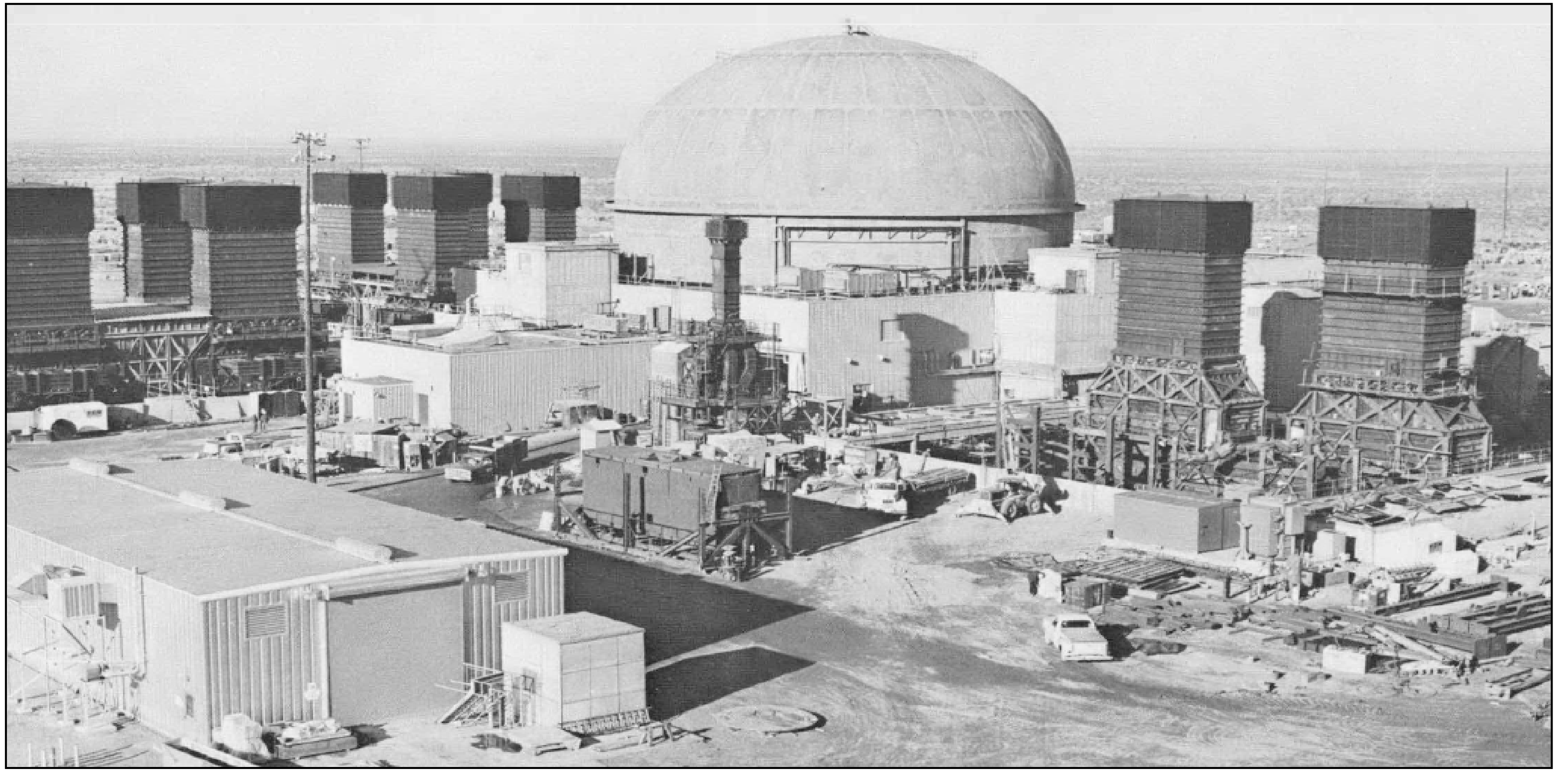
FFTF Knowledge Management and Preservation

PNNL-SA-176044

Ron Omberg (Ph.D.)
Pacific Northwest National Laboratory

21 June 2023

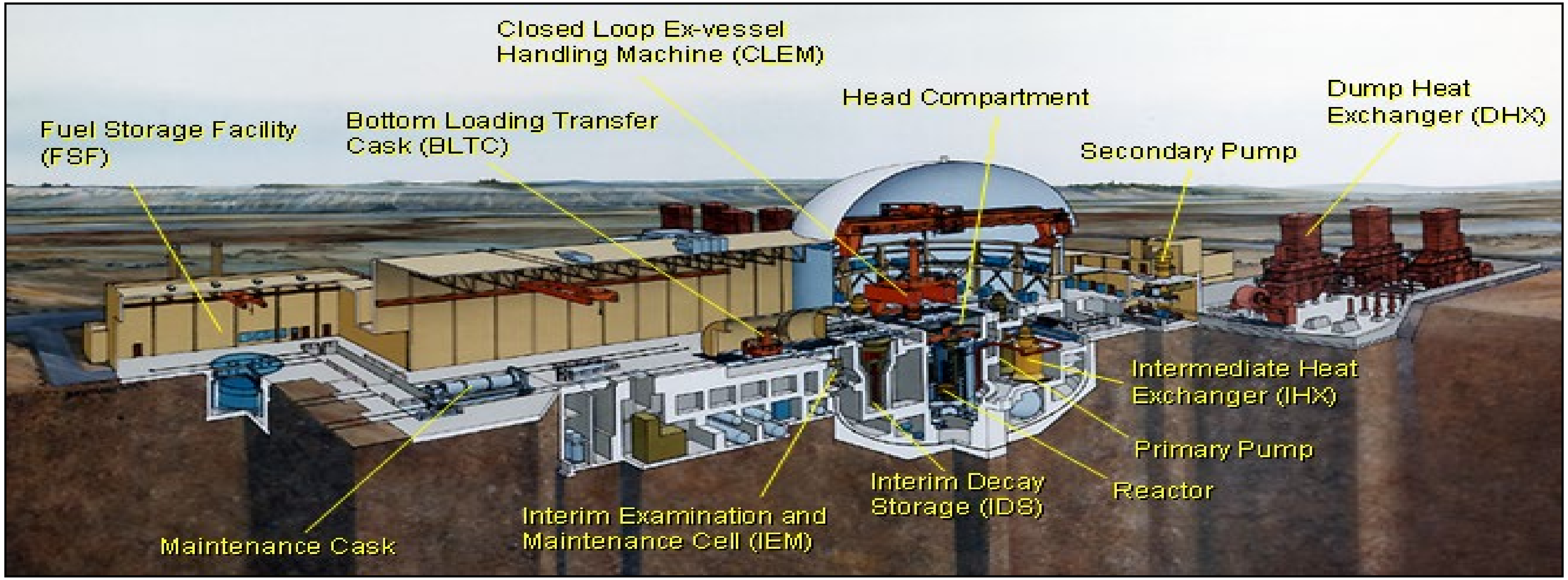
Our Objective
Preserve the Knowledge



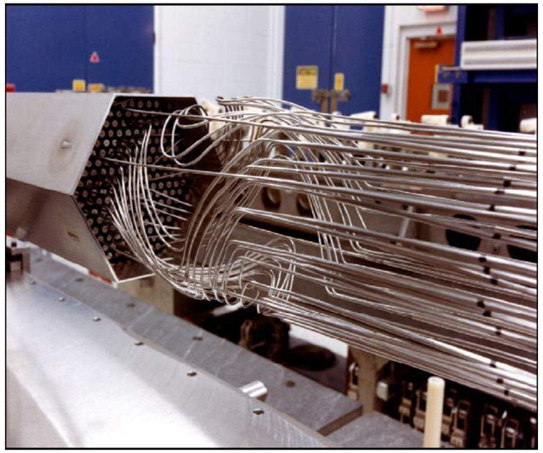
Four and a Half Years into Construction
June 1970 – December 1974

Completed FFTF Reactor Plant

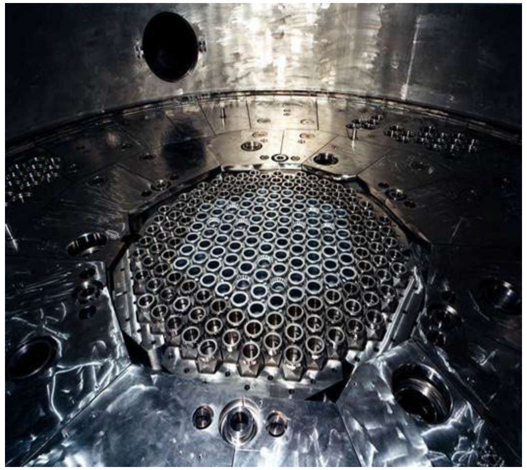
- Sodium Fill – July to December 1978
- Criticality – February 1980
- Power Operation – December 1980
- End of Operation – March 1992



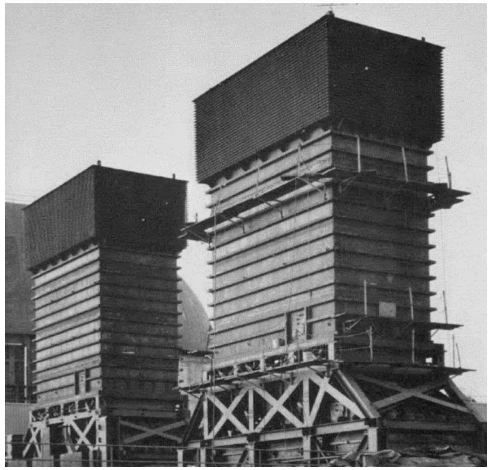
What Knowledge Are We Preserving
- Test Data, Design, Construction -



Irradiation Test Data



Core Before Sodium Fill



Dump Heat Exchangers

How To Organize?

- Knowledge Consists Of:
  - ~ 80,000 Drawings
  - ~ 500,000 Records
- Almost All Hard Copies
- Located in Multiple Record Holding Areas (RHAs) on the Hanford Site
- And So, the Question Is, How to Organize This to Make It Most Useful
- Chris Grandy (ANL) and Ron Omberg (PNNL) Talked and Came Up With the Lessons Learned Approach

Value of the Lessons Learned Approach

- Can Focus Knowledge Preservation on a Timely Issue
- Can Identify What Was Done Well Given What Was Known at the Time
- Likewise, Can Identify What Could Have Been Done Better Given What Is Known Now
- All Relevant Documentation Located, Retrieved, Digitized, and Attached
- Performed by People with First-Hand Experience
- Thereby Incorporating Undocumented Knowledge and Experience

Implementing the Lessons Learned Approach

- ❑ Pick a Topic
- ❑ Might Be, Depending Upon the Year:
  - ❑ *Control Rod Drive Mechanisms*
  - ❑ *Shield Design*
  - ❑ *Ar-41 Management*
  - ❑ *Thermal Transient Usage*
  - ❑ *B_4C Swelling in Control Rods*
  - ❑ *Thermal Striping*

Lesson Learned Format

- Write to:
  - What Did We Do?
  - Why Is/Was It Important?
  - What Was the Outcome?
  - If We Had to Do It Over, What Would We Change or Do Differently?
- Lesson Learned Reports Are on the Order of a Dozen Pages and So Easily Readable
- Key Takeaways Are Summarized
- All References Are Digitized and So Electronic

Lessons Learned Topics Over the Years (~40)

FY15	FY16	FY17	Fy18	FY19	FY20	FY21	FY22
Acceptance and Startup Testing	Heat Exchange Performance	Reactor Physics Startup Testing	Containment Design and Performance	Physics Data Notebook	Control Rod Absorber Design and Performance	FFTF CRDM Design, Testing, and Performance	FFTF Sodium Chemistry Experience
Sodium Natural Circulation and Decay Heat Removal	Thermal Transient Usage	Recording, Archiving and Recovery of FFTF Data	In-Reactor Thimble Characterization	Control Rod Worth, Burnup, and Shutdown Margins	Reactivity Anomaly Detection Monitoring	Thermal Striping Concerns and Evaluations in the FFTF	FFTF Core Restraint System Design
Secondary Sodium Flow Oscillations	Sodium and NaK System Deactivation	Integrated Leak Rate Testing	Core Reload Design	Withdrawal Loads	Pin Power and Heat Deposition Methodology		Control Rod Materials Including B4C
Cesium Release from Failed Fuel and Transport within Reactor Plant	Sodium Thermal Stratification	Control Rod Absorber Assemblies and Boron Carbide Pellets in FFTF	Final Safety Analysis Report (FSAR)				
Gas Entrainment and Accumulation in Sodium and NaK Systems	Sodium Pump Flooding/Shaft Bowing Seizure	Sodium Spill and Fire Testing at the Hanford Site					
Sodium Spills and Fires	Sodium Vapor Trap Design and Operation	Designing for Ease of Decommissioning					
Primary System Pressure Drop Increase	Monitoring and Tracking of System and Component Performance	Removing Non-Drainable Sodium from the FFTF Reactor Vessel					
Bowing in Reactor Assemblies	Sodium and NaK Fill Process						
	Deactivation of Primary Loop Isolation Valves						

U.S. DEPARTMENT OF ENERGY

Nuclear Energy

Acknowledgements

The Author Wishes to Acknowledge the Support and Funding Over the Years from:

Alice Caponiti (Office of Nuclear Energy (NE))

Brian Robinson (NE)

Bo Feng (ANL), Chris Grandy (ANL) and Bob Hill (ANL)

Also, Encouragement and Support from:

Tom O'Connor (Versatile Test Reactor (NE))

Frank Goldner (Versatile Test Reactor (NE))

Meet the Presenter

Mr. Hiroki Hayafune, with the Japanese Atomic Energy Agency, serves as the Deputy Director General, Sector of Fast Reactor and Advanced Reactor R&D. He joined JAEA in 1988 and has participated in Monju and SFR developments. Mr. Hayafune is recognized as a Subject Matter Expert in advanced reactor design.

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Knowledge management and preservation from Joyo, Monju and JSFR experiences

Hiroki HAYAFUNE

Sector of Fast Reactor and Advanced Reactor R&D

Japan Atomic Energy Agency

Problems

Background;

- SFR is a strong candidate for future energy, but SFR development is delayed.
- Many researchers and engineers involved in Monju development are retired.

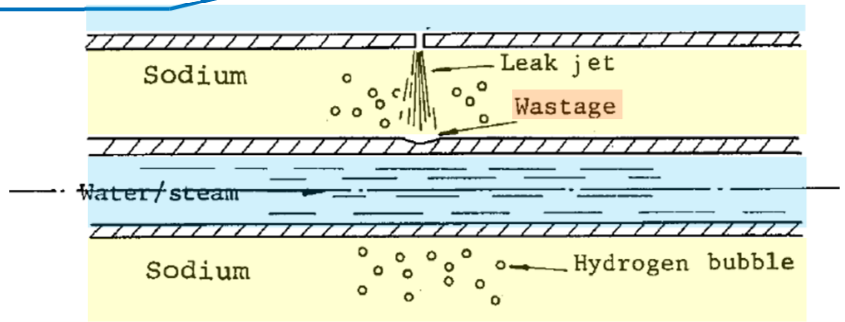
Problems;

- Previous knowledges for SFR development are being impaired.
  - R&D results for basis of design, safety….
  - Know-hows of the plant design
  - Sodium treatment techniques
  - Supply chains
  - Etc.

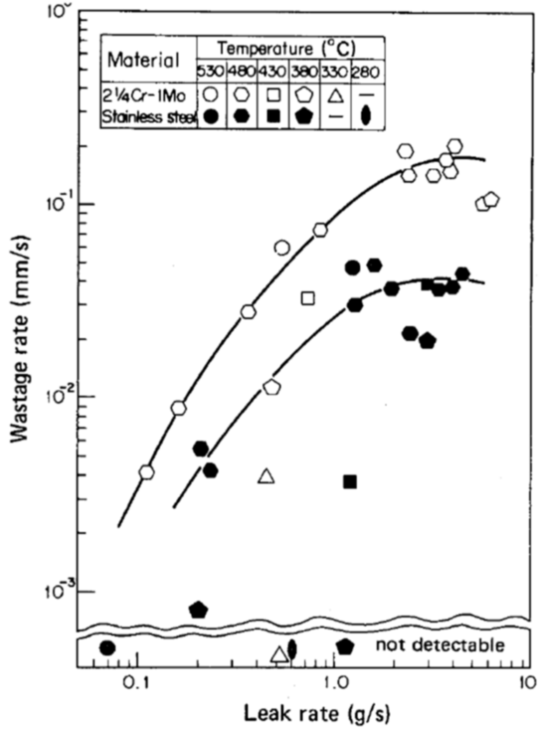
Examples; SFR R&D Knowledges to be preserved

- Design Knowledge:
  - Design standards
  - Safety standards
  - Systems and Component design methods
  - Know-Hows of design
- R&D Knowledge
  - Experimental Bases of designs
  - Experimental Bases of safety
  - Evaluation codes
  - Experimental techniques
- Fabrication and Construction
  - Materials
  - Large Components
  - Claddings, Tubes, etc
- Operation and Maintenance
  - Sodium treatment techniques
  - Limitations for operation
  - Maintenance standards
  - Devices
- Decommissioning
  - Monju decommission

Example-1; Steam Generator tube design



Steam generator tube WASTAGE phenomena



Correlations depend on experiments

<u>Previous design</u>;

- Tube design depends on <u>correlations</u>
- Correlations;
  - have error range
  - have limited parameter range

<u>Future design</u>;

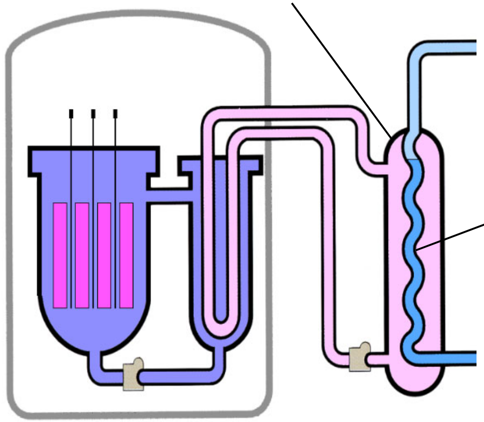
- Mechanistic theory-based analysis;
  - advances reasonable tube design

Experimental Correlations

<u>Design by analysis</u>

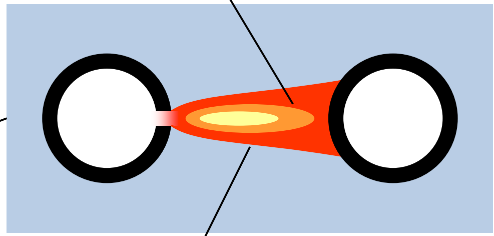
Example-1; Design by analysis

Steam generator (SG)



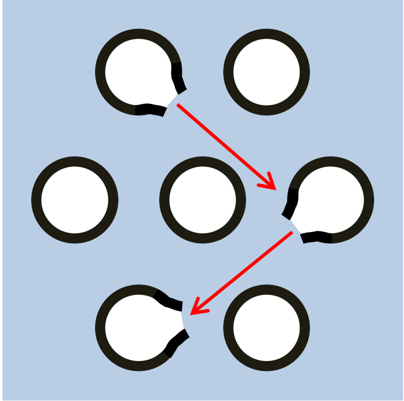
Sodium-cooled fast reactor

High velocity, high temperature jet



$Na+H_2O \rightarrow NaOH+1/2H_2$

target-wastage

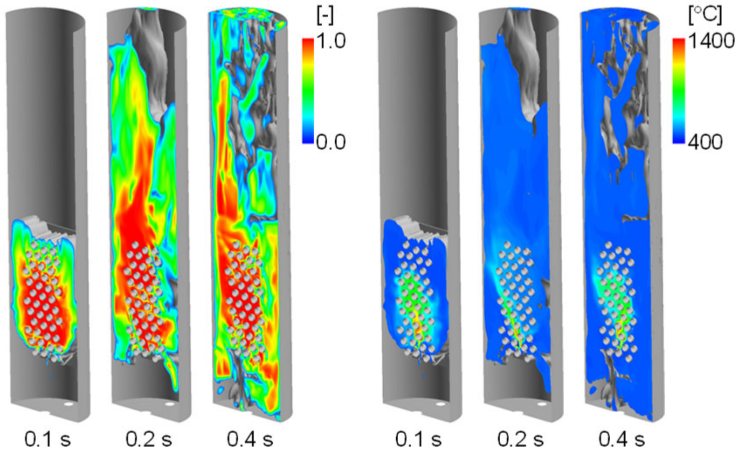


Failure propagation

Future SGs ⇨ Design by analysis

Mechanistic theory-based sodium-water reaction analysis code, SERAPHIM

- Multi-fluid model considering compressibility
- Surface reaction model (gas-liquid reaction)
- Gas-phase reaction model (gas-gas reaction)



