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License Amendment Request to Remove Instrumented Fuel Element Requirements from Technical Specifications

Robert Schickler
Steven Reese

Oregon State University Radiation Center

January 8th, 2020

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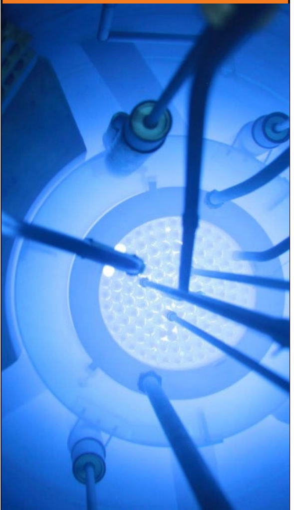

Background

The Oregon State TRIGA® Reactor (OSTR) is a 1 MW_{th} research reactor with pulsing capability that provides irradiation services for researchers throughout the world.

The OSTR has been operating since 1967. It was originally licensed for 250 kW and was shortly thereafter upgraded to 1 MW.

The OSTR converted from HEU fuel to 30/20 LEU fuel in Fall 2008.

At that time, Oregon State received 2 instrumented fuel elements (IFEs) and installed one in-core and kept the spare in dry storage.

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Core Configuration (May 2018)

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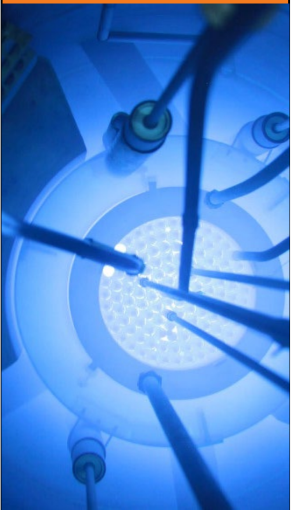
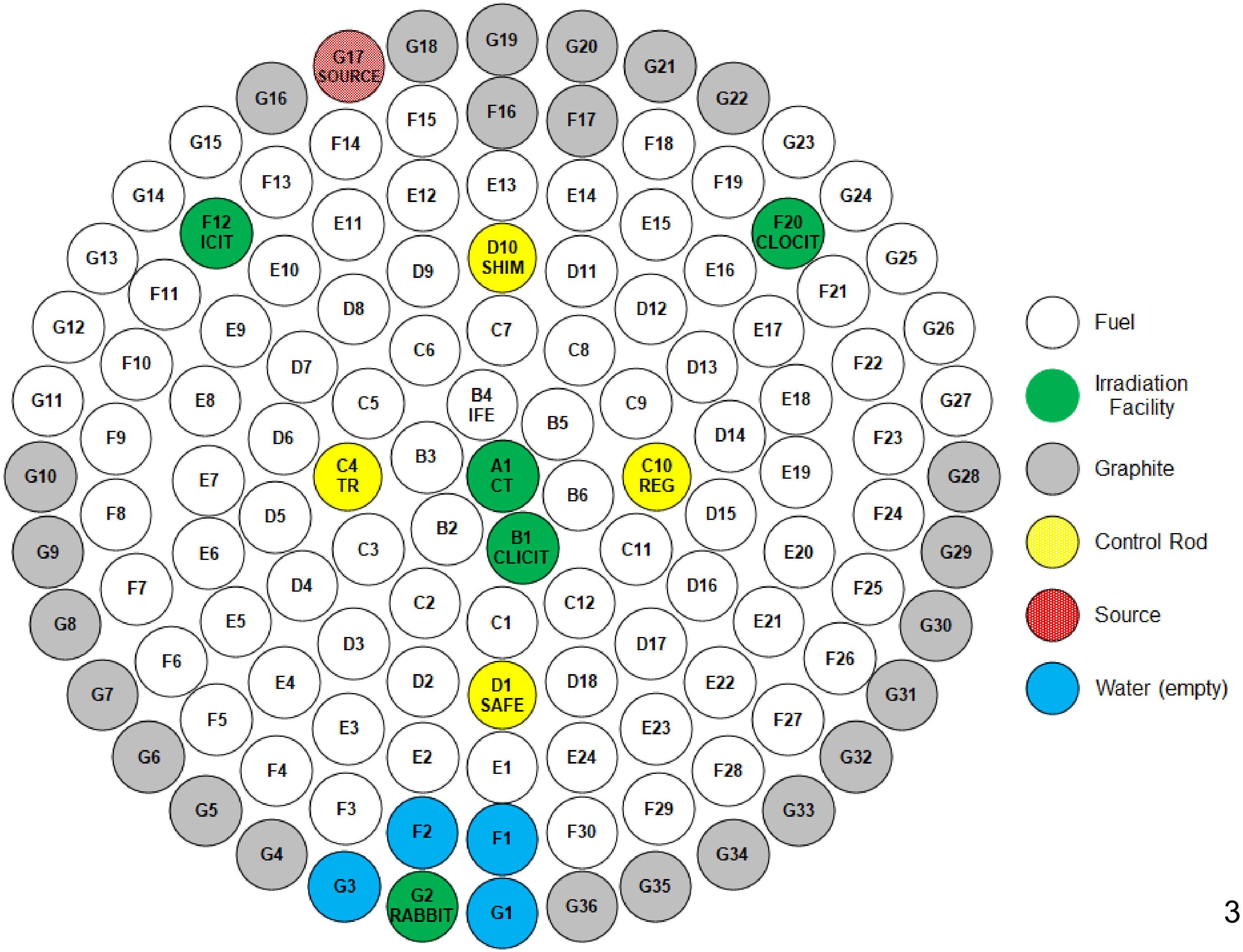


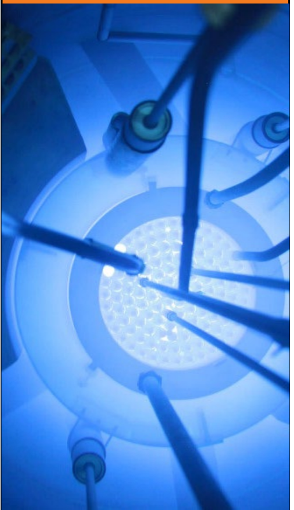


Timeline

May 2018: Performed a $2.20 pulse (administrative limit of $2.25) for Nuclear Engineering Reactor Lab course and noticed a 45°C jump on IFE temperature the next day, from ~340°C to ~385°C. No other indication of problems.

July 2018: Temperature continued to rise to ~410°C. Fuel inspection was performed on IFE and surrounding elements. All found to be acceptable with no visible defects or swell.

October 2018: Temperature continued to rise to ~450°C (LSSS of 510°C). Attempted to install spare IFE in core only to find that two of three thermocouples were failed open. Spare IFE was removed and original IFE was re-installed.