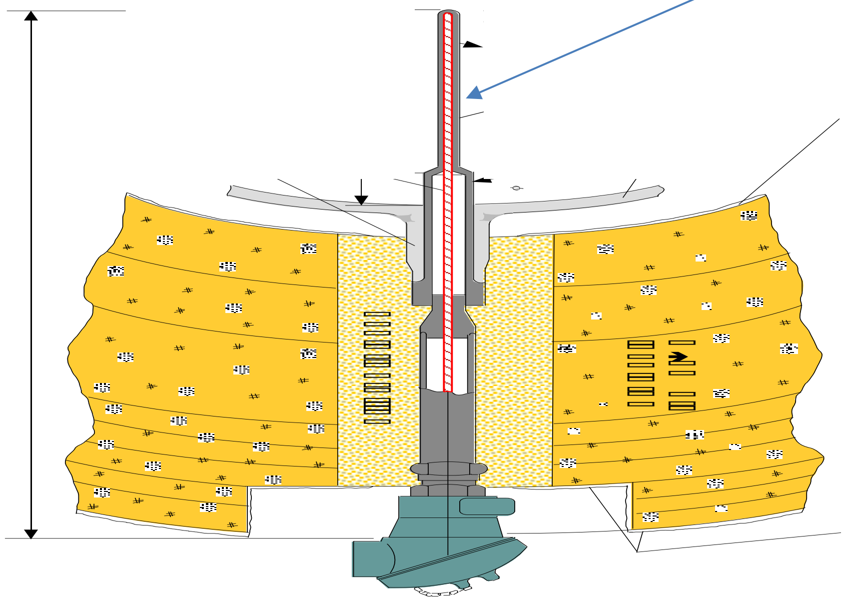
Void fraction

Gas-phase temperature

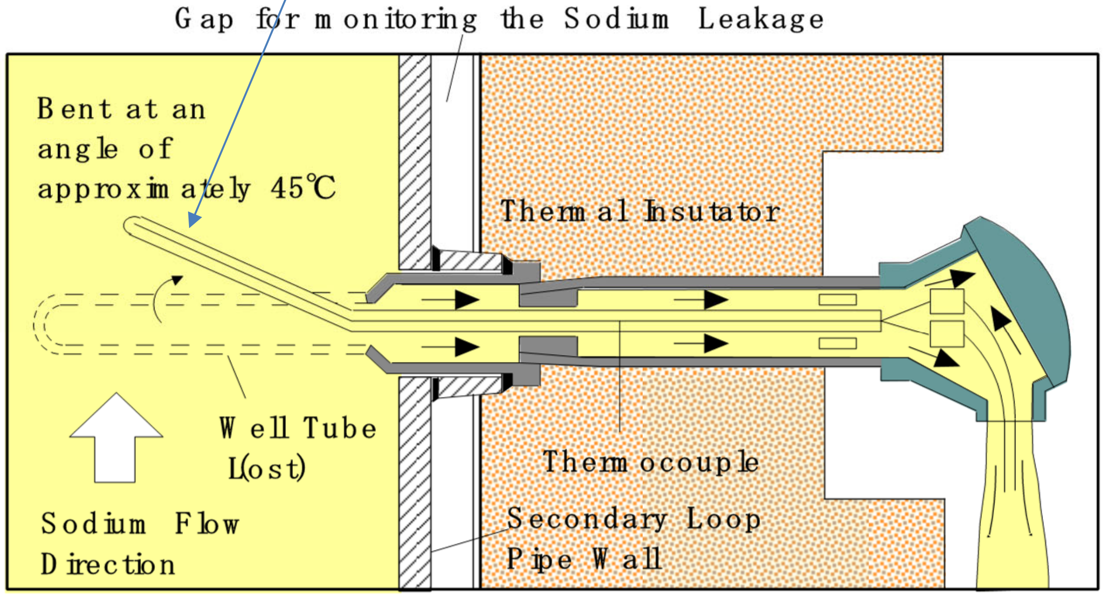
Example of analysis

Example-2; Knowledge from trouble in Monju

Thermo-couple in Main Piping of secondary loop



Well tube was broken by **Symmetric Vortex**

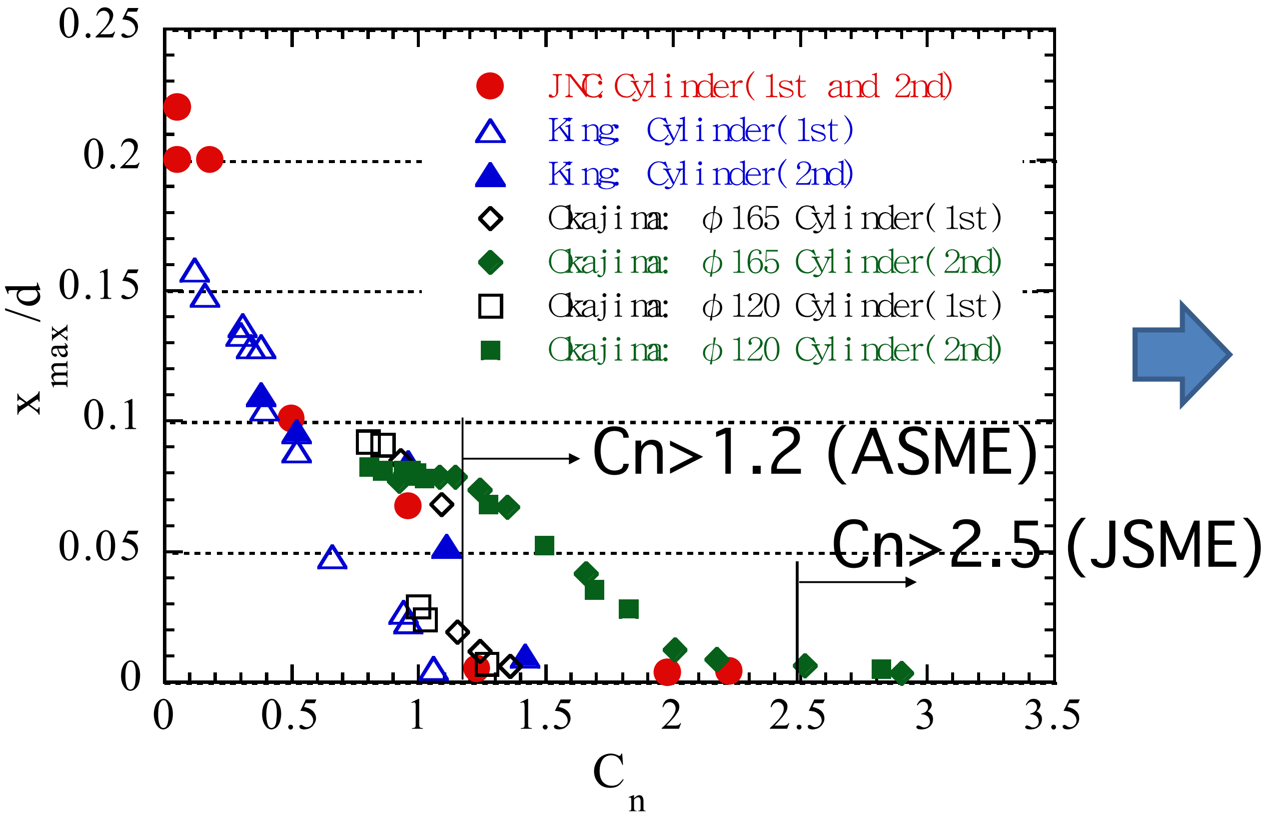


The Sodium Leak Flow Path

Sodium leakage through thermo-couple well in Monju secondary loop piping

Design Guideline is required for future design

Example-2; Design Guideline Development



Experimental Study for “Symmetric Vortex”

ASME guideline('95)	JSME guideline('98)
Vr<3.3 and Cn>1.2	Vr<3.3 and Cn>2.5

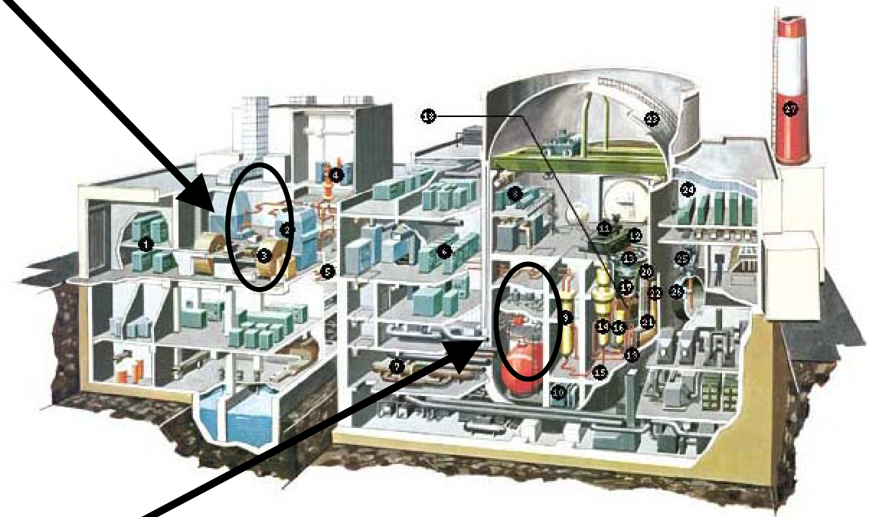
Vr<3.3 :**Avoidance Criterion for VIV in cross-flow direction**

Cn>1.2 :**Suppression Criterion for VIV in flow direction**

Design Guideline for “Symmetric Vortex”

Example-3; Large Components Replacement in Joyo

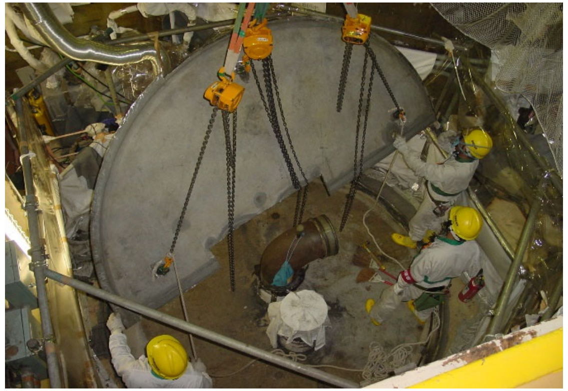
Dump Heat Exchanger
(DHX) × 4



IHXs and Sodium Air Coolers were replaced, due to power increase of Joyo (100-> 140 MW) in 2001

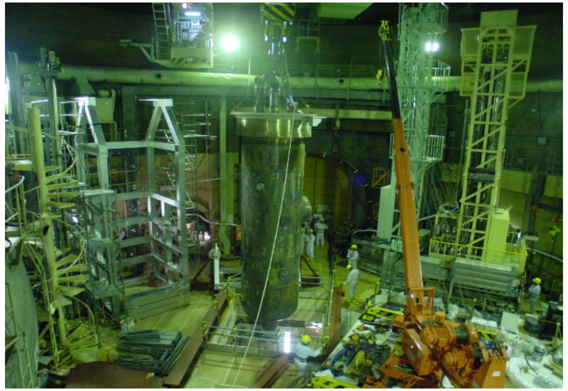
Intermediate Heat Exchanger
(IHX) × 2

Example-3; Experience from the Infrequent Operation

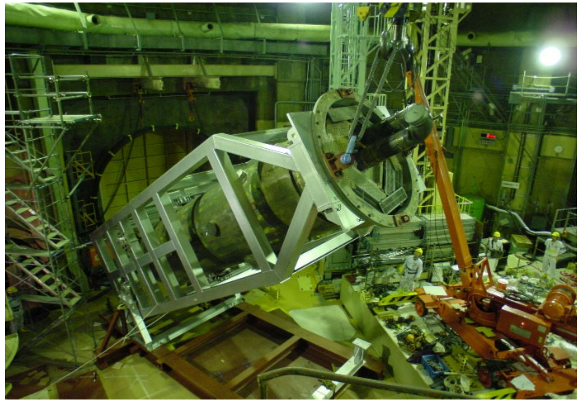


Remove Radiation Shielding Plate

Lift-up



Remove



Transportation Cask laid down

Move from RCV to Maintenance Build.

Lift-down inside Pit of Maintenance Building

<u>Knowledge</u> from the operations;

- Sodium purity control
- Radioactive sodium and Corrosion products treatment
- Reduction of residual sodium

<u>Knowledges</u> to be preserved

Example-4; Sodium treatment techniques in humans

"Sodium School" at Monju site in JAEA

Sodium loop for the training

Sodium treatment training

Sodium fire extinction training

Continuous Trainings and Educations are required

How to manage and preserve the knowledges?

- A lot of R&D results are in the archives.
- Mechanistic theory-based analysis methods are under development.
- Design standards and guidelines are developed.

Not enough
for future design use

Design support system is required in future plant design with previous knowledges.

- To support designers in evaluations and designs
- To support verification and validation of numerical analysis

ARKADIA system development

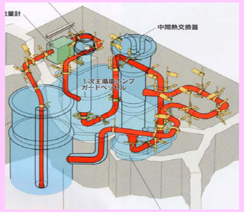
ARKADIA

*(**A**dvanced **R**eactor **K**nowledge- and **A**I-aided **D**esign **I**ntegration **A**pproach through the whole plant lifecycle)*

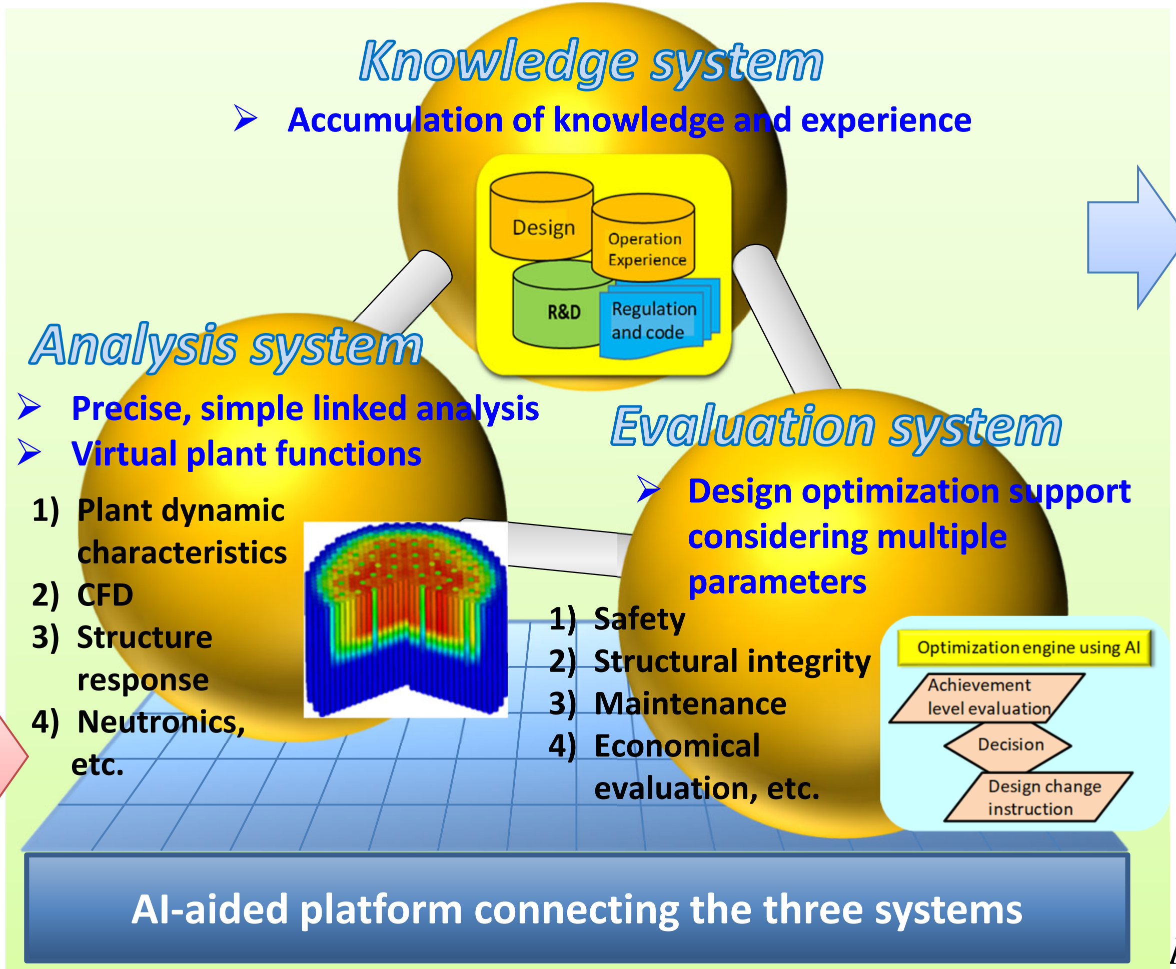
- ➢ proposes competitive plants that meet goals of energy security, economical efficiency, environment, and safety (3E+S);
- ➢ reduces development cost and duration;
- ➢ **builds up and transfers knowledge, develops the technology, and cultivates human resources**.

Input

- Plant design example

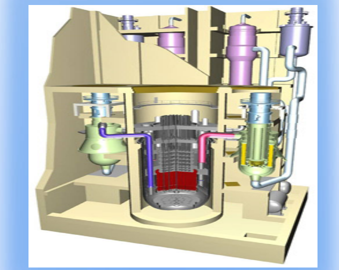


- Evaluation indicators
  - ➢ Development targets
  - ➢ Design requirements



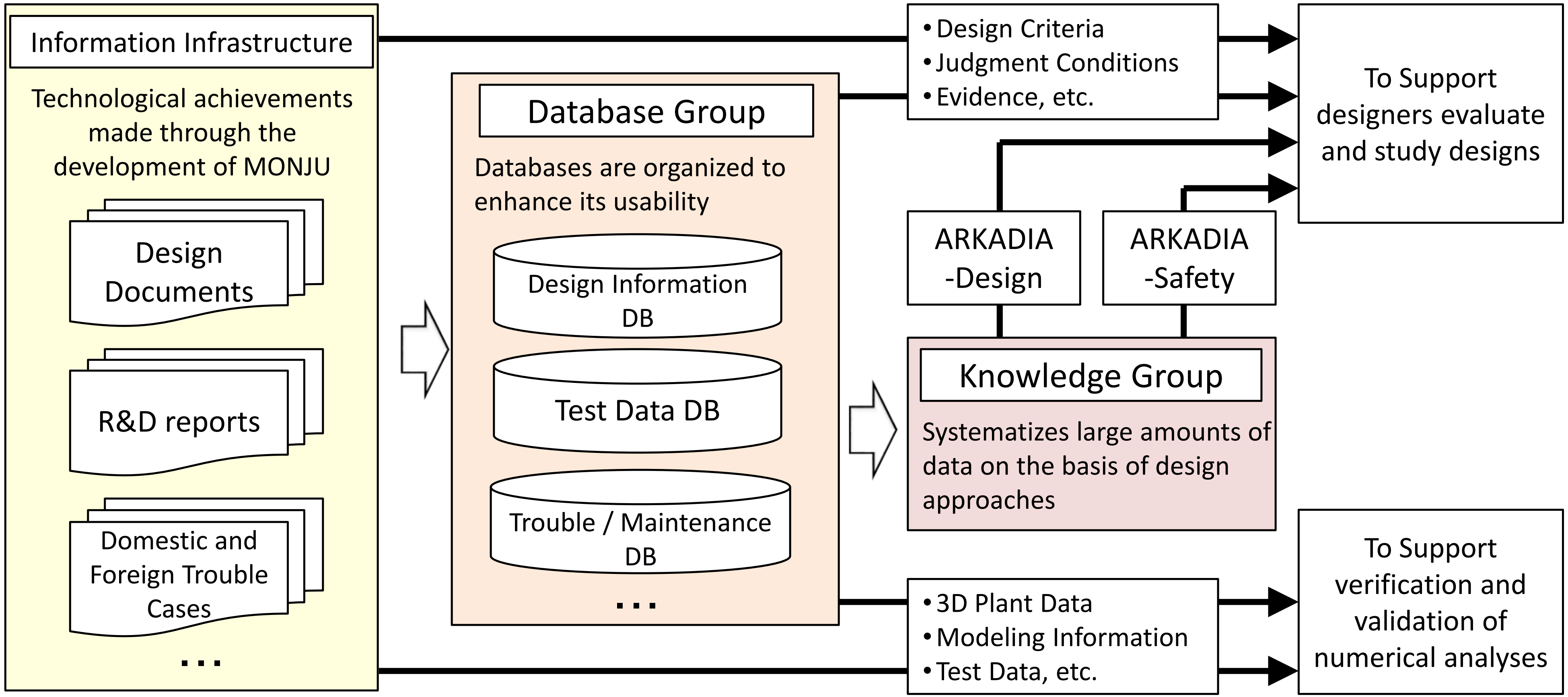
Output

- Proposal of optimized option



- Sufficiency of evaluation indicators
  - ➢ Safety
  - ➢ Reliability
  - ➢ Competitiveness
  - ・・・

Knowledge Management System in ARKADIA



Summary

- **<u>Knowledge management and preservation</u>** are important issue for future SFR development.
  - ✓ To prevent knowledge impairment
    → Knowledge base
  - ✓ To support the plant system designers
    → Knowledge base, Evaluation, and Design should be strongly connected
- JAEA develops "**ARKADIA**" for future SFR development.
- **<u>Education and training</u>** are also important for the operators and technicians.

Meet the Presenter

Mr. Cal Doucette, with ARC Clean Energy Canada, has over 30 years of engineering experience in the petrochemical, wood products, air pollution control, solvent recycling, telecommunications, consulting engineering, and nuclear industries. Most recently, Mr. Doucette served as a design engineering section head and system responsible operations specialist with Canadian Nuclear Laboratories. In addition, Mr. Doucette was the project manager for emergency core cooling strainer installations, lead engineer for the NRU vessel leak repair project and responsible for the processing of legacy liquid waste through the CNL liquid waste immobilization system. Mr. Doucette earned his Bachelor of Chemical Engineering degree from McGill University.

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www.arc-cleantech.com

ARC CLEAN TECHNOLOGY CANADA

OPERATING EXPERIENCE AND TECHNICAL DEVELOPMENTS, THE ARC-100 SODIUM FAST REACTOR

August 2023

Cal Doucette P.Eng. (APEGNB/PEO)
Director, Engineering
Arc Clean Technology
Brunswick Square
1 Germain Street, Saint John, NB E2L 4V1

arc
CLEAN TECHNOLOGY

Incorporating Operating Experience

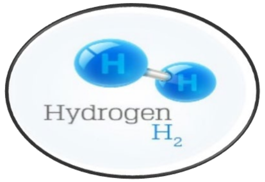
- Why Sodium Fast Reactor?
- Sodium Fast Reactor 101
- Collecting and Sharing Experience
- SMR Technology and Evolution
- Canadian Example - CANDU Evolutionary Path
- Genesis of ARC-100
- ARC Evolutionary Path
- ARC-100 Design Overview

Why Sodium Fast Reactor

Integration with Industry & Energy Systems

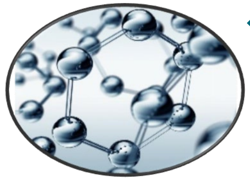
Industrial Application

Industry

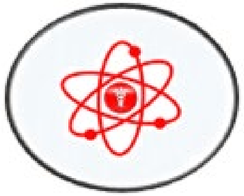


Hydrogen Production

Clean Water



New Chemical Processes



Medical Isotopes

ARC-100

Inherent Safety

Power control characteristics | Exceptional load following characteristics

Offering energy flexibility and versatility to partner with renewables

Electrical Generation

Heat Storage for Peaking Ability

Electrical Grid

Renewables

Wind Power

Hydro Power

Solar Power

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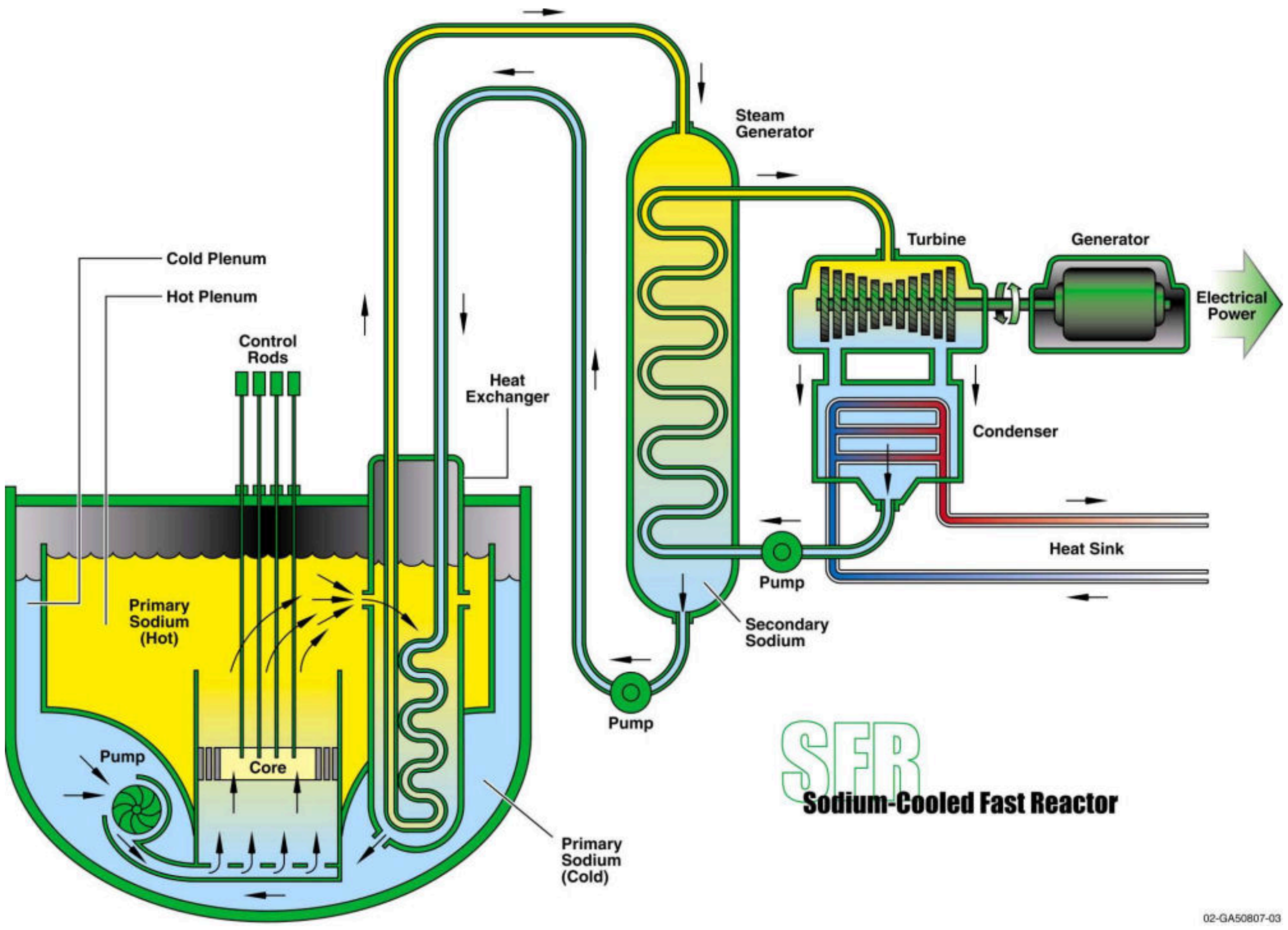
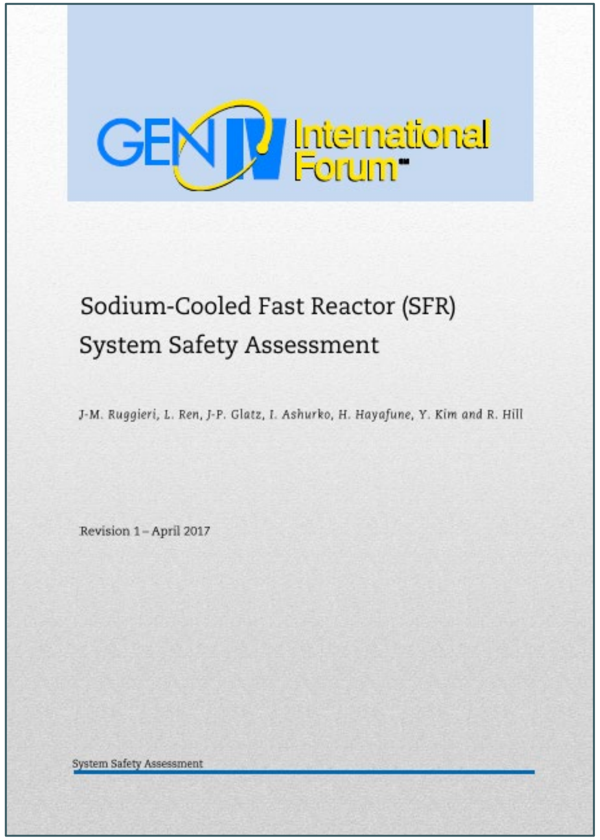
Sodium Fast Reactor 101



- The SFR uses liquid sodium as the reactor coolant, allowing operation at low pressure, reducing driving force for off-site dose.
- Much of the basic technology for the SFR has been established in former fast reactor programs, so this is not a FOAK Technology.
- High level of safety achieved through inherent and passive means with significant safety margins.
- SFR’s with superheated steam are ideal for electrical grid and industrial heat applications that create broad market opportunities.

Maturity Level of SFR's


GEN IV International Forum

Sodium-Cooled Fast Reactor (SFR)
System Safety Assessment

J-M. Ruggieri, L. Ren, J-P. Glatz, I. Ashurko, H. Hayafune, Y. Kim and R. Hill

Revision 1 – April 2017

System Safety Assessment

"When developing a new reactor system, accumulation of experiences on fuel, safety, and material behavior is extremely important and takes a significant amount of time.

In the SFR case, those experiences have been successfully accumulated during the past seventy years world-wide. There were and are a lot of experimental and prototype SFRs.

Fuel/subassembly specification, safety designs and materials of future SFRs are based on perspectives from those experiences.

From the safety point of view, a lot of large demonstration experiments have been conducted and evaluation tools validated by those experiments have been developed."

There is an extensive experience base in both experimental and prototype SFR's supporting it as a *Mature Technology*

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Fast Neutron Reactors

WORLD NUCLEAR ASSOCIATION

	MWe	MW (thermal)	Operation
USA			
EBR 1	0.2	1.4	1951-63
EBR II (E)	20	62.5	1963-94
Fermi 1 (E)	61	200	1963-75
SEFOR		20	1969-72
Fast Flux Test Facility (E)		400	1980-93
UK			
Dounreay FR (E)	15	65	1959-77
Protoype FR (D)	250	650	1974-94
France			
Rapsodie (E)		40	1967-83
Phenix* (D)	250	563	1973-2009
Superphenix (C)	1240	3000	1985-98
Germany			
KNK 2 (E)	20	58	1972-91
India			
FBTR (E)	13	40	1985-
PFBR (D)	500	1250	under const.

Japan			
Joyo (E)		50, 75, 140	1978-2007 2021 restart?
Monju (D)	280	714	1994-96, 2010.
Kazakhstan			
BN-350* (D)	135	750	1972-99
Russia			
BR 1/2 (E)		1/0.1	1955, 1956
BR 5 Obninsk (R)		5	1958-71
BOR 60 Dimitrovgrad (R)	12	60	1969-
BR 10 Obninsk (R)		8	1973-2002
BN-600* Beloyarsk 3 (D)	600	1470	1980-
BN-800 Beloyarsk 4 (E)	864	2100	2014-
MBIR (E)	40	150	under const.
China			
CEFR (E)	20	65	2010-
CFR600 (D)	600	1500	under const.

SFR Technology is supported worldwide by a significant number of Reactor facilities that developed from FOAK through an evolutionary path to demonstration/commercial sizes

Fast Neutron Reactors – historical and current
E = experimental, D = demonstration or prototype, C = commercial, R = research

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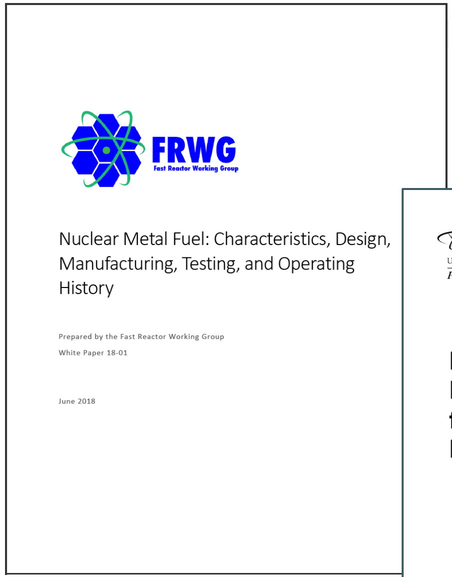
Collecting & Sharing Experience

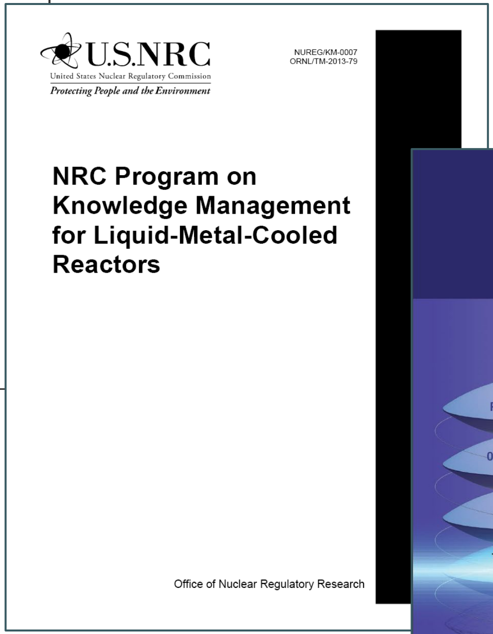

FRWG
Fast Reactor Working Group

Nuclear Metal Fuel: Characteristics, Design, Manufacturing, Testing, and Operating History

Prepared by the Fast Reactor Working Group

White Paper 18-01

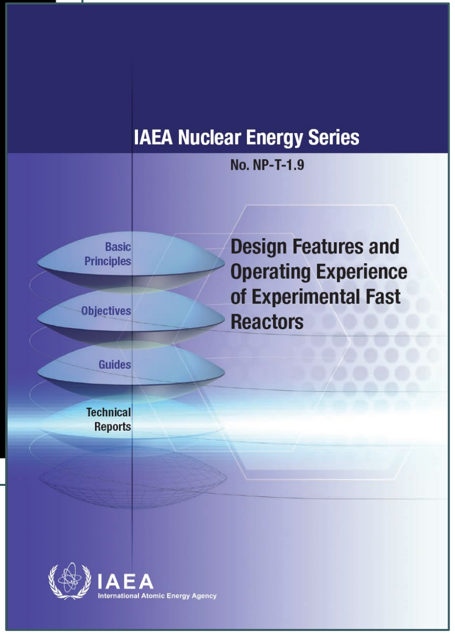
June 2018


U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG/KM-0007
ORNL/TM-2013-79

NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors

Office of Nuclear Regulatory Research


IAEA Nuclear Energy Series

No. NP-T-1.9

Design Features and Operating Experience of Experimental Fast Reactors

IAEA
International Atomic Energy Agency

- About 20 fast neutron reactors (FNR) have already been operating, some since the 1950s, and some supplying electricity commercially.
- Over 400 reactor-years of operating experience has been accumulated that has been documented in OPEX documents.
- Design information, Operating reports and issues as well as Decommissioning information are readily available through the IAEA, US DOE, OSTI , INL, ORNL, ANL as well as research papers on various aspects of SFR technology.
- Fast neutron reactors are a technological step beyond conventional power reactors and are poised to become mainstream.
- Advanced Reactor Concepts (ARC) set up in 2006 has developed a sodium-cooled fast reactor based on the EBR-II and incorporated lessons learned from the significant past and current operating experience.

Cross-Section of SMR Technology

Micro-SMRs

- OKLO
- U-BATTERY Local Modular Energy
- ULTRA SAFE NUCLEAR
- Westinghouse

Grid Scale SMRs

Gen III+

Water Cooled

- NUSCALE
- ROLLS ROYCE
- GE HITACHI (BWRX300)
- HOLTEC INTERNATIONAL

Gen IV (Advanced Reactors)

Sodium Cooled Fast Reactors

- ARC CLEAN ENERGY
- TerraPower (Natrium)

High Temperature Gas Reactors

- X-energy
- framatome
- GENERAL ATOMICS

Molten Salt

- TERRESTRIAL ENERGY
- moltex clean energy
- Kairos Power
- TerraPower

Grid Scale Gen IV Advanced SMR Comparison

	Sodium Cooled Fast Reactors		High Temperature Gas Reactors		Molten Salt			
	ARC CLEAN ENERGY	TerraPower (Natrium)	X energy	GENERAL ATOMICS	TERRESTRIAL ENERGY	moltex clean energy	Kairos Power	TerraPower
Size (MW)	**100MW**	**345MW**	**80MW**	**265MW**	**195MW**	**300MW**	**140MW**	**345MW**
Outlet Temp	510C	540C	565C	850C	700C	590C	650C	755C
Refuel Cycle	240 Months	24 Months	Online	360 Months	Online	Online	Online	Online
CNSC Status	VDR I Complete VDR II In Progress	-	VDR I& II In Progress	-	VDR I Complete VDR II In Progress	VDR I Complete VDR II No Status	-	-
NRC Status	-	Preapplication Activities	Preapplication Activities	Preapplication Activities	Preapplication Activities	-	Preapplication Activities	Preapplication Activities
Grants- Canada	☑				☑	☑		
Grants- USA	☑	☑	☑	☑	☑	☑	☑	☑
Operational Experience	450+ Years	450+ Years	~50 Years	~50 Years	~5 Years	~5 Years	~5 Years	~5 Years
Currently in Operation	Yes (BN-600, BN-800)	Yes (BN-600, BN-800)	–	–	–	–	–	-

SFRs that are Deployable by 2030

Argonne NATIONAL LABORATORY

ANL/ART-88 Rev. 02

Research and Development Roadmaps for Liquid Metal Cooled Fast Reactors

Nuclear Engineering Division

Table 2.1.3 U.S. SFRs Deployable by 2030s

Design parameter	PRISM [a]	ARC-100	TWR-P	FASTER [b]
Developer	GE-H	ARC, LLC	TerraPower	DOE
Power, MWt/MWe	471/165	250/100	1475/600	300/120
	840/311			
Primary system type	Pool	Pool	Pool	Pool
Fuel form	Metal	Metal	Metal	Metal
Fuel composition - Start-up core - Eq. core	 U-Zr U-TRU-Zr [c]	 U-Zr U-Zr	 U-Zr U-Zr	 U-Pu-Zr U-Pu-Zr
Coolant outlet temperature, °C	~500	550	510	510
Power conversion	Steam	Steam or SCO_2 Brayton [d]	Steam	Steam
Ave. driver burnup, GWd/t	66	TBD	150	34
Cladding material	HT-9	HT-9	HT-9	HT-9
Primary sodium pump	EM	Mechanical	Mechanical	Mechanical

a) General Electric has different variants of PRISM: PRISM/Mod A (471 MWt), PRISM/Mod B (840 MWt), and S-PRISM (1000 MWt).

b) FASTER [Grandy 2016] is a test reactor concept, not for commercial deployment.

c) TRU = transuranic.

d) SCO_2 = supercritical CO_2.

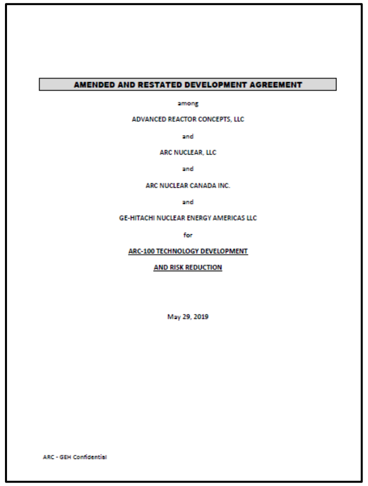
Genesis of ARC-100


United States Patent
Walters
Patent No.: US 8,767,902 B2
Date of Patent: Jul. 1, 2014
SMALL, FAST NEUTRON SPECTRUM NUCLEAR POWER PLANT WITH A LONG REFUELING INTERVAL
Inventor: Leon C. Walters, Idaho Falls, ID (US)
Assignee: Advanced Reactor Concepts LLC, Reston, VA (US)

CA 3024458 A1 2018/04/26
3 024 458
DEMANDE DE BREVET CANADIEN
CANADIAN PATENT APPLICATION
ADVANCED REACTOR CONCEPTS LLC, US
WALTERS, LEON C., US
Titre : MISE A NIVEAU DE LA PUISSANCE DE SORTIE DE CENTRALES NUCLEAIRES PRECEDEMMENT DEPLOYEES
Title: UPGRADING POWER OUTPUT OF PREVIOUSLY-DEPLOYED NUCLEAR POWER PLANTS

- U.S. Department of Energy (DOE) National Laboratories historically provided technical support to promoted development of fast reactors and several Countries/Companies entered into agreements.
- “Advanced Reactor Concepts” was able to obtain a U.S. Government agreement for a scope to support a family of Patent Applications for a segment of the market that matched its mission of “SMALL, FAST NEUTRON SPECTRUM NUCLEAR POWER PLANT WITH A LONG REFUELING INTERVAL”.
  - Small modular sodium cooled reactor with metallic fuel nominally producing 100 MWe with a range of 50 to 100 MWe and with a long core life of approximately 15 to 20 years.
  - Basis was aimed at compatibility with smaller grids and smaller capital outlay.
  - Long core life was for energy security and safeguards to facilitate international nonproliferation regime even for widespread worldwide deployment in developing Countries.
  - Secondary Patent for uprate to 200 MWe with same core and nominal ½ original core life.
- Canadian Patent Applications have also been filed.