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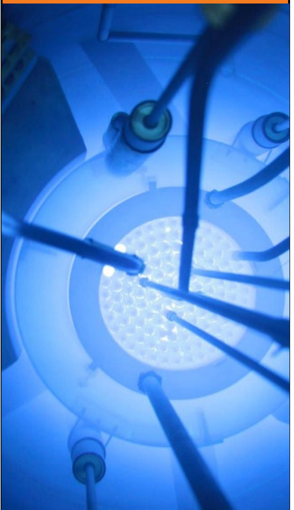
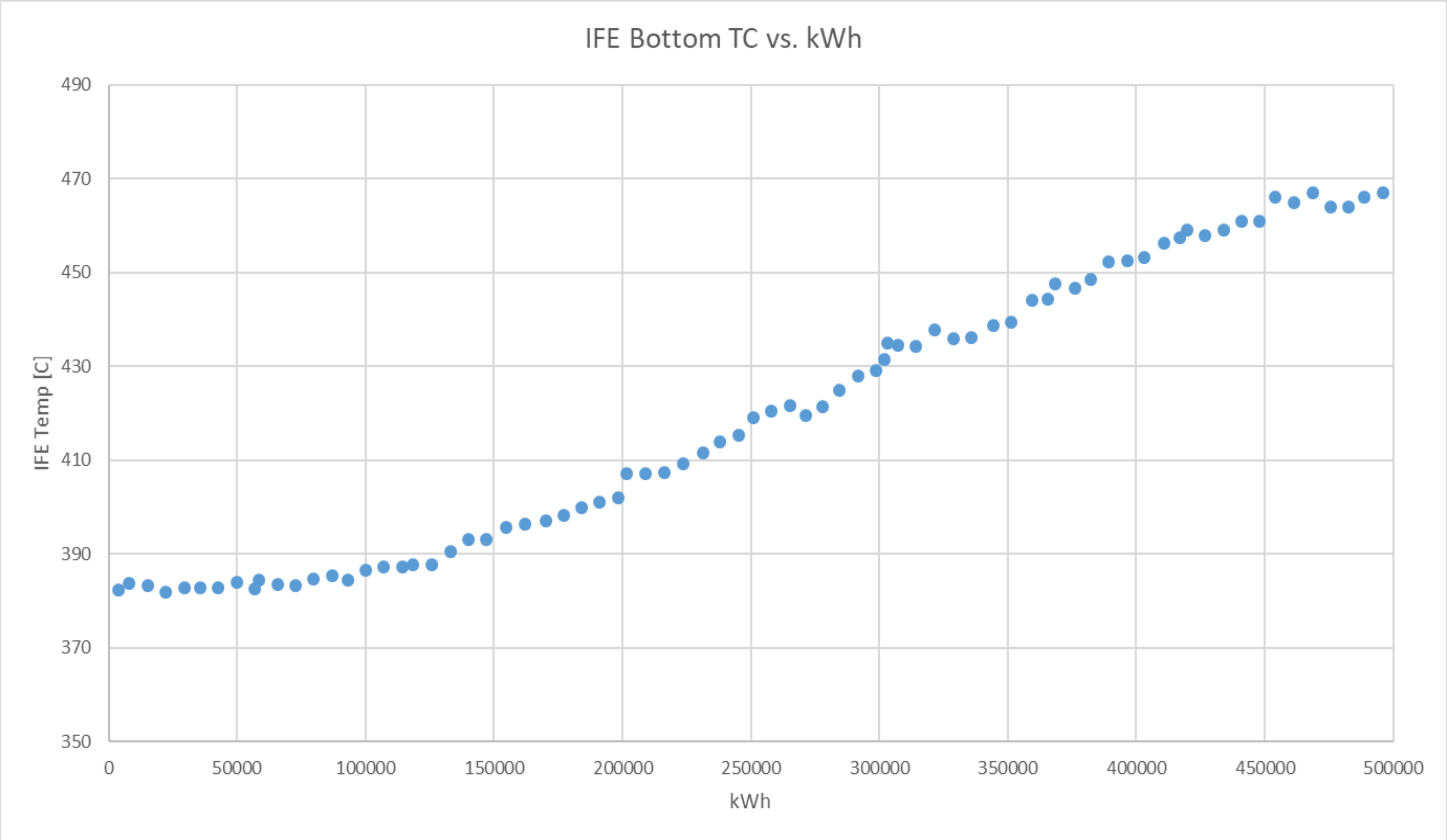


Timeline

November 2018: Submitted LAR to NRC to allow operation without IFE as long as pulsing is precluded. New LSSS to be based on power level. At this point, fuel temperature reached 470ºC (LSSS of 510ºC).



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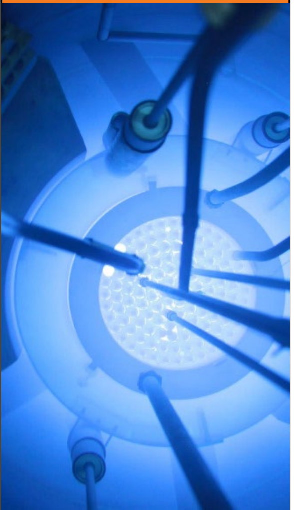