Genesis of ARC-100 (continued)



AMENDED AND RESTATED DEVELOPMENT AGREEMENT

among

ADVANCED REACTOR CONCEPTS, LLC

and

ARC NUCLEAR, LLC

and

ARC NUCLEAR CANADA INC.

and

GE-HITACHI NUCLEAR ENERGY AMERICAS LLC

for

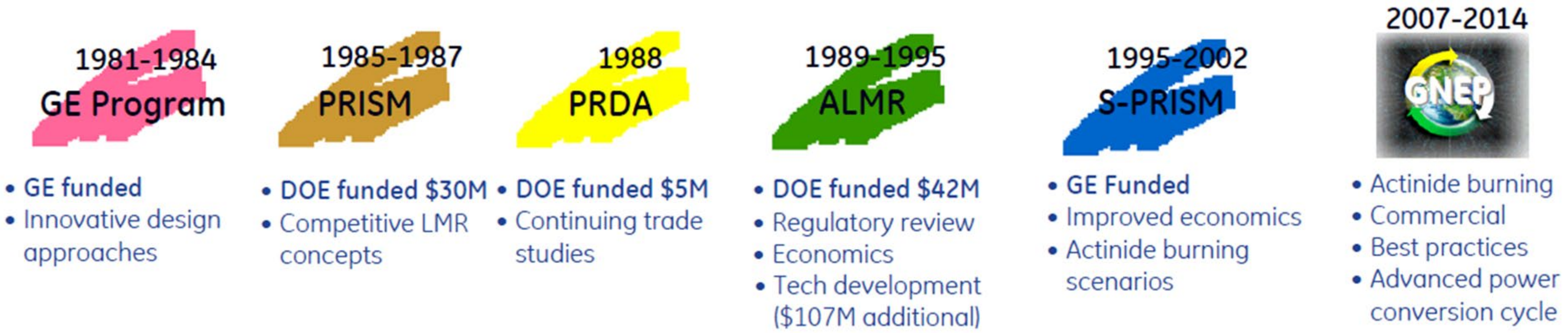
ARC-100 TECHNOLOGY DEVELOPMENT

AND RISK REDUCTION

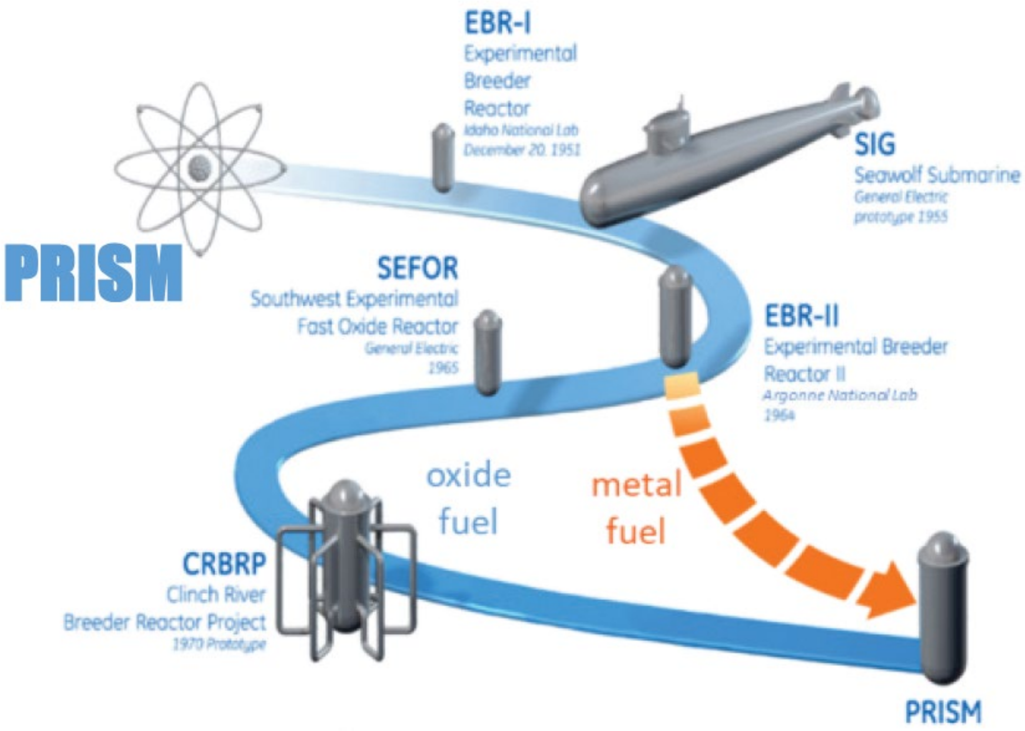
May 29, 2019

ARC - GEH Confidential

- Agreement between ARC and GEH related to ARC-100 Technology Development for global nuclear power generation, desalination, and industrial heat markets.
  - Recognizes the two reactor designs focus on different objectives and markets
    - ARC-100 is a nominal 100-200 MWe SMR which is designed for efficient and flexible electricity generation, operating for 10-20 years without refueling.
    - GEH PRISM with a capacity of approximately 165-311 MWe is refueled every 24 months.
  - Leveraging GEH Intellectual Property to accelerate ARC-100 Preliminary & Detailed Design
  - Design has progressed through several Regulatory Body reviews (U.S. NRC & UK ONR)



GEH Evolutionary Path



Advanced Liquid Metal Reactor (ALMR, 1994-1995)

- PRISM design, initiated in the early 1980s, used as reference for DOE ALMR Program
- Submitted six-volume Preliminary Safety Information Document (PSID)
- NUREG 1368 NRC issued Preapplication Safety Evaluation Report

  "no obvious impediments to licensing"

Global Nuclear Energy Partnership (GNEP, 2007-2009)

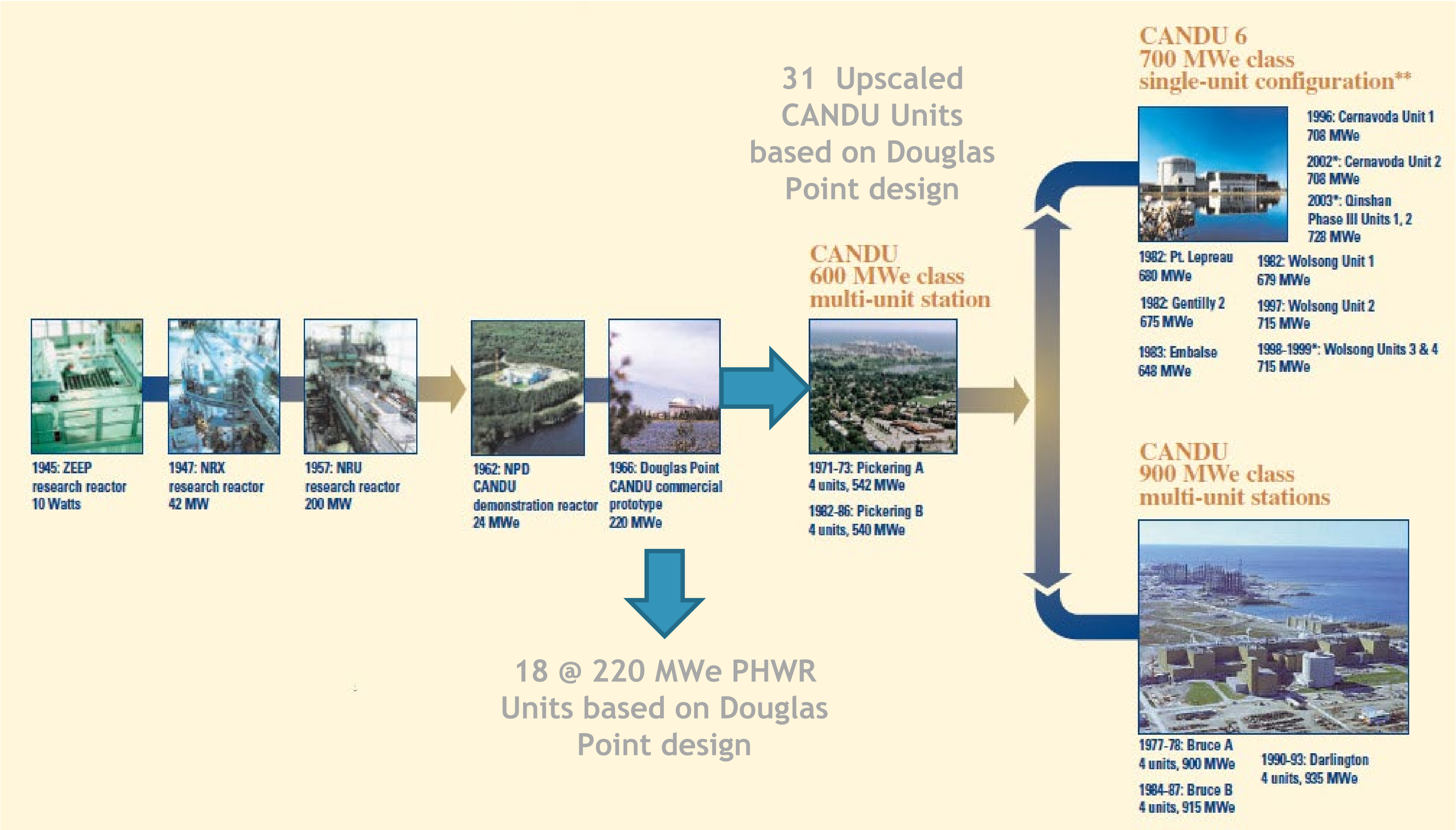
- International cooperation program … closing the fuel cycle
- Submitted PRISM preliminary Design Control Documents (DCD) which NRC docketed for training for advanced reactor licensing

  "found to be of high technical quality"

Probabilistic Risk Assessment (PRA, 2016)/Licensing

- Modernization Program (LMP, 2018)
- GEH developed PRISM PRA with Argonne National Laboratory (ANL)
- PRISM PRA used for table-top demonstration of Southern-led, DOE supported, risk-informed licensing approach for Gen IV reactors

Canadian Example - CANDU Evolutionary Path



Example of a design that has gone through an evolution from concept to commercial demonstration that then expanded to a fleet of Reactors

NOTE: the 24 MWe commercial demonstration was 1st scaled up successfully to 220 MWe (~ 10 x) and then all the way up to 935 MWe (~ 40x).

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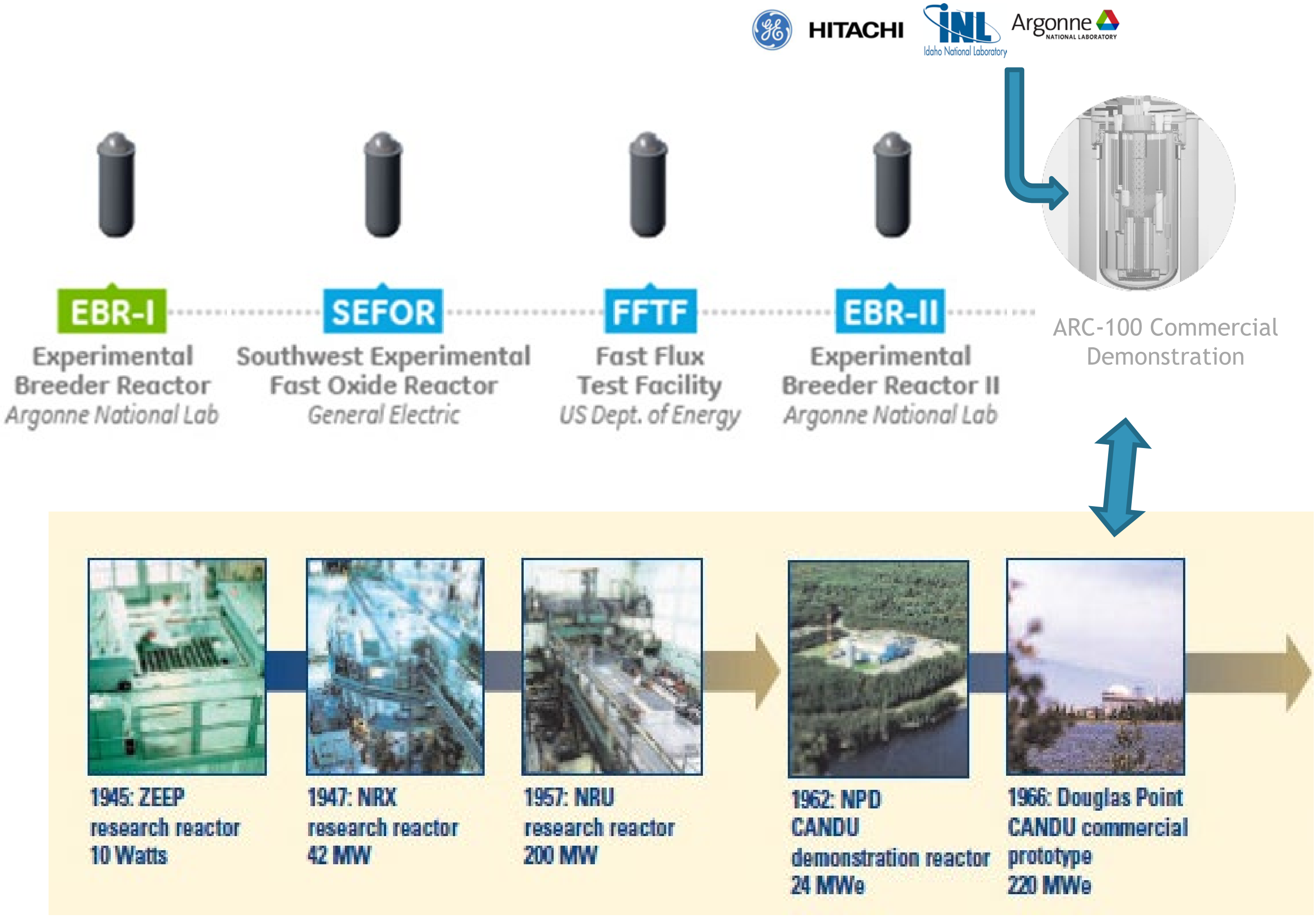
ARC-100's Evolutionary Path

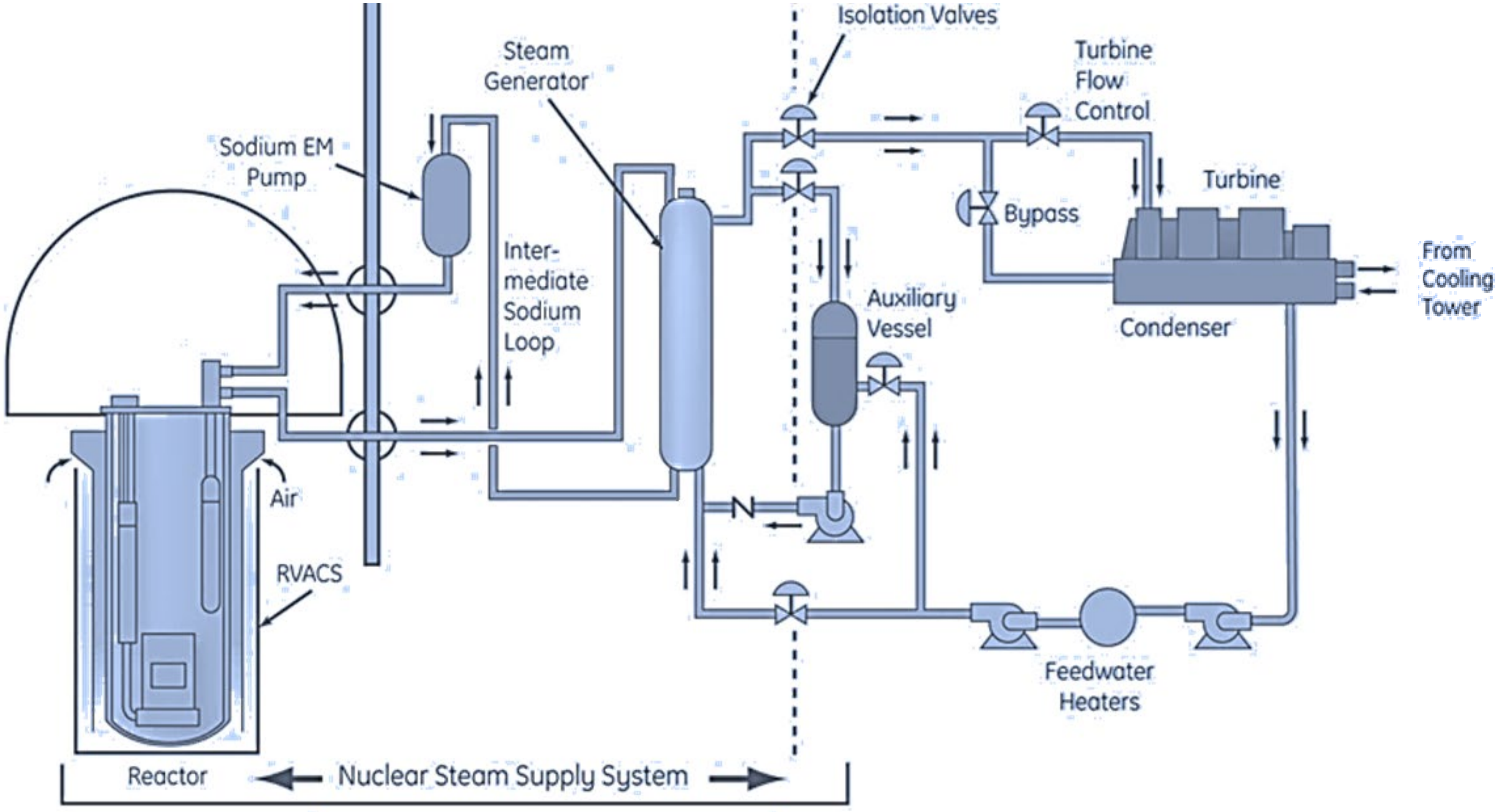


The path of the ARC-100 to a commercial demonstration has the same type of evolution as the CANDU Reactors did in Canada with incremental increases in capacity and dedicated units to fuel qualification.

NOTE: From the EBR-II of 20 MWe to the ARC 100 MWe commercial demonstration it is a scale up of 5x, and from FFTF of 400 MWt to the ARC thermal design of 286 MWt it is actually not a scaling up of thermal power of the reactor but within the envelope.

Scale up factor to unit size for commercial demonstration from EBR-II/FFTF to ARC-100 not significant in comparison to the past scaling factors in past successful Canadian nuclear development.

ACR-100 Design Overview

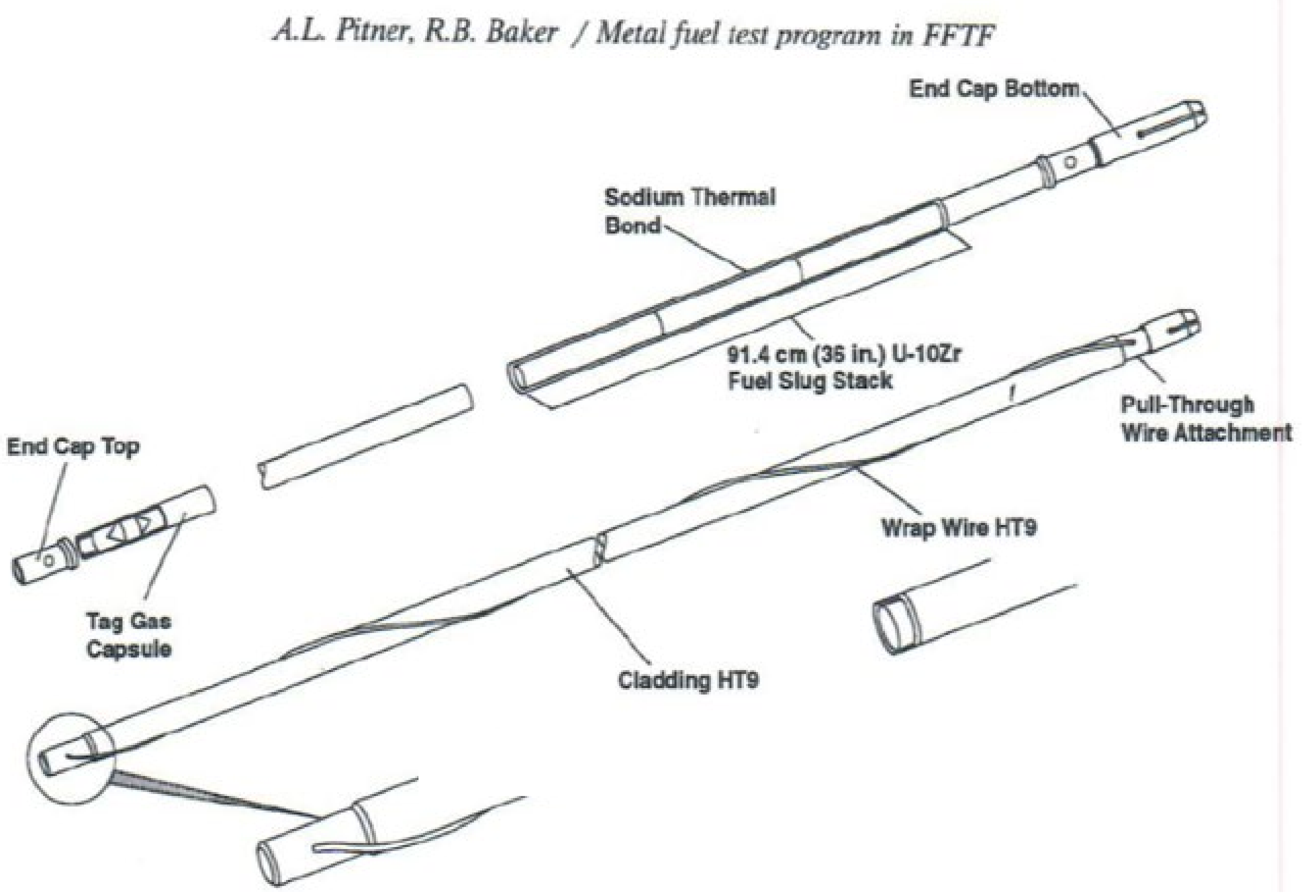


MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer, country of origin	ARC Nuclear Canada, Inc., Canada
Reactor type	Liquid metal cooled fast reactor (pool type)
Coolant/moderator	Sodium
Thermal/electrical capacity, MW(t)/MW(e)	286 / 100
Primary circulation	Forced circulation
NSSS Operating Pressure (primary/secondary), MPa	Non- pressurized
Core Inlet/Outlet Coolant Temperature (ºC)	355 / 510
Fuel type/assembly array	Metal fuel (U-Zr alloy) based on enriched uranium
Number of fuel assemblies in the core	99
Fuel enrichment (%)	Avg. 13.1
Core Discharge Burnup (GWd/ton)	77
Fuel Cycle (years)	20
Reactivity control mechanism	Control Rods
Approach to safety systems	Passive, diverse, redundant
Design life (years)	60
Plant footprint (m^2)	56 000
RPV height/diameter (m)	15.6 / 7.6
Distinguishing features	Inherent reactor safety with passive, diverse and redundant decay heat removal. Core lifetime of 20-years without refueling.