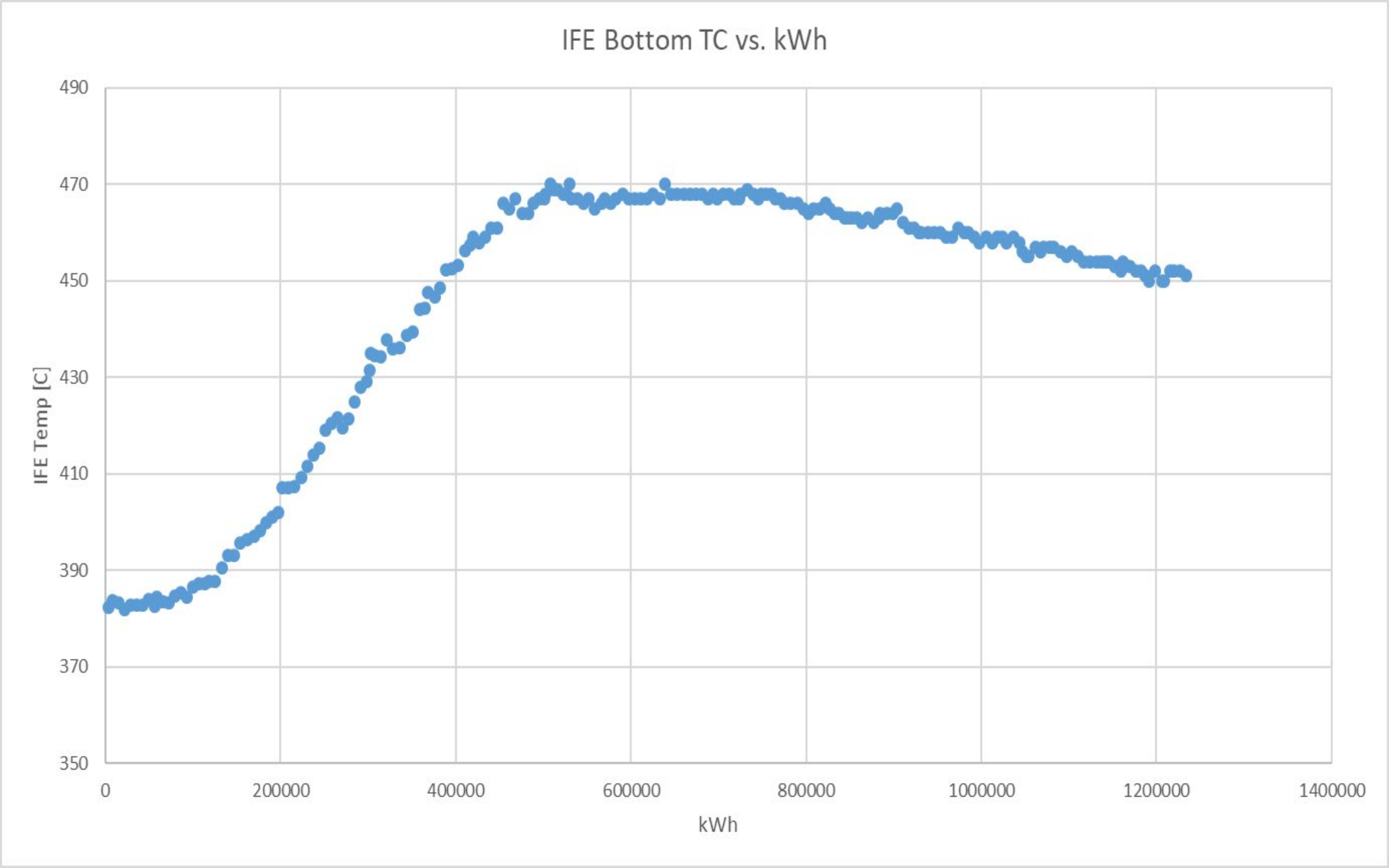
Timeline

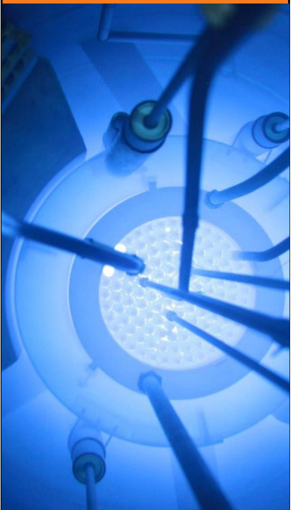
December 2018: Received spare IFE from Penn State. Tested thermocouples and they were all operable. This IFE was stored in anticipation of possible need for immediate installation.

Additional analysis would be needed before insertion due to differences in erbium content (Penn State IFE has 0.9% erbium, OSU Tech Specs require nominal 1.1% erbium content).

Timeline

April 2019: After peaking at 470°C, temperature gradually decreased to 450°C, reducing immediacy of IFE replacement. Still working with NRC on LAR.

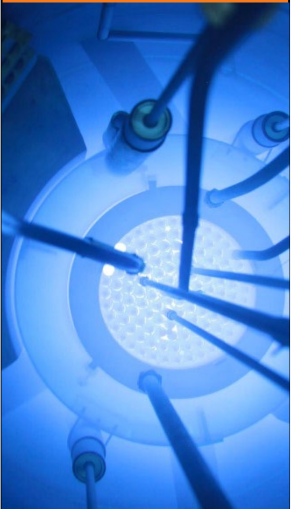




Timeline

June 2019: LAR requested approved! We would like to thank Mike Balazik for being incredibly helpful in getting this completed in a timely fashion.

LSSS now based on exceeding 1.1 MW on power channels with pulse mode precluded.



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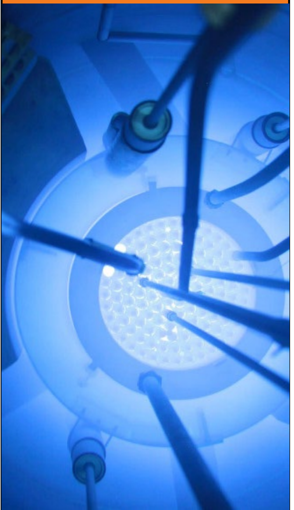
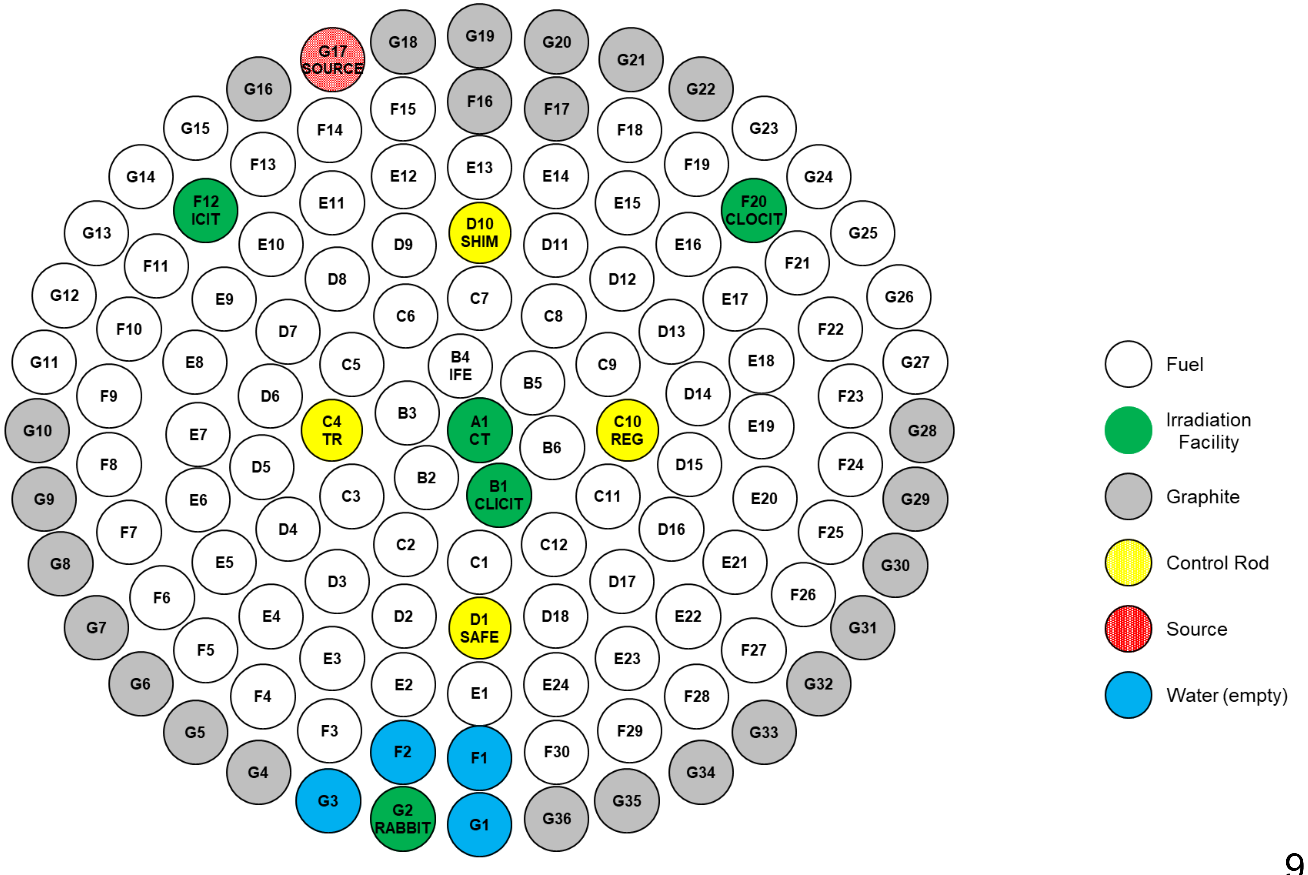


Before Core Reconfiguration

IFE removed from service on 7/29/19 and fuel temperature meter disconnected. Core reconfigured for operation without IFE.



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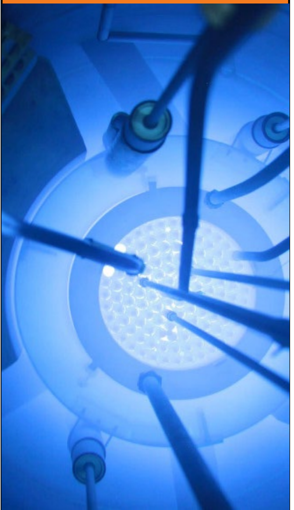
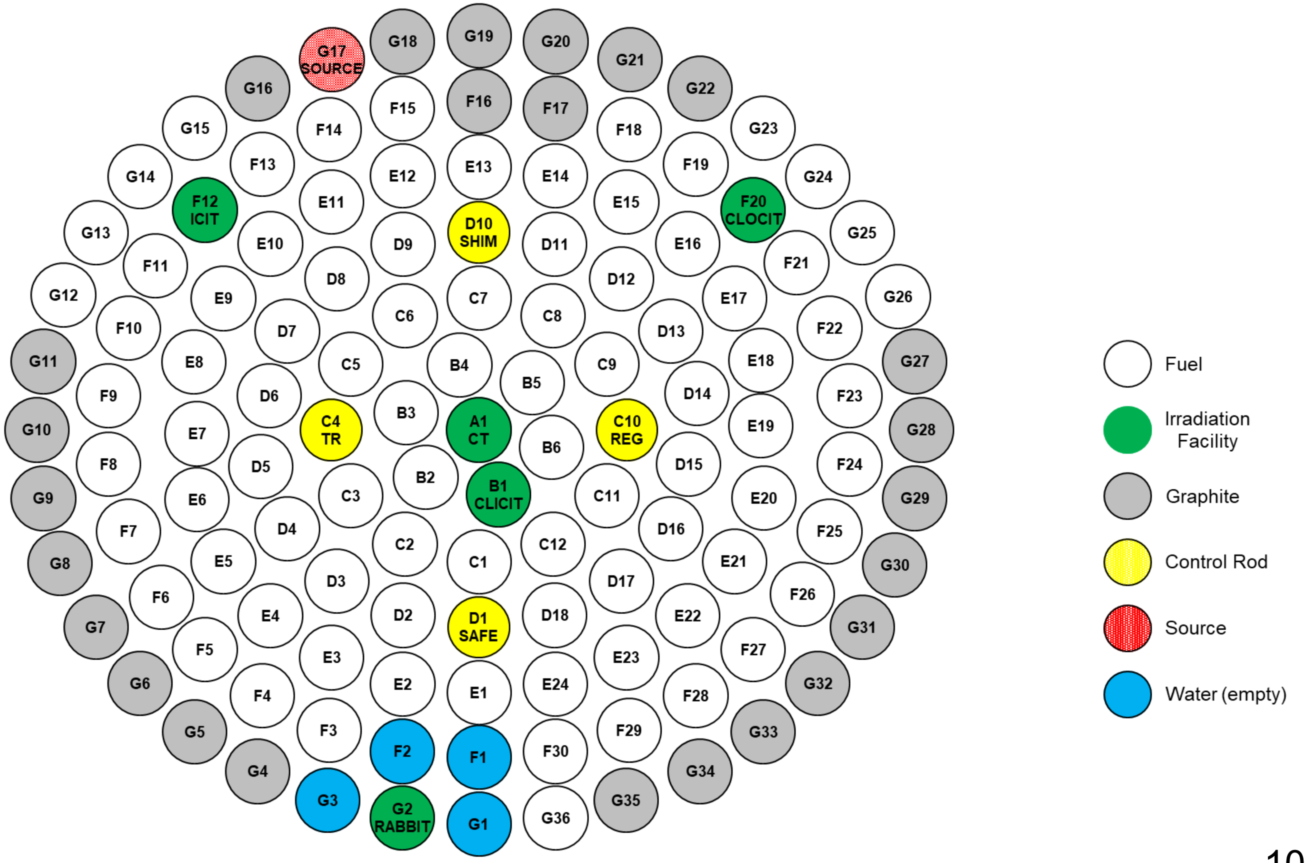


After Core Reconfiguration

IFE removed from service on 7/29/19 and fuel temperature meter disconnected. Core reconfigured for operation without IFE.

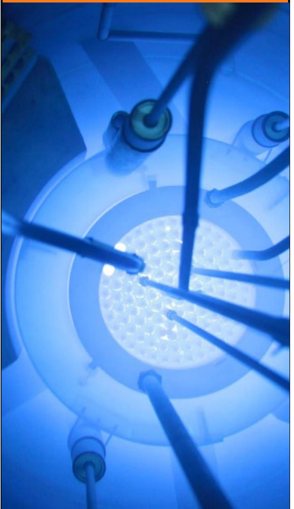


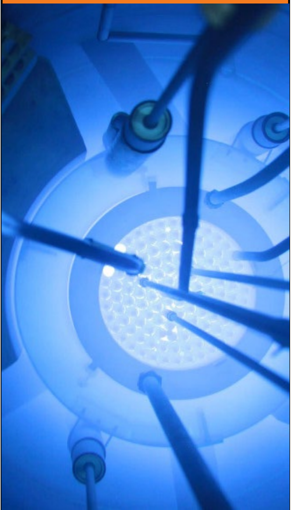
Current Status

Pursuing new LAR to remove IFE requirements from Tech Specs and allow for pulsing without an IFE.

Need to show:

1) Maximum temperatures in steady state are such that an IFE is not required to monitor temperature
2) Maximum pulse reactivity is such that temperature limits are not exceeded





Important Temperature Values

1150°C

This is the safety limit for OSTR fuel, based upon ultimate failure of the fuel cladding.

830°C

This is the temperature recommended by Argonne National Laboratory, referencing eutectic formation. This has yet to be substantiated.