Sodium Bonded U-10Zr Fuel Pin

Graphic is FFTF fuel

- High reliability based on OPEX
- Fuel pin dimensions (length, diameter) vary based on reactor core design
- ARC-100 active fuel length = 150 cm
  - Comparable to length in blanket assemblies in EBR-II/FFTF
- Gas plenum to fuel ratio = 1.5



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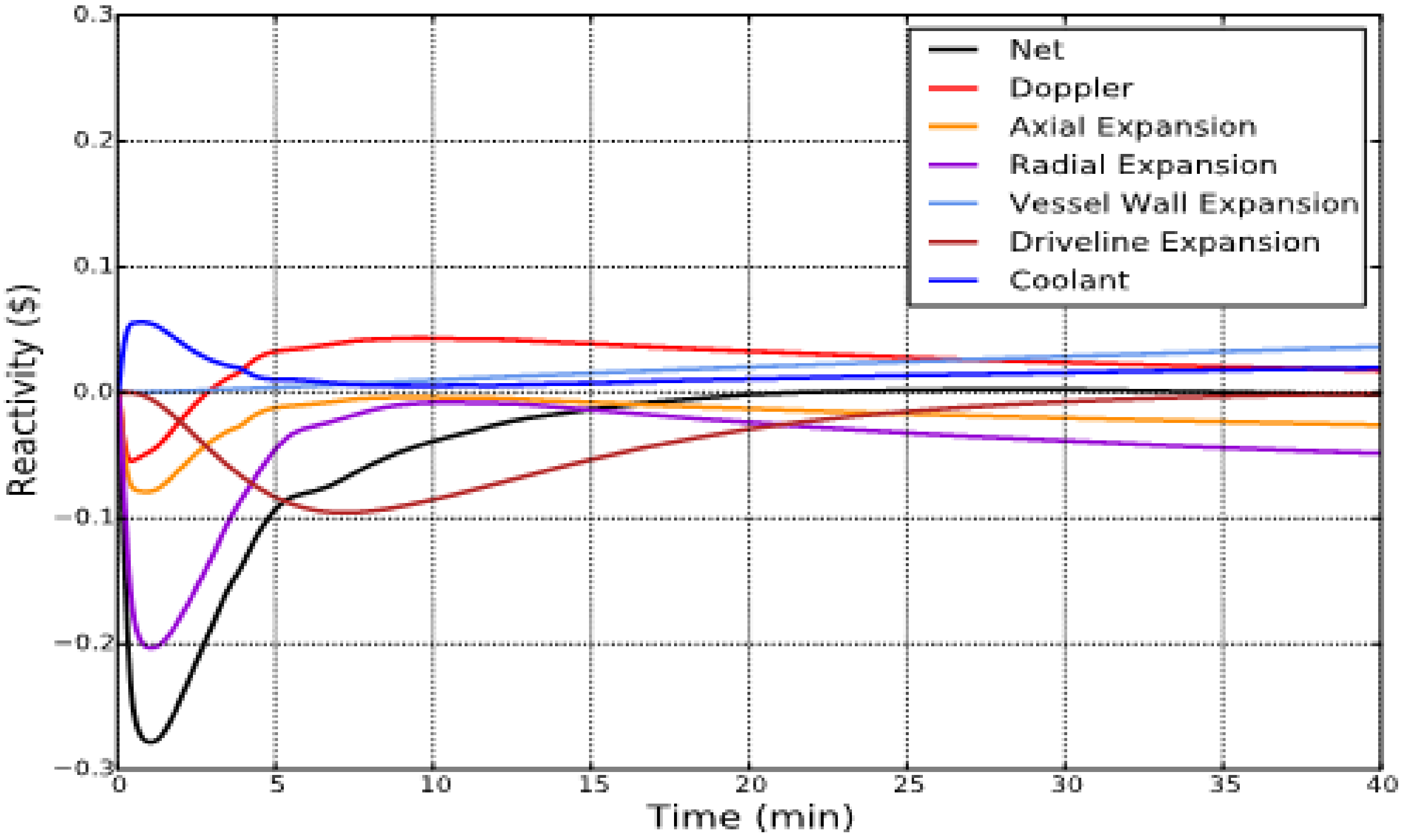
Inherent Safety

- Unprotected Station Blackout
- Loss of power to all pumps
- Failure of RPS to scram the control rods
- Large negative reactivity from radial expansion and axial expansion feedbacks counters increasing power-to-flow ratio



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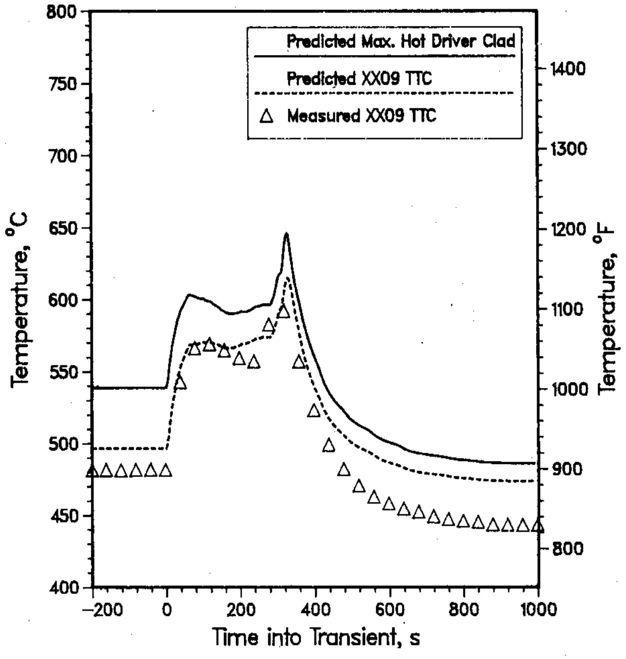
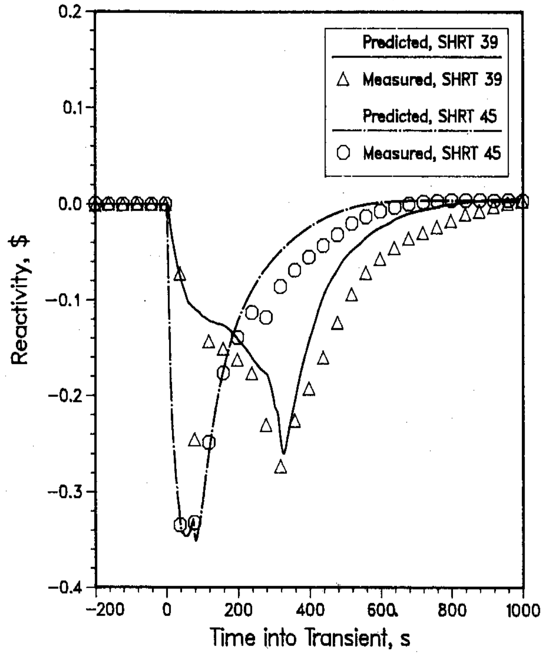
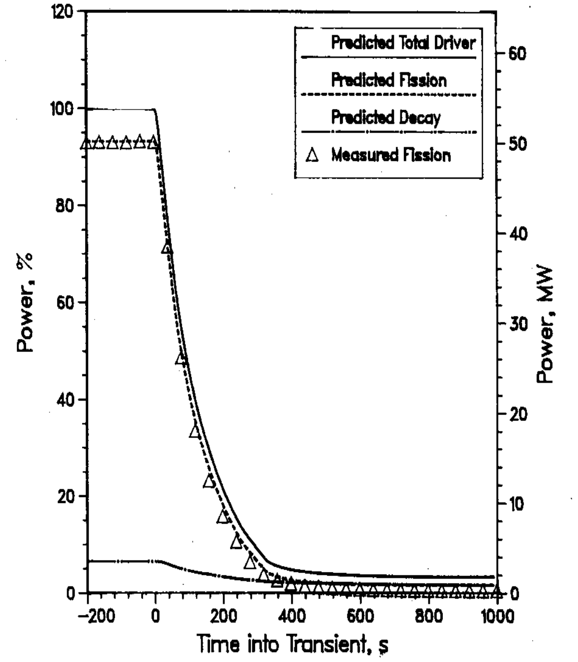
EBR-II's "Inherent Safety" Demonstration

https://www.youtube.com/watch?v=Sp1Xja6HlIU







Event causing temperature rise

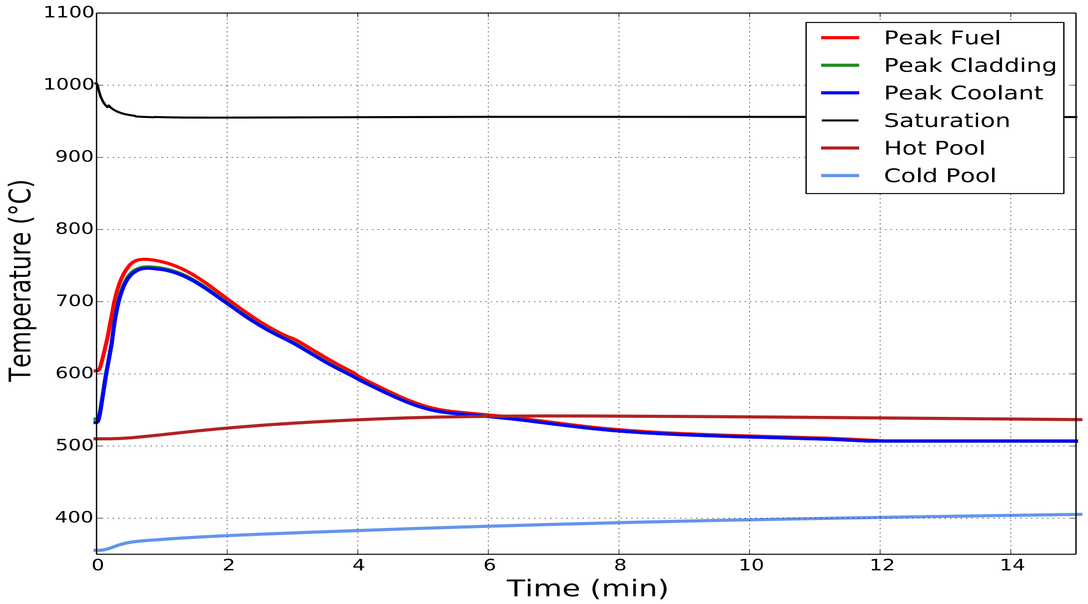
Inherent introduction of negative reactivity

Safely reducing reactor power to decay heat levels

The Unprotected Station Blackout Does Not Damage the Plant

- In-core temperatures peak within first minute
- Decrease below pre-transient temperatures as power decreases
- Large fuel melting and sodium boiling margins maintained
- Clad temperatures rise barely above slow eutectic threshold
- Hot pool temperature peaks at 542°C below Service Level D limit of 704°C

Margin	Nominal	UTOP
Peak Fuel Temp	605°C	759°C
Fuel Melting Margin	596°C	442°C
Peak Cladding Temp	543°C	748°C
Peak Sodium Temp	539°C	750°C
Sodium Boiling Margin	463°C	190°C

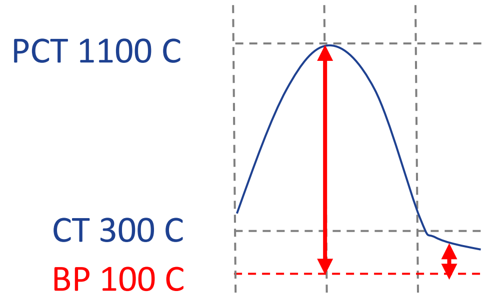


Beyond Design Basis Accidents

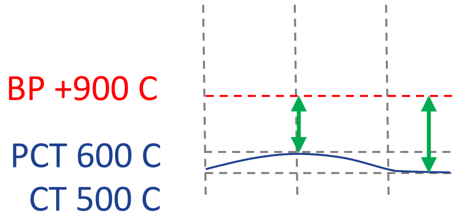
- Severe accidents, those that would lead to fuel damage, are precluded in the ARC-100 by virtue of the self-protecting response of the reactor.
- Anticipated Transients Without Scram (ATWS) do not lead to fuel damage as demonstrated by actual tests in the EBR-II.
- Regardless, tests have been undertaken in the TREAT facility by created a cladding breach with a burst of 4 times power that also demonstrated strong negative reactivity.
- In the event of Beyond Design Basis Accident
  - Fuel retains the majority of fission products
  - Sodium scrubs a significant amount of those which might be released from fuel

The selection of metallic fuel, in combination with sodium coolant, results in Inherently Safe Characteristics

Oxide fuel



Metallic fuel

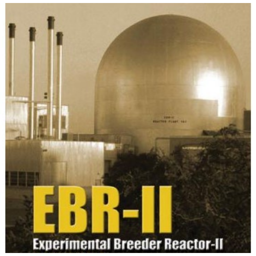


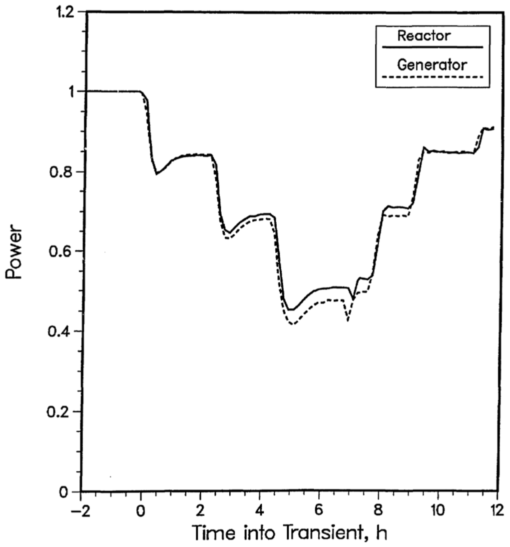
Strong heat transfer
- Metal fuel
- Metal cladding
- Metal coolant

PCT = Peak Centerline Temperature
CT = Cladding Temperature
BP = Boiling Point *(atmospheric)*

Load Following

Demonstrated

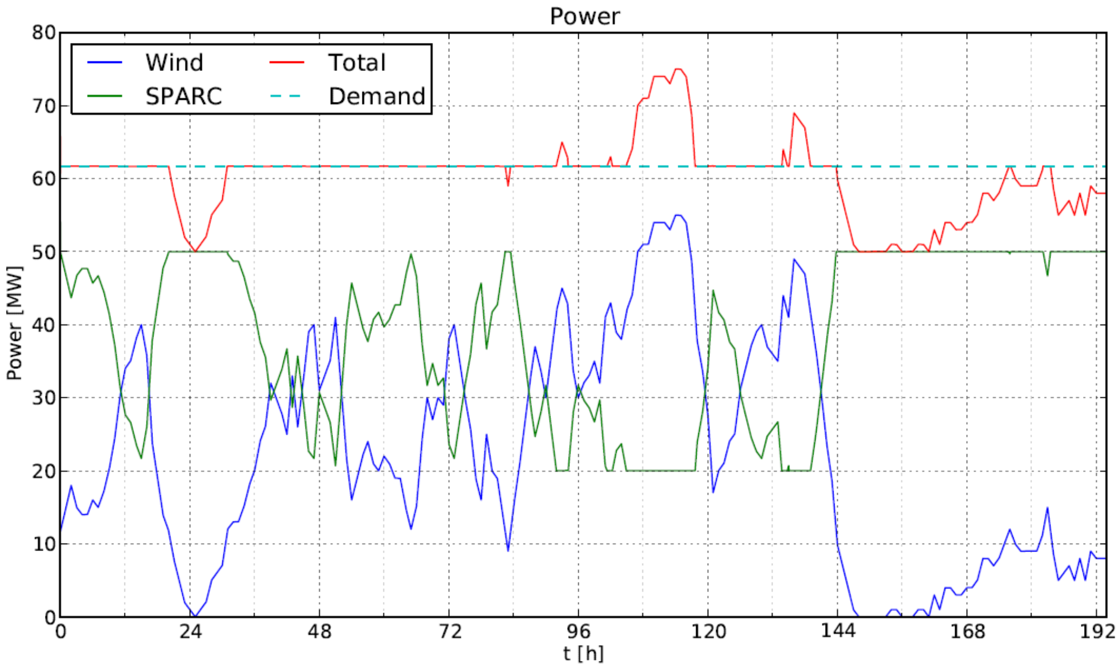




- Demonstrated a SFR with metal fuel can be passively controlled over a large power range.

Simulated

UPPSALA UNIVERSITET

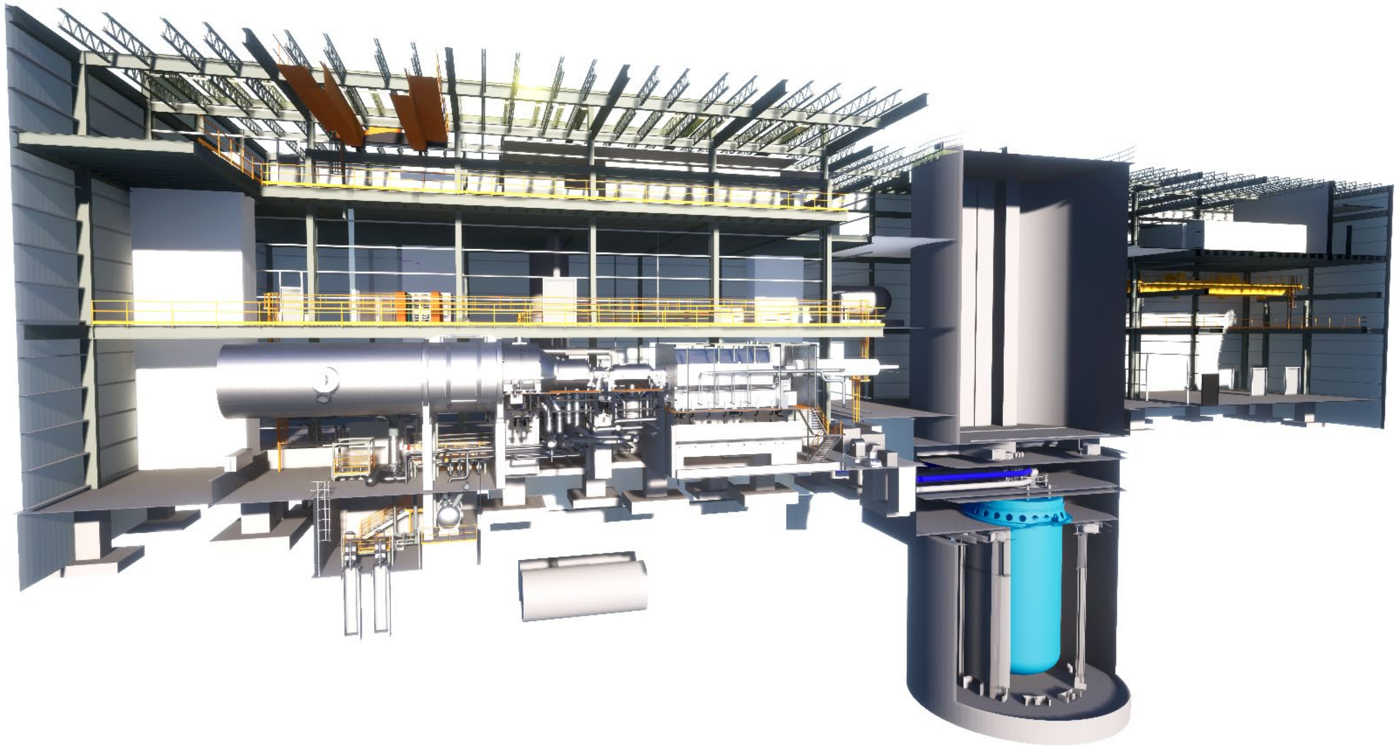


- Dynamic simulation of the reactor, secondary circulation and steam generator. The reactor was able to load follow at 6% of rated power per minute between 100% and 40%.
- Fast reactors are not affected by Xenon poisoning versus WRs.