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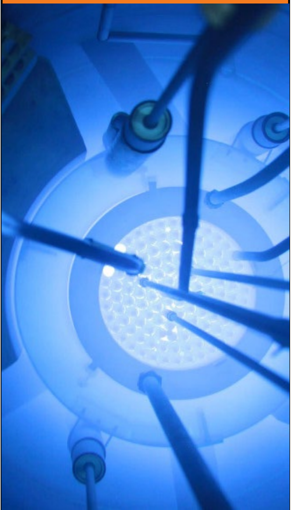

Maximum Power-Per Element

In order to determine the maximum fuel temperature in steady state, the Maximum Power-Per-Element must be determined.

Neutronic analysis (using MCNP) was performed in order to calculate the Maximum Power-Per-Element in various core configurations at 1.1 MW (maximum licensed power).

Using F4 flux tallies with an FM multiplier card, MCNP can calculate the power produced in each fuel element.

Thus MCNP can be used to determine the Maximum Power-Per-Element, i.e. the Hot Channel.

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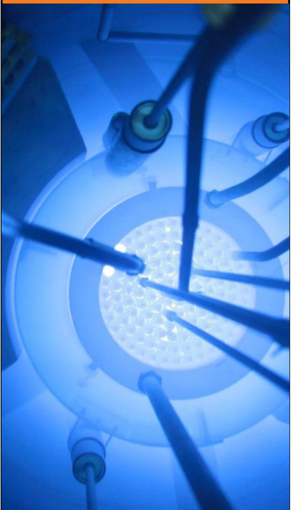

Maximum Power-Per Element

During fuel conversion, hot channels were determined for three different core configurations (ICIT, CLICIT and NORMAL) at three different stages of core life (BOL, MOL, EOL).

These configurations were dependent on the contents of the B1 grid location. Either an irradiation facility (ICIT/CLICIT) or fuel element (NORMAL) would be located in B1.

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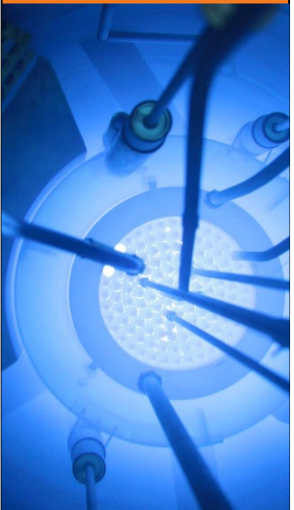

Maximum Power-Per Element

The following Maximum Power-Per-Element values were thus determined for each core configuration.

Core Configuration	Hot Channel	Hot Channel Thermal Power [kW]
BOL ICIT	B6	18.47
MOL ICIT	B6	18.52
EOL ICIT	B6	17.61
BOL CLICIT	B3	17.03
MOL CLICIT	B3	17.03
EOL CLICIT	C7	16.35
BOL NORMAL	B3	17.77
MOL NORMAL	B3	17.80
EOL NORMAL	B3	17.02

It is important to note that OSTR has eliminated the ICIT and NORMAL configurations from regular operations.

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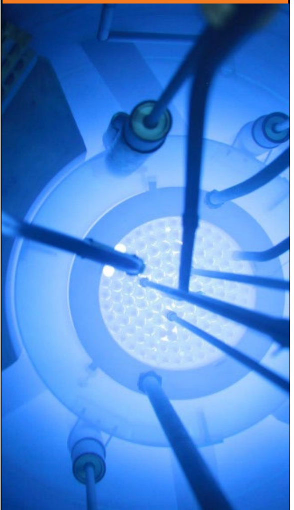

Thermal Hydraulic Analysis

Once the Maximum Power-Per-Element and Hot Channels were determined, the Hot Channel Peaking Factor (Maximum Fuel Rod Power/Core Average Fuel Rod Power) was calculated. Then another MCNP calculation was performed using an FMESH card to obtain a 20 radial by 20 axial mesh tally, which was used to determine:

- Hot Channel Axial Peaking Factor
  - Maximum Axial Power in hot channel/Average Axial Power
- Hot Channel Radial Peaking Factor
  - Maximum Radial Power in hot channel/Average Radial Power
- Effective Peaking Factor (product of three factors)

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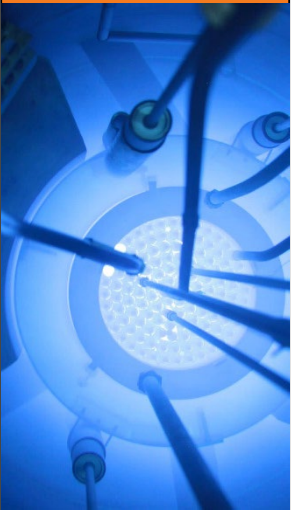


Peaking Factors

The peaking factors for all nine configurations are:

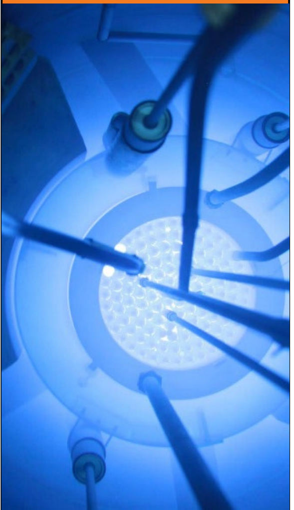
Core Configuration	Hot Channel Peak Factor	Axial Peak Factor	Radial Peak Factor	Effective Peak Factor
BOL ICIT	1.477	1.221	1.562	2.817
MOL ICIT	1.482	1.225	1.434	2.603
EOL ICIT	1.409	1.181	1.304	2.170
BOL CLICIT	1.362	1.221	1.536	2.554
MOL CLICIT	1.363	1.225	1.406	2.348
EOL CLICIT	1.308	1.212	1.275	2.021
BOL NORMAL	1.422	1.219	1.538	2.666
MOL NORMAL	1.424	1.222	1.409	2.452
EOL NORMAL	1.362	1.178	1.267	2.033

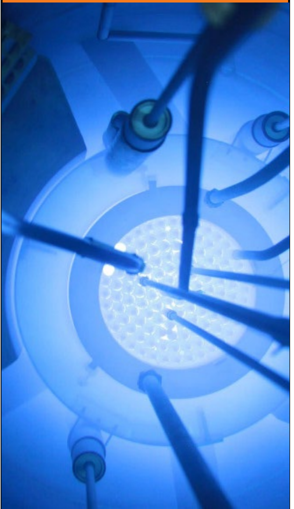
The ICIT is the most conservative core configuration at all points during core lifetime.

Again, it is important to note that OSTR has eliminated the ICIT and NORMAL configurations from regular operations.