DEMONSTRATION OF EBR-II POWER MANEUVERS
WITHOUT CONTROL ROD MOVEMENT - DE88 010011
By L. K. Chang, D. Mohr, H. P. Planchon, E. E. Feldman, and
N. C. Messick, Argonne National Laboratory

Load following with a passive
reactor core using the SPARC design
Sebastian Leo Eile Svanström
UPTEC ES 16 023
13 Juni 2016

ACR-100 Layout

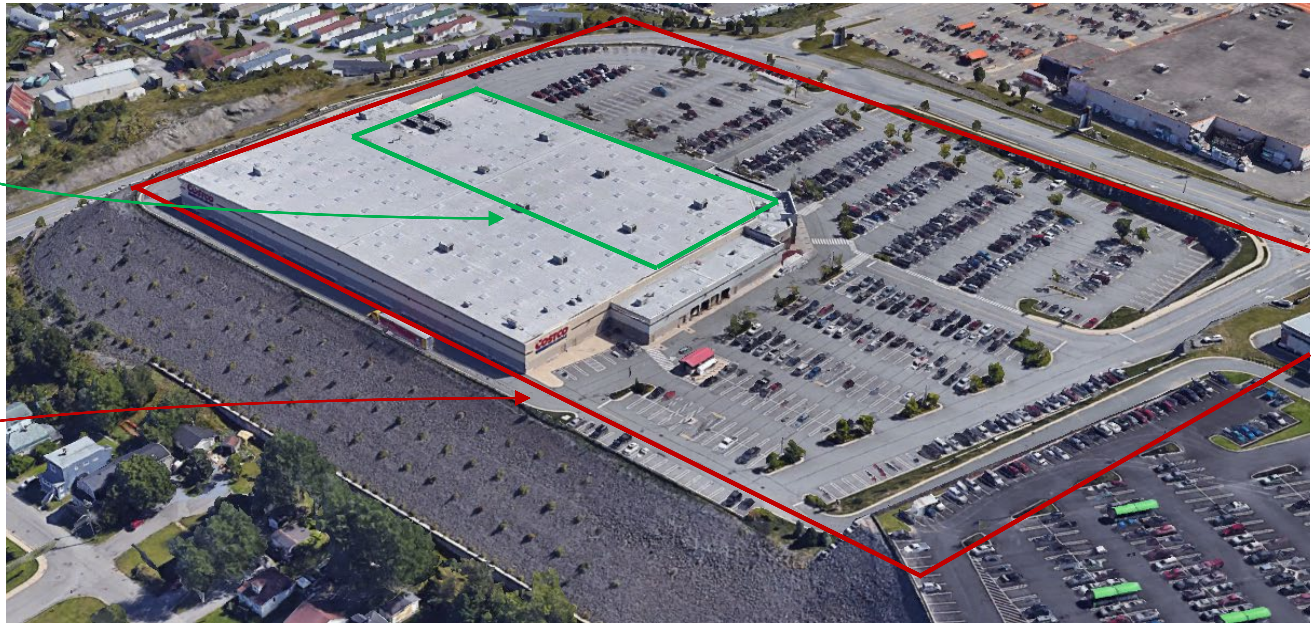


Relative Size – Discount Warehouse Store

ARC-100 Site

- Overall Site 190 m x 285 m = 54,000 m^2
- Main Building 50 m x 108 m = 5,400 m^2
- All Buildings included 10,075 m^2

Discount Warehouse Store -Saint John



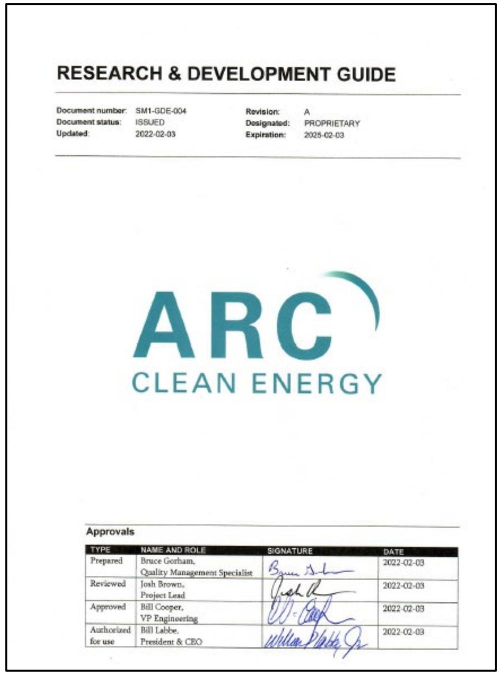
- Overall Site 210m x 265m = 55,650 m^2
- Main Building 100 m x 130 m = 13,000 m^2

Technical Developments

- Technology Readiness Level
- Technology Readiness Assessment
- Technical Optimization Activities
- Engineering Partners
- Design Synergies
- Integrated Energy Systems

Technology Readiness Level (TRL)

SM1-GDE-004, Research and Development Process Guide



RESEARCH & DEVELOPMENT GUIDE

Document number:	SM1-GDE-004	Revision:	A
Document status:	ISSUED	Designated:	PROPRIETARY
Updated:	2022-02-03	Expiration:	2025-02-03

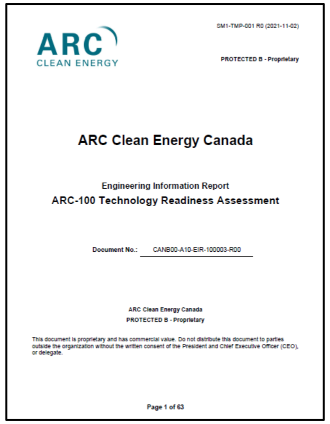
ARC CLEAN ENERGY

Approvals

TYPE	NAME AND ROLE	SIGNATURE	DATE
Prepared	Bruce Gorham, Quality Management Specialist		2022-02-03
Reviewed	Josh Brown, Project Lead		2022-02-03
Approved	Bill Cooper, VP Engineering		2022-02-03
Authorized for use	Bill Labbe, President & CEO		2022-02-03

State	TRL	Description of Technology Maturity
Basic Research and Development	1	Basic principles observed and reported
	2	Technology concept and/or application formulated
	3	Analytical and experimental critical function and/or characteristic proof of concept
Engineering-scale development and demonstration	4	Component and/or system validation in laboratory environment
	5	Laboratory scale – similar system validation in relevant environment
	6	Engineering/pilot-scale – prototypical system validation in relevant environment
Commercial demonstration and deployment	7	Full-scale, prototypical system demonstrated in relevant environment
	8	Actual system completed and qualified through test and demonstration
	9	Actual system operated over the full range of expected conditions

- TRL Threshold (TRL_T) is ≤ 5
- This is the technology readiness level which establishes the boundary between Research & Development and Demonstration & Deployment readiness states.


SM1-TMP-001 R0 (2021-11-02)

PROTECTED B - Proprietary

ARC Clean Energy Canada

Engineering Information Report

ARC-100 Technology Readiness Assessment

Document No.: CANB00-A10-EIR-100003-R00

ARC Clean Energy Canada

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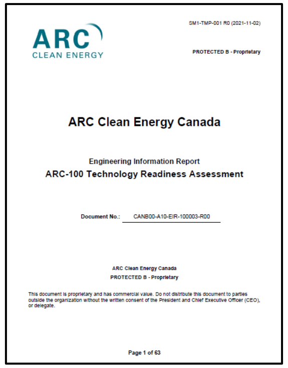
Technical Readiness Assessment

	October 2015	June 2018	June 2022	
	PRISM [1]	**SFR by Early 2030s[2]**	**ARC-100**	**SYSTEMS**
Nuclear Heat Supply	5			
Fuel Element (fuel, cladding assembly)	5	7-8	**8**	J10, J11
Reactor Internals	6	7	**6**	F14
Reactor Control	6	7	**7**	C12
Reactor Enclosure	5	7	**6**	B11
Operations/Inspection/Maintenance	5	6	**6**	A72, A80, F21, T71, U95
Core instrumentation	5,3	6	**8 (4 for Optional individual Fuel Assembly Duct Flow)**	C01, C02, C03
Heat Transport	4			
Coolant Chemistry Control/Purification	6	6	**8**	M21, M31
Primary Heat Transport System	6	6	**8 (4 for Option for in-vessel EM Pumps)**	B21
Intermediate Heat Exchanger (component)	6	7	**8 (3 for Option for Kidney Shaped Hx)**	B21
Intermediate Heat Transport System	NA	NA	**7 (4 for Option of in-line cooled EM Pumps)**	B22
Sodium Valves/Piping/Heating	4	6	**8**	M50
Auxiliary cooling (RVACS)	NA	NA	**7**	E11
Residual Heat Removal (DRACS)	5	6	**7**	E23

Technical Readiness Assessment (cont.)


ARC CLEAN ENERGY

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Engineering Information Report

ARC-100 Technology Readiness Assessment

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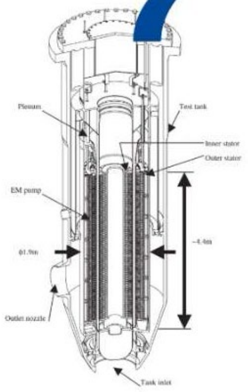
Power Conversion	7			
Turbine/Reheater/Condenser	7	8	**8**	N31, N21, N22, N25, N61
Reheater/Condenser	7	NA		
Steam generator	7	7	**7**	B23
Pumps/Valves/Piping	7	7	**8**	
Electrical System	NA	NA	**9**	R01, R02, R03
Balance of Plant	4			
Fuel handling and Interim Storage	4	5-7	**6**	F15, F42
Waste heat rejection	6	NA	**8**	W24
Instrumentation and Control	6	6	**8**	
Radioactive waste management	6	6	**6**	K10, K20, K40
Safety	6			
Inherent (passive) safety features	6	6	**6**	A17, A21
Active safety system	6	6	**7**	C12/C42
Fire Protection Systems (Sodium/Non-Sodium)	NA	NA	**8**	M43, P16
Licensing	3			
Safety Design Criteria and Regulations	3	4	**4**	
Licensing Experience	3	3	**3**	
Safety and Analysis tools	5	7	**6**	
Safeguards	3			
Proliferation resistance – intrinsic design features (e.g., SNM accountability)	3	NA	**3**	
Plant Protection – intrinsic design features	3	NA	**8**	C95, Y86

Technical Optimization Activities

Primary & Intermediate Loop Pumps

- Base Case is with Centrifugal Pumps with higher TRL.
- Option for self cooled Electromagnetic Pumps for both in-vessel and in-line (GEH R&D) – being done in tandem to Natrium/VTR effort to eliminate moving parts.
- Already have tested EM Pump that is 6.5 times larger than our needs, so downsizing exercise.

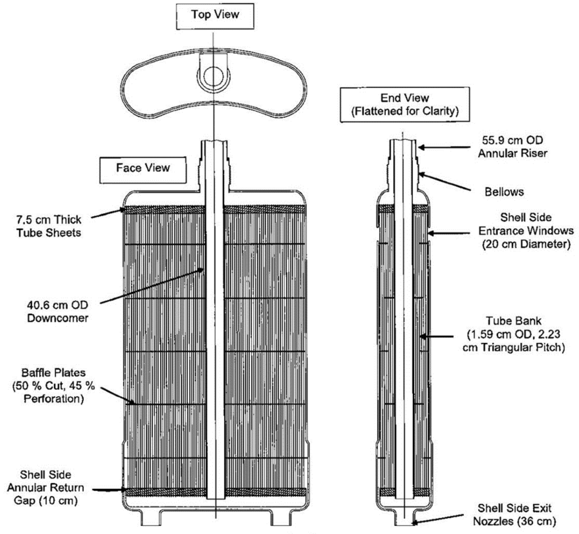


EM Pump Structure

Installation of EM Pump at Test Facility

Intermediate Loop Heat Exchanger

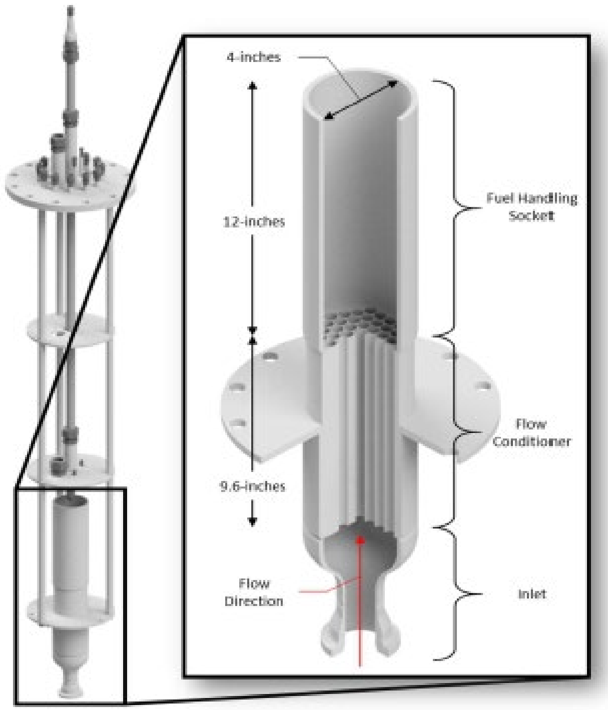
- Base Case is with Round Heat Exchanger with higher TRL.
- Option for Kidney Shaped Heat Exchanger (GEH) – being done in tandem to Natrium/VTR efforts to reduce external piping and penetrations in Pool Reactor Cover.
- Already fully modelled for PRISM-S design, so also downsi

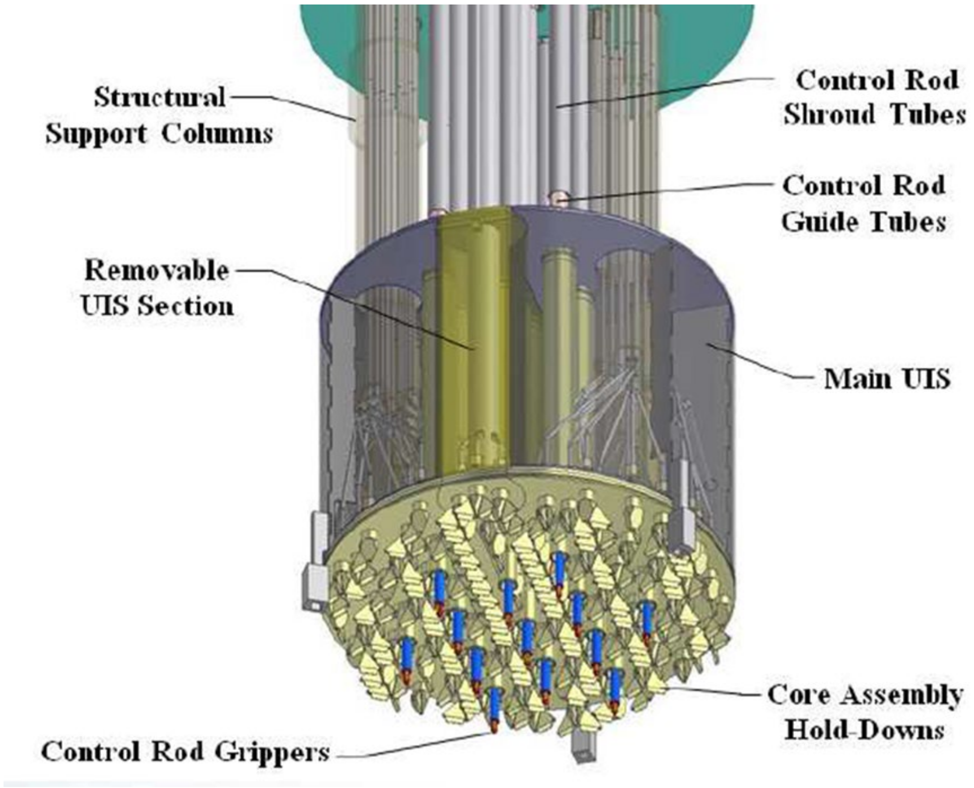


Technical Optimization Activities

Fuel Assembly Low Flow Detection

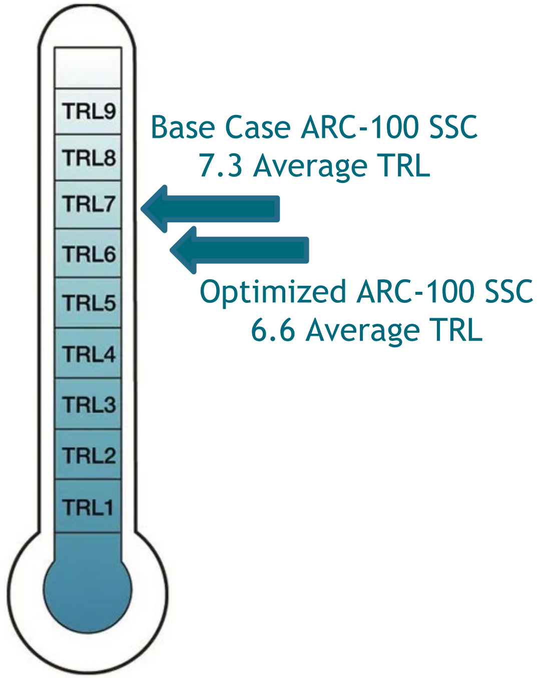
- Base Case is with Thermocouples on grouped Fuel Assemblies.
- Option for individual Flow Senser located in the Upper Internal Structure (UIS) for each Fuel Assembly as defense in depth in conjunction with Thermocouples (ANL R&D) – being done in tandem to VTR effort.
- “Full-Scaled” SFR fuel handling socket designed to simulate flow exiting a single subassembly for the study flow sensors in a single-jet configuration.