Thermal Hydraulic Analysis

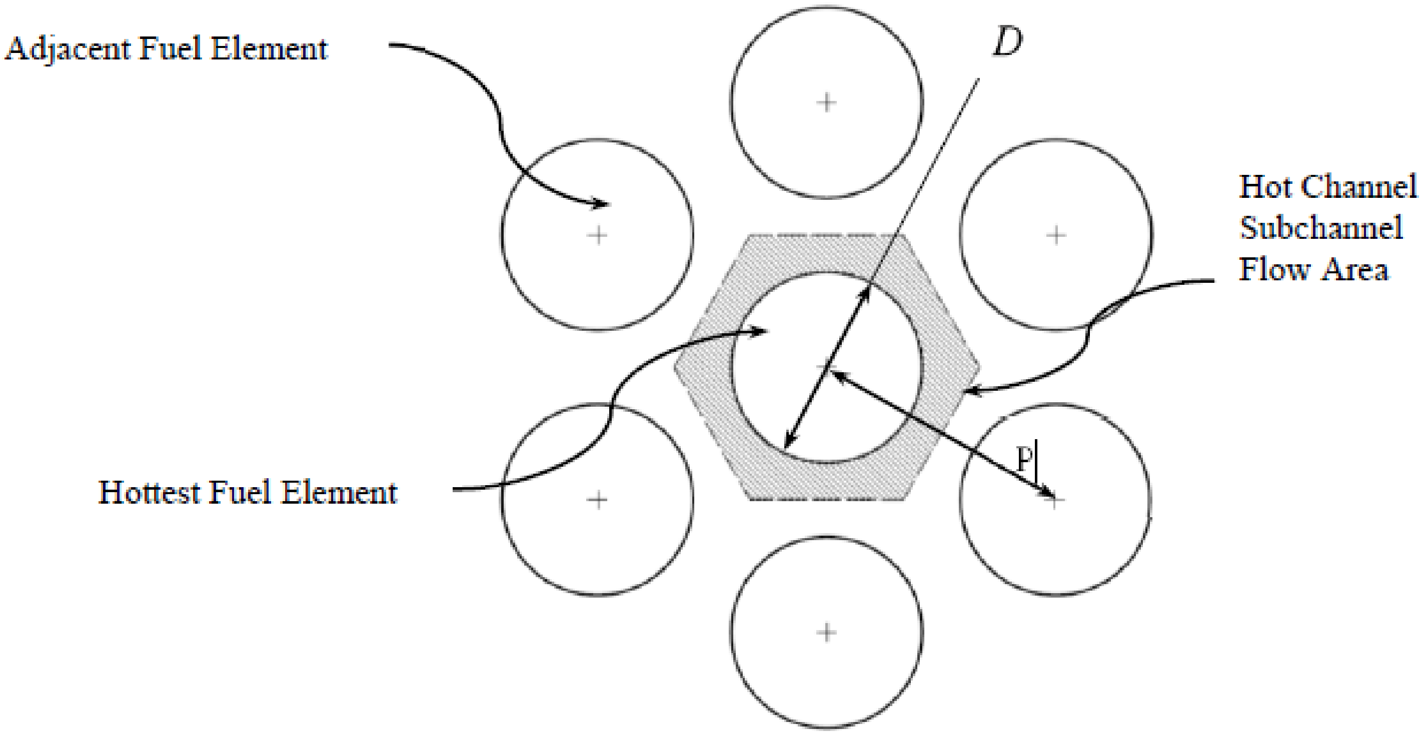
Since the ICIT is the most limiting core configuration, thermal hydraulic analyses were performed in the ICIT core using RELAP5-3D to determine the Departure from Nucleate Boiling Ratio (DNBR).





Thermal Hydraulic Analysis

The most limiting pitch was found to be in the B-Ring, thus this was the subchannel flow area used in the analysis.



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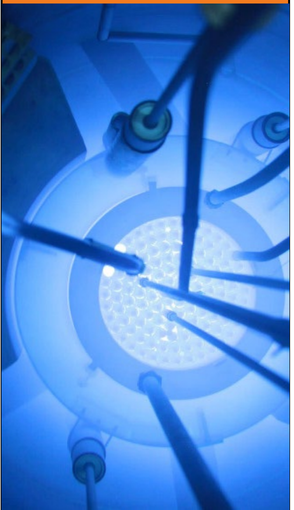
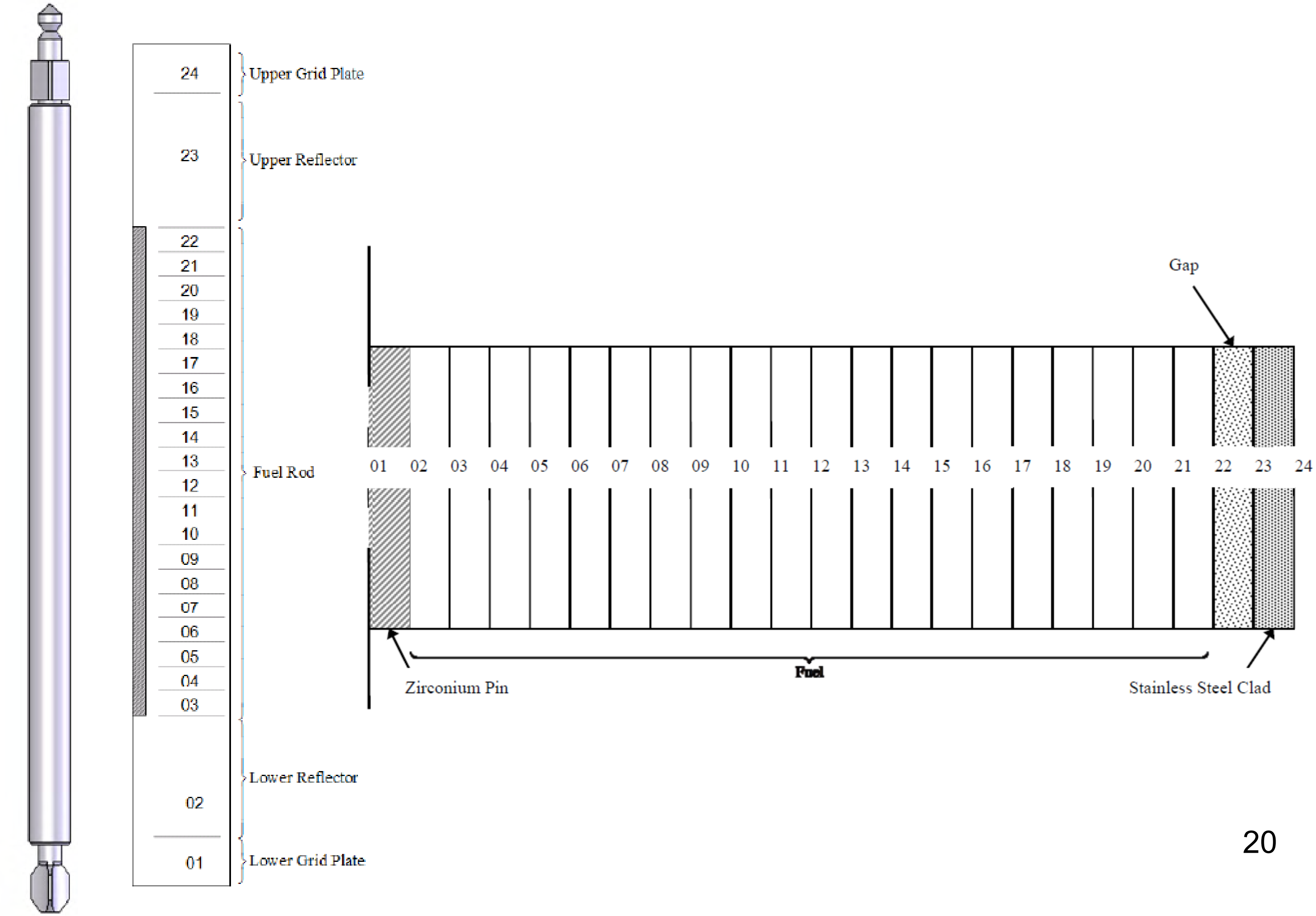


Thermal Hydraulic Analysis

The RELAP5-3D model used 25 axial and radial nodes.