TRL Assessment Observations

- All System, Structure & Component (SSC) TRL Levels at Demonstration & Deployment readiness states
- Several SSC Options for Operational and Maintenance Optimization in Development Stage.
- No SSC's in Research Stage.
- Licensing & Safeguards TRL Threshold (TRL_T) is ≤ 5 due to first time Licensing in Canada of the SFR Technology.
  - Mitigation activities in progress:
    - CNSC VDR Process
    - NB Power Pre-Licensing Activities
    - CNSC - USNRC Cooperation Agreement
    - Generation IV International Forum Safety Design Criteria for SFR Technology



Engineering Partners – Areas of Planned Focus

GE HITACHI	Nuclear Steam Supply System Design	HATCH	AE & Balance of Plant Design
Argonne National Laboratory	Reactor Physics & Thermohydraulics	UNITED	Reactor Building Civil Structural Design
INL Idaho National Laboratory	Fuel Fabrication Technology Experts	Worley energy | chemicals | resources	Cooling Options Study & Management System Development
Canadian Nuclear Laboratories	Fuel Pin Fabrication Demonstration	Laurentis Energy Partners	Fuel Handling Design & In-Service Inspection/Aging Management Program
KINECTRICS	Probabilistic & Deterministic Safety Analysis	FUNDY Engineering	Project Description for Environmental Impact Assessment
CNSC CCSN	Vendor Design Review Phase II	tlg SERVICES	Decommissioning Plan

Design Synergies

EBR II

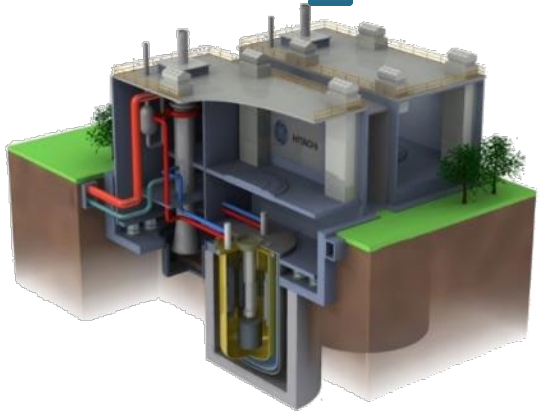
- 20 MWe
- Variable fuel cycle
- Test/demonstration

✓**Pool design**
✓**Metal fuel**
✓**Passive safety**
✓**Sodium cooled**
✓**Fast neutron**
✓**Small**

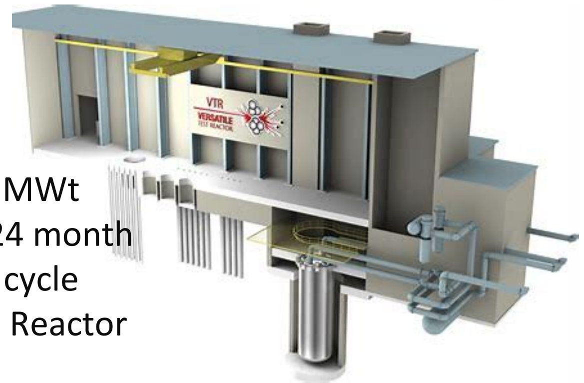
Current Design Projects

HITACHI
PRISM

- +300 MWe
- 12-24 month fuel cycle

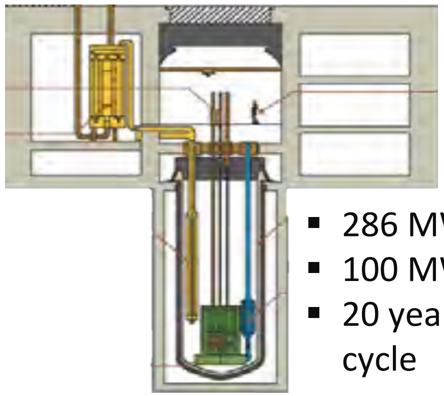


U.S. DEPARTMENT OF ENERGY



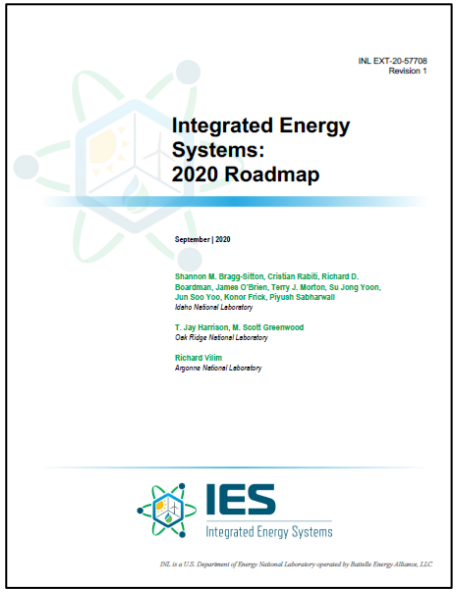
- 300 MWt
- 12-24 month fuel cycle
- Test Reactor

ARC CLEAN ENERGY
ARC-100



- 286 MWt
- 100 MWe
- 20 year fuel cycle

Integrated Energy Systems


INL EXT-20-57708
Revision 1

Integrated Energy Systems: 2020 Roadmap

September | 2020

Shannon M. Bragg-Sitton, Cristian Rabiti, Richard D. Boardman, James O'Brien, Terry J. Morton, Su Jong Yoon, Jun Soo Yoo, Konor Frick, Piyush Sabharwall
Idaho National Laboratory

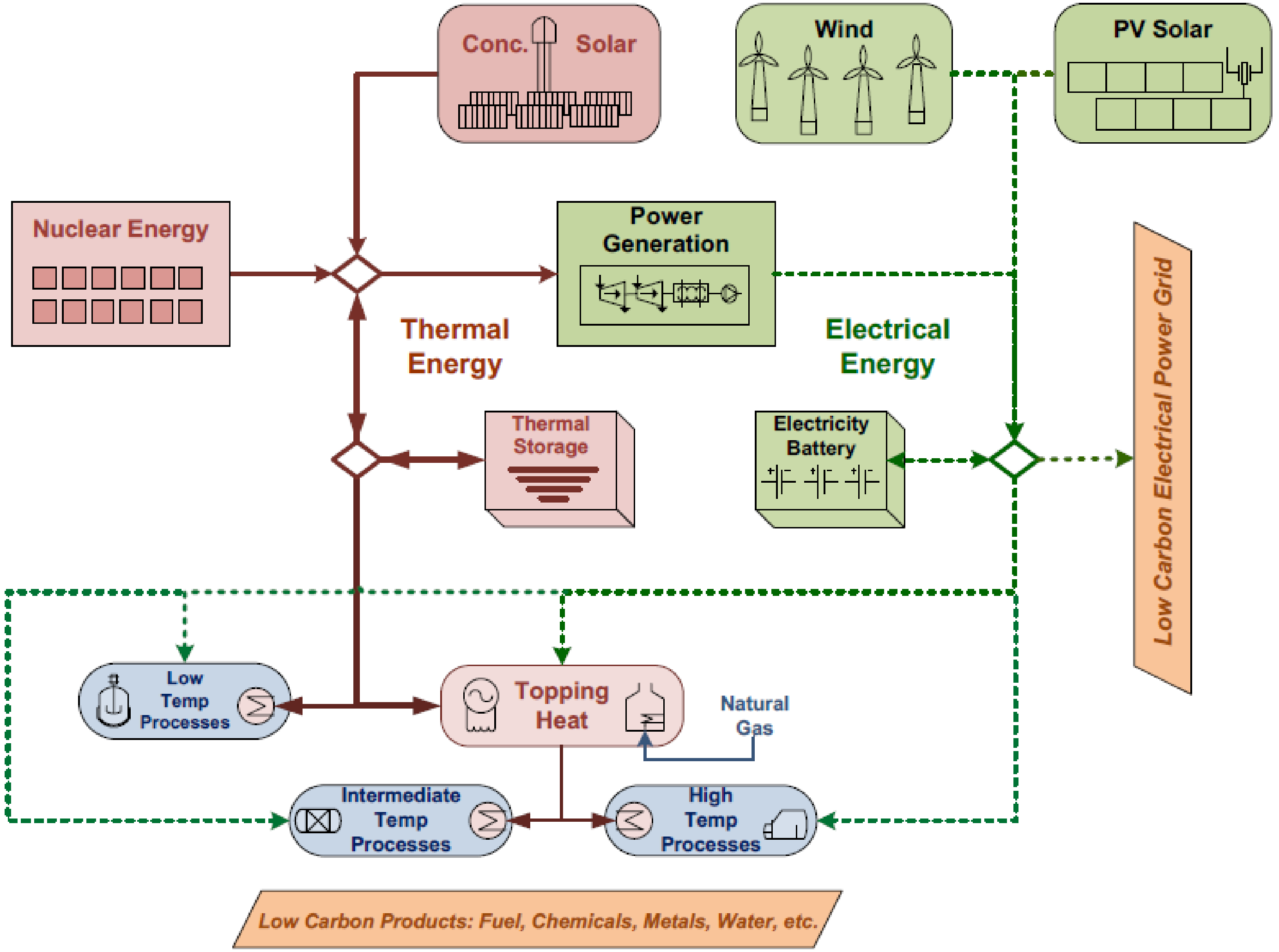
T. Jay Harrison, M. Scott Greenwood
Oak Ridge National Laboratory

Richard Vilim
Argonne National Laboratory

IES
Integrated Energy Systems

INL is a U.S. Department of Energy National Laboratory operated by Battelle Energy Alliance, LLC

- Topping heat may or may not be necessary for intermediate and high temperature processes as a function of the outlet temperature of the selected nuclear reactor technology




Thank you
arc
CLEAN TECHNOLOGY

Meet the Presenter

Mr. Patrick Alexander started his career and developed his passion for the nuclear industry as a Reactor Operator on a Nuclear Submarine. He earned his Bachelor of Science degree in Nuclear Energy Engineering Technology from Thomas Edison State University and a Master in Engineering Management from the University of Texas at Arlington. He joined the Commercial Nuclear industry at Comanche Peak Nuclear Power Plant as an I&C Technician and ultimately became a Senior Reactor Operator, Shift Technical Assistant and a qualified Shift Manager. His passion for the future of nuclear power brought him to TerraPower as Principal Engineer for Operations, and where he now serves as the TerraPower Operations Manager.

Email: palexander@terrapower.com

NATRIUM

a TerraPower & GE-Hitachi technology

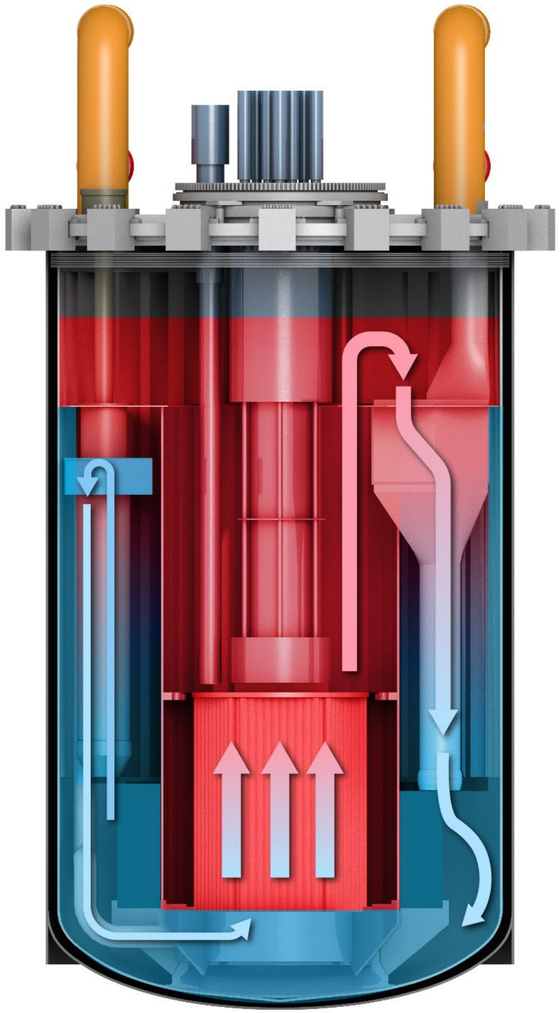
Utilizing the Past to Realize the Future

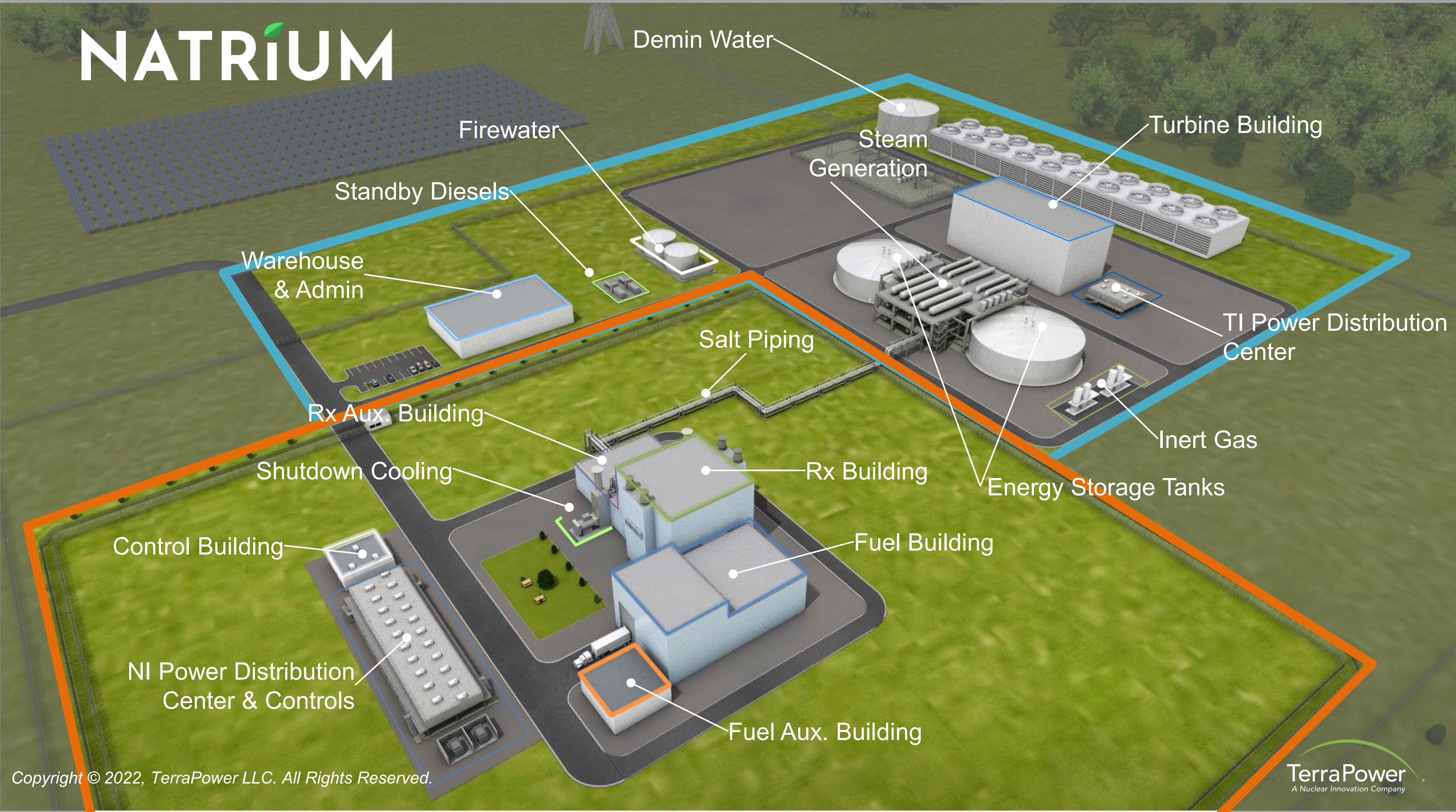
TerraPower
A Nuclear Innovation Company

What's Different?

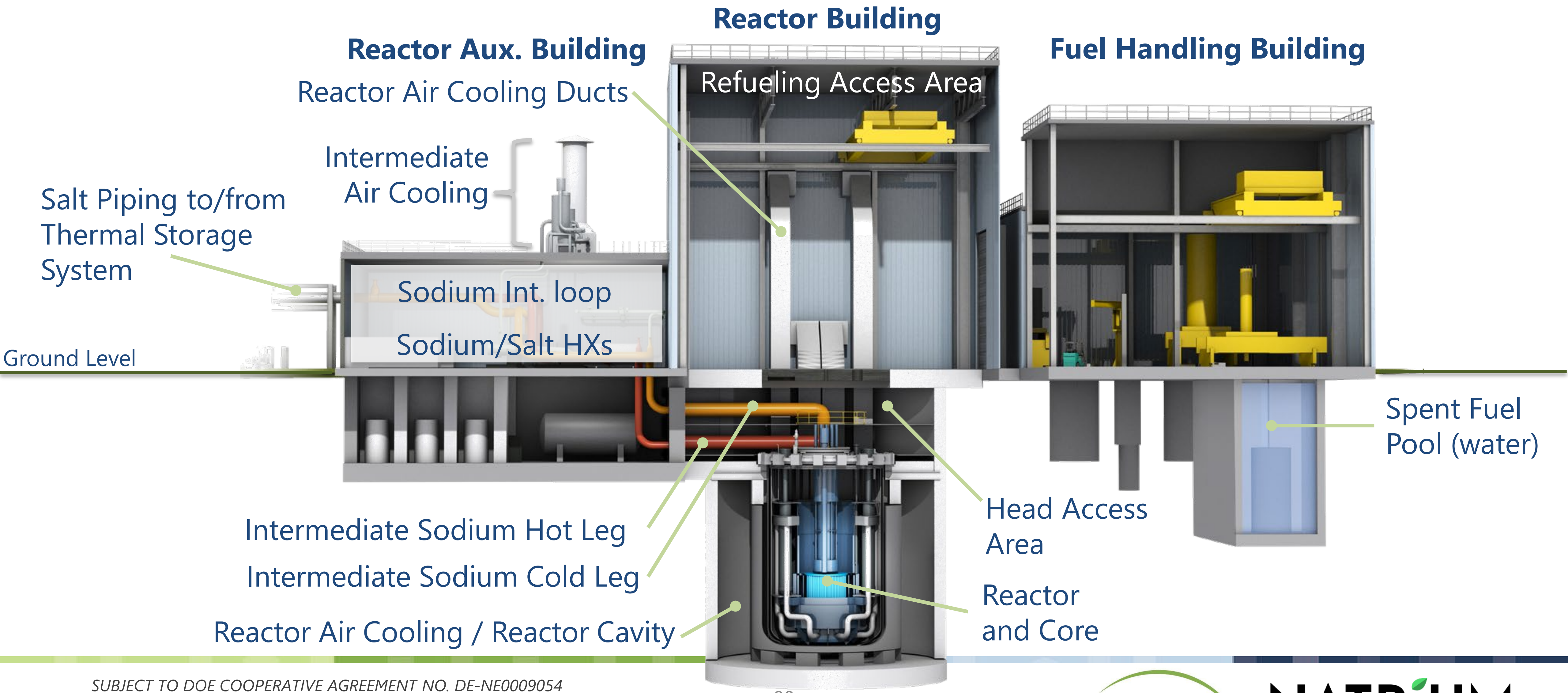
Leverage inherent features: drive down cost

- Compact systems, less "nuclear sprawl"
- Low pressure
- Efficient heat transfer
- Pool design with large coolant inventory
- EI/NI Separation
- Modularity
- Parallel Construction
- Emergency Planning Zone Reduced




NATRIUM
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
TerraPower
A Nuclear Innovation Company

Plant Overview

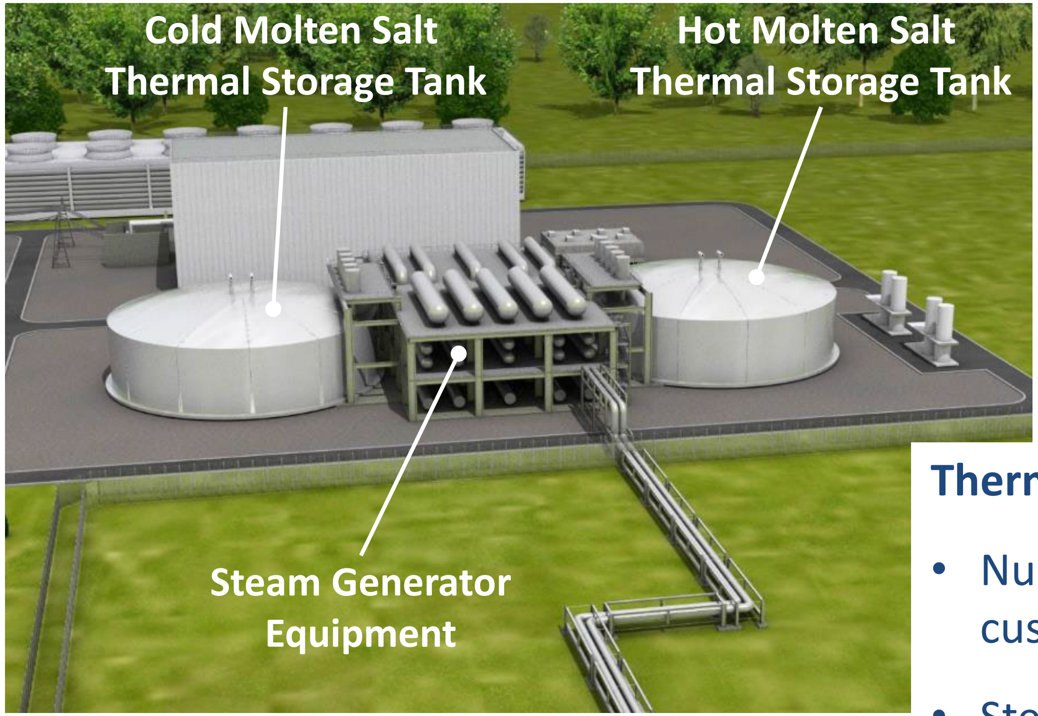


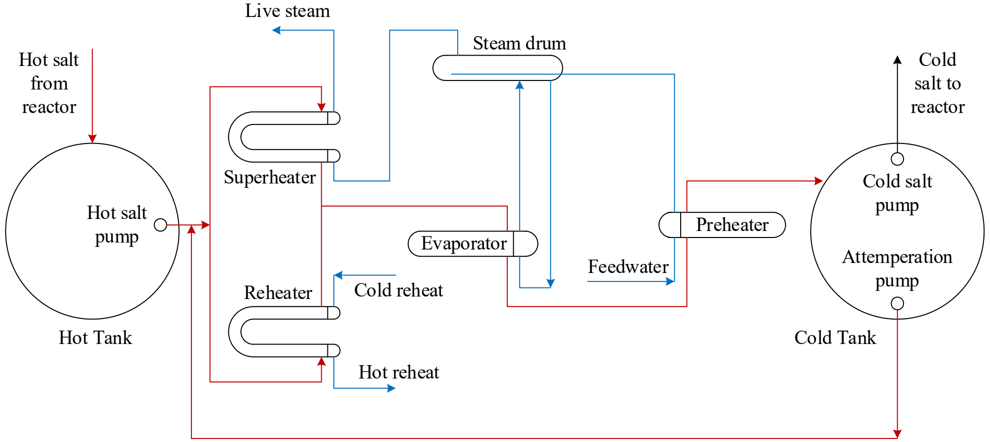
SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