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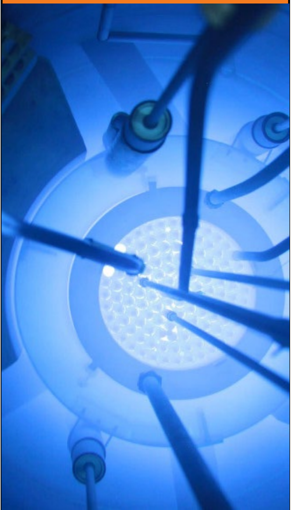

Thermal Hydraulic Analysis

Two correlations were used in the thermal hydraulic analysis:

1) 2006 AECL Groeneveld Lookup Tables
   - Most current method for calculating CHF values
   - Likely the most applicable correlation
2) Bernath Correlation
   - Traditionally used as a supplement in research reactor SARs
   - Produces most limiting CHF values (most conservative)
   - Originally created in 1961 for PWR assemblies.

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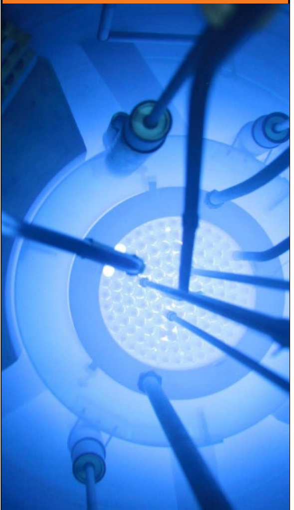


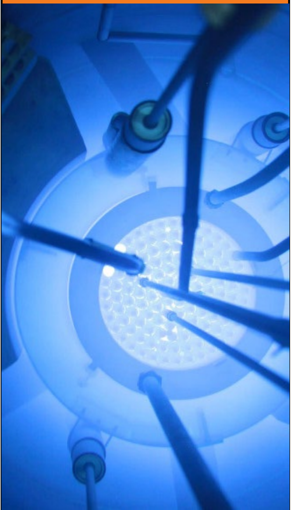
Thermal Hydraulic Analysis

These are the results of steady-state TH analysis for each ICIT core analyzed:

Parameter	BOL	MOL	EOL
Flow rate for hottest rod [kg/s]	0.0843	0.0844	0.0812
Maximum flow velocity [m/s]	0.2339	0.2352	0.2245
Maximum wall heat flux [kW/m^2]	504.49	507.74	465.55
Maximum fuel centerline temperature [°C]	448.13	457.66	438.39
Maximum clad temperature [°C]	131.93	131.46	130.57
Exit clad temperature [°C]	126.36	125.98	125.87
Exit bulk coolant temperature [°C]	101.32	101.40	100.78
MDNBR [Groeneveld 2006]	4.796	4.754	5.048
MDNBR [Bernath]	2.083	2.060	2.202

Each configuration has a DNBR greater than 2.

The Bernath correlation is overly conservative, as the Groeneveld DNBR values are over twice as large.

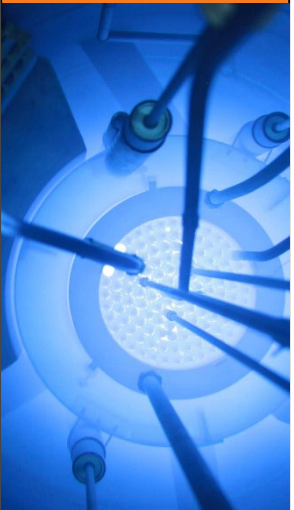


Thermal Hydraulic Analysis

Calculated temperatures (in °C) for various hot channel powers (in kW) in the LEU BOL ICIT core:

$P_{\text{hot-channel}}$	T_{max}	T_{cladding}	T_{coolant}
14	371	129	95
16	406	130	99
18.47	448	131	101
20	474	132	102
22	508	133	103

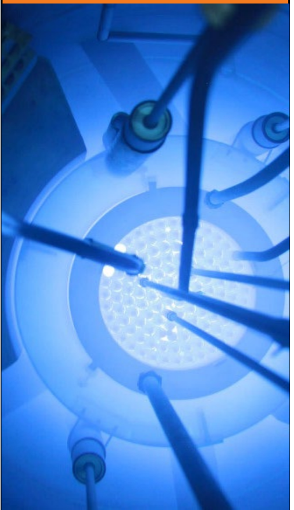
Note that the highlighted power is the hot channel at 1.1 MW.



Thermal Hydraulic Analysis

Calculated temperatures (in °C) for various hot channel powers (in kW) in the LEU MOL ICIT core:

$P_{hot\text{-}channel}$	T_{max}	$T_{cladding}$	$T_{coolant}$
14	378	129	95
16	413	130	99
18.52	458	131	101
20	483	132	102
22	518	133	103



Thermal Hydraulic Analysis

Calculated temperatures (in °C) for various hot channel powers (in kW) in the LEU EOL ICIT core:

$P_{hot\text{-}channel}$	T_{max}	$T_{cladding}$	$T_{coolant}$
14	375	129	95
16	410	130	99
18.02	438	131	101
20	480	132	102
22	514	133	103