TerraPower NATRIUM

Energy Island Thermal Storage

Above picture is of a Solar Salt Plant



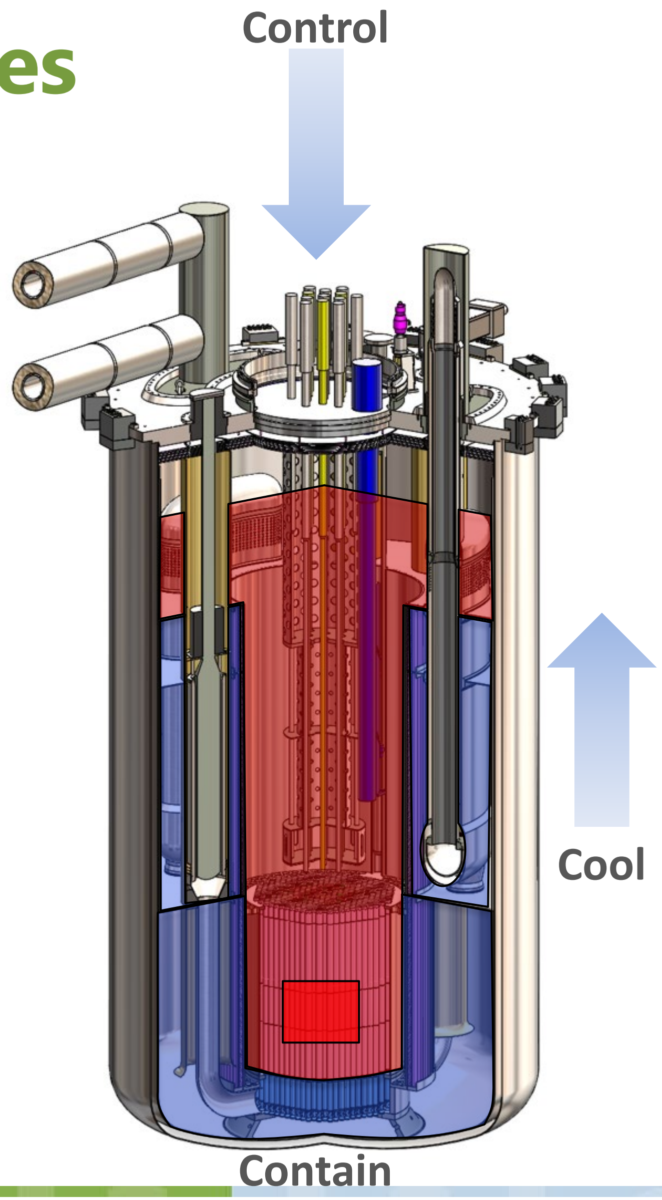


Thermal Storage

- Number of tanks based on customer's energy need
- Steam generator trains based on size of turbines
- Turbine size based on customer's power need
- Proven Technology

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 SR Operator Actions or SR AC power required for Safe shutdown
- 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
- 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

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TerraPower NATRIUM

Knowledge Transfer

There are three methods for effective knowledge capture employed on the Natrium project:

1. Utilizing personnel who have previous SFR experience
2. Reviewing previous SFR design documentation, OE, and Lessons learned
3. Strategic Partnerships

Personnel with SFR experience (Current and Past)

- Denny Newland –
  - SFR Experience: Started as Operations Engineer after College at FFTF. Moved up to Shift Operations Manager, Assistant Operations Manager, Program Manager, Assistant Plant Manager, Plant Manager, and the LMFBR Program Manager.
  - Natrium: Assisted in the development of the operational concepts and programs for the Natrium plant.
- John Truax –
  - SFR Experience: Started at Westinghouse on Sodium System designs for FFTF. Joined FFTF as an Operations Supervisor and shift testing leader for commissioning. Shifted to a leadership role as a Technical Support Manager managing plant chemistry, systems analysis, and the experiment review committee. Later become the Outage manager in charge work planning and execution and the refueling group and was the dry cask program owner. Later helped to established decommissioning plant
  - Natrium: Assisting in refueling system development and input into several operational programs including commissioning.
- Dave Lucoff –
  - SFR Experience: 40 years of experience in Sodium Fast Reactors including Core Manger for FFTF and worked at Handford Site as Core designer, safety tester, and Operations Manager.
  - Natrium: Assisting the development of the Natrium Fuel program and systems.
- Craig Smith –
  - SFR Experience: Operator at FFTF, learned and studied all the sodium systems and maintained the plant during decommissioning phase.
  - Natrium: Training Program owner responsible for developing Natrium training.
- Owen Nelson –
  - SFR Experience: Reactor Operator at FFTF operating the reactor plant systems while in operation
  - Natrium: Assisting in development of Re-fueling systems in Natrium plant.

Knowledge Transfer

- Knowledge from these SFR experts is transferred through:
  - Mentorship
    - Direct transfer to a Jr engineer
  - Direct input into design
  - Documentation
    - Document strategies used from previous SFRs that apply to Natrium

SFR Historical Documents

- TerraPower has a database of historical SFR documents:
  - FFTF
  - EBR-II
  - Prism
  - CRBR
  - Fermi-1
  - Phenix
  - SuperPhenix
  - SNR-300
  - Monju

- This includes Design documents, Operating Experience, and Lessons Learned.

SFR Historical Documents

- Prior to beginning work on Natrium System design or program development Natrium Engineers review historical reference repository related to their system.
  - Provides good basis for design
  - Provide past problems and their resolved solutions
  - Includes Operating Experience and Lessons Learned that can be utilized in current project
- This historical information is key to ensuring the design is based on solid data and provides the ability to better model the systems.

Phenix IHX Leak and Changes

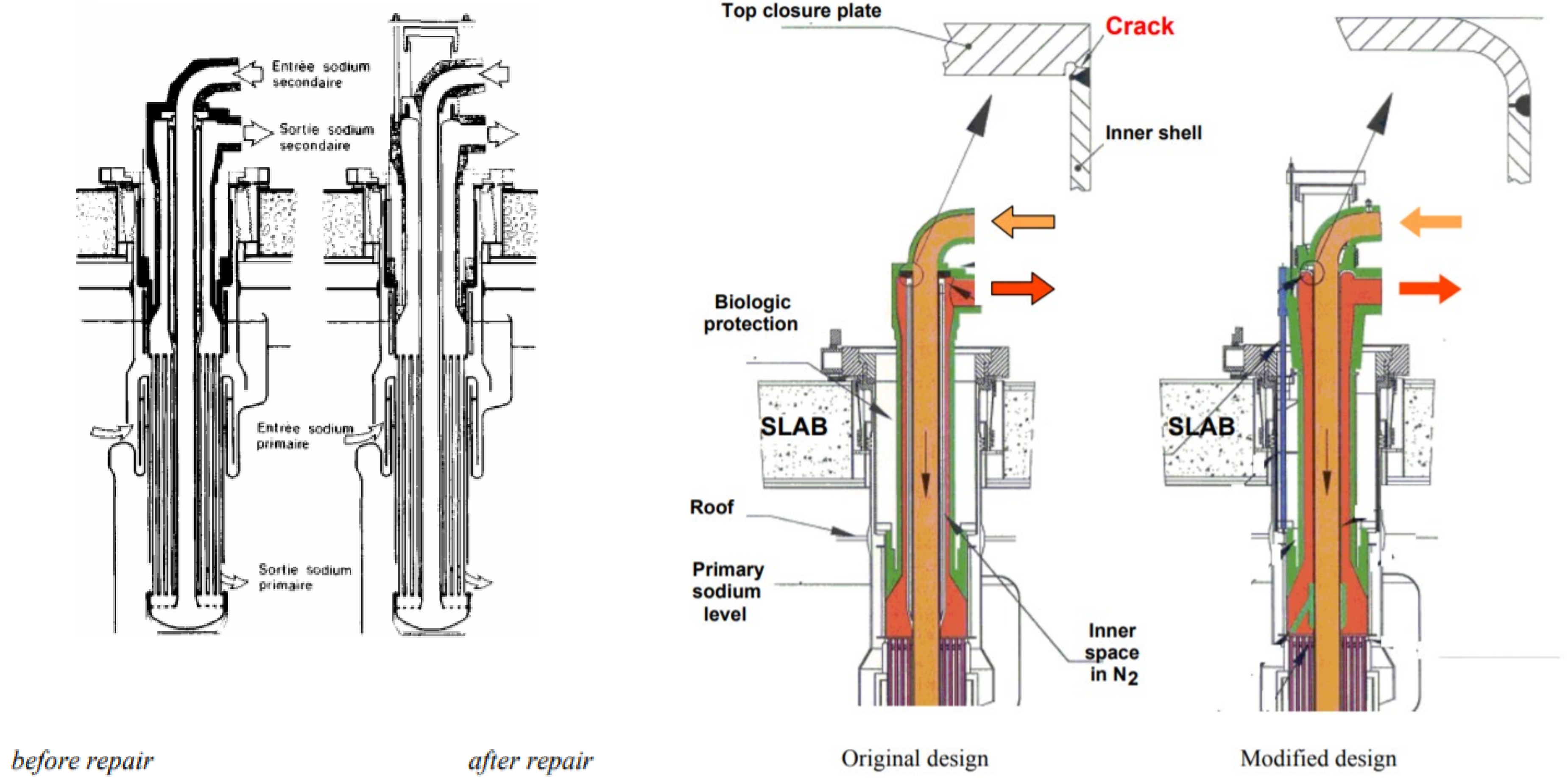


FIG. 3. PHENIX intermediate heat exchanger (IXH): places of failure, repair and modification. [(a)-before modification; (b)- after modification (flexible design elements and a flow-mixing device in the sodium header at the tube plate outlet)].

PFR Decay Heat Removal Air Dump Heat Exchanger Design Changes

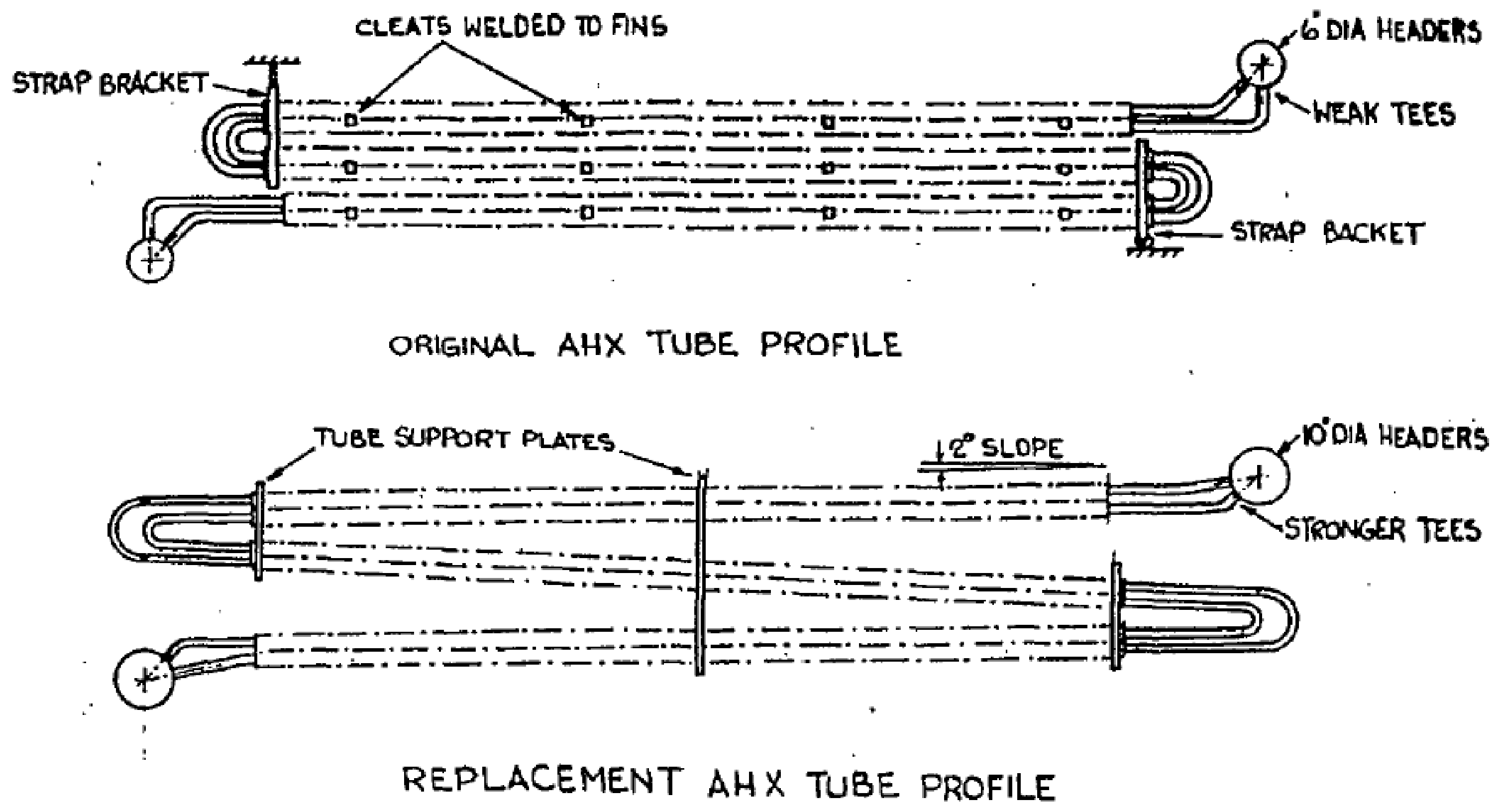
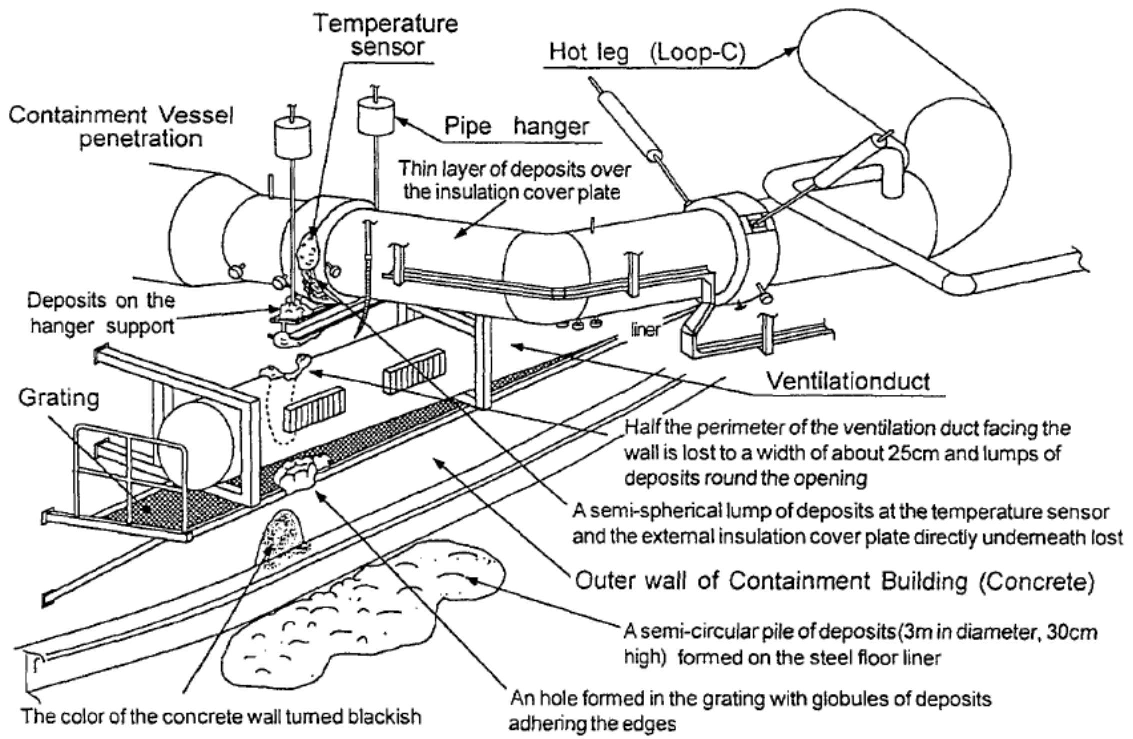
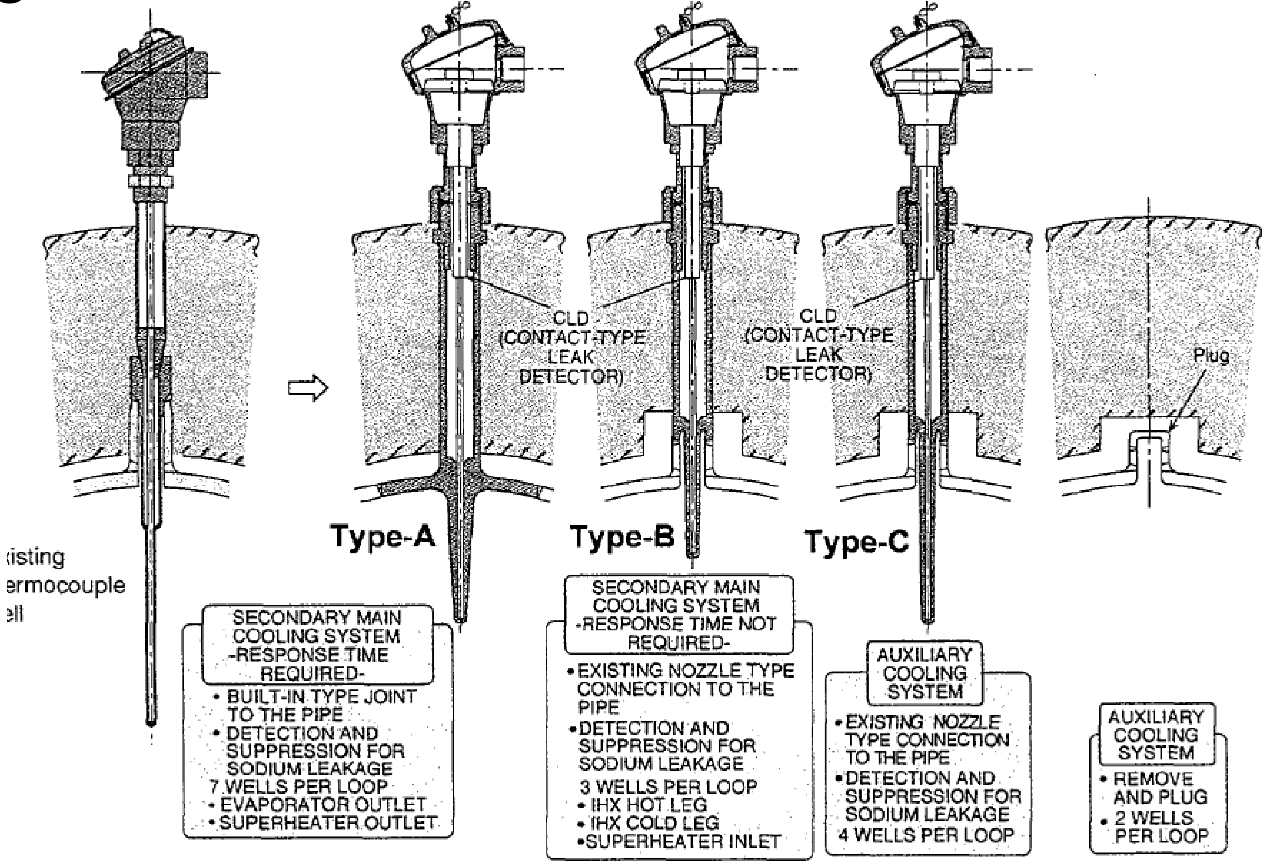


Figure 7.2. The Original and Replacement PFR THermal Syphon Air Heat Exchangers

Monju Leak

- Flow induced vibration of a thermocouple
- Redesign of thermocouple wells





TerraPower NATRIUM

Strategic Partnerships

- Work with PNNL and INL
  - PNNL has a contingent of SFR experts who consult on the Natrium Project
  - FFTF and EBR II historical records were retrieved from PNNL and INL to be referenced by the Natrium Project
- Work with JAEA
  - Historical Documentation sharing

TerraPower NATRIUM

Questions?

Upcoming Webinars

Date	Title	Presenter
26 July 2023	Off-Gas Xenon Detection and Management in Support of MSRs	Dr. Hunter Andrews, ORNL, Dr. Praveen Thallapally, PNNL, USA
31 August 2023	Corrosion and Cracking of SCWR Materials	Prof. Lefu Zhang, Shanghai Jiao Tong University, China
27 September 2023	EPRI Virtual Reality Training	Mr. Bob Eller, EPRI, USA