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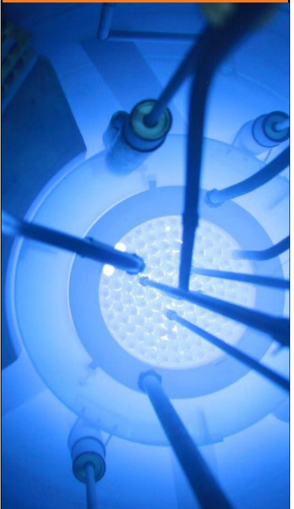
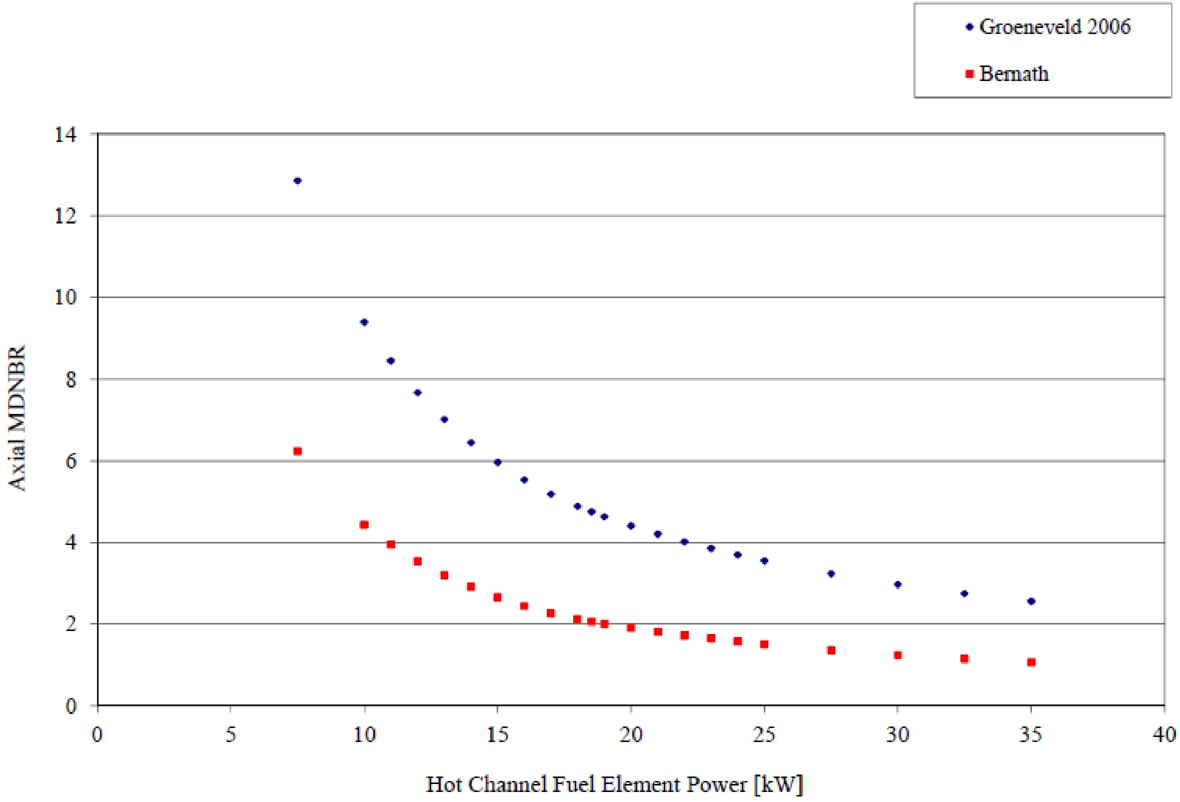
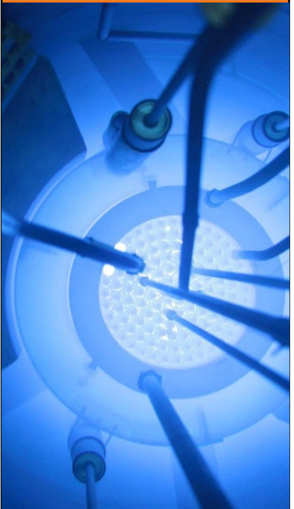


Thermal Hydraulic Analysis

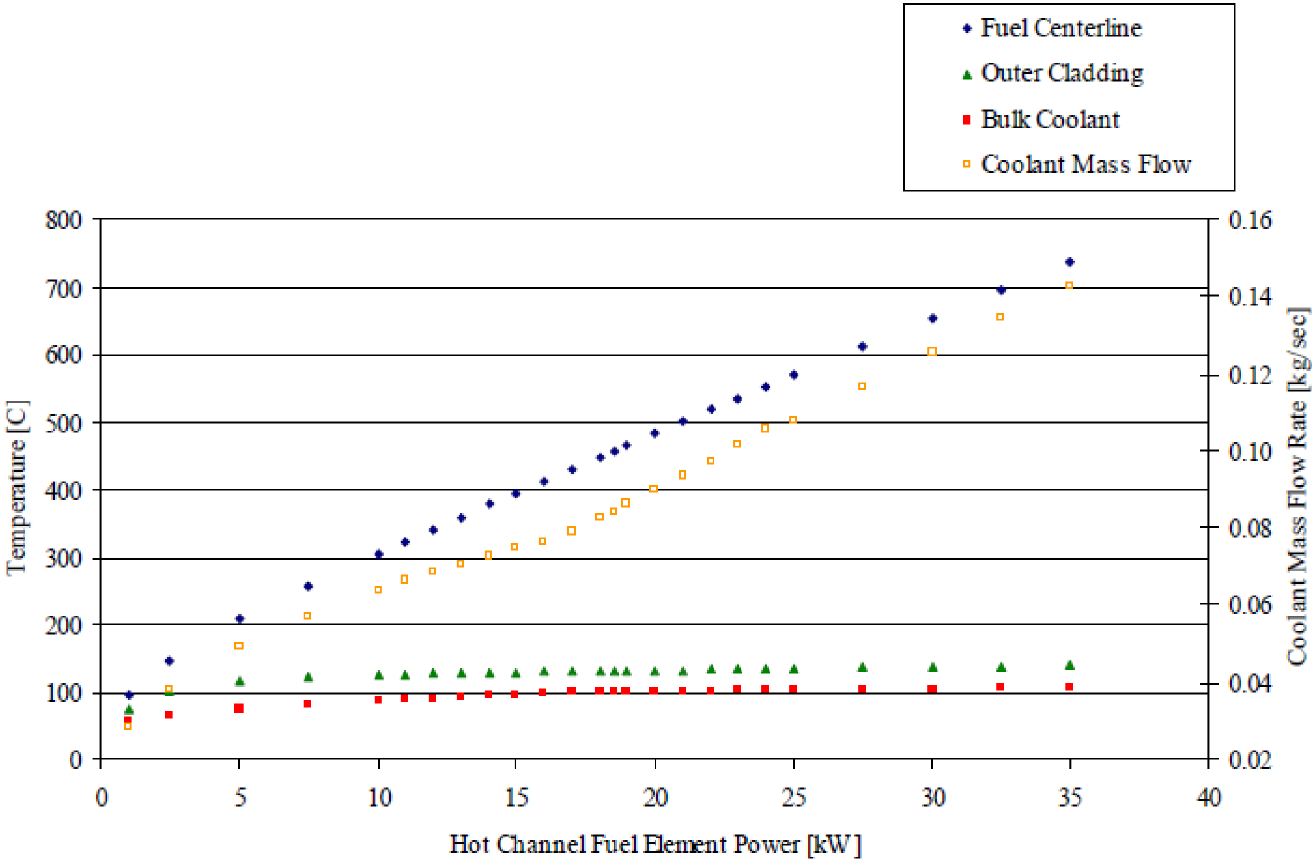
Using the limiting LEU MOL ICIT core, the DNBR reaches a value of 2 at approximately 19.85 kW according to Bernath. This is significantly larger than the 18.52 kW hot channel power-per-element. Note that Groeneveld predicts DNBR reaching 2 at a power greater than 35 kW.





Thermal Hydraulic Analysis

These are the calculated hot channel properties for the LEU MOL ICIT core. Even at 35 kW of power, the fuel centerline does not exceed 800°C.



Steady-State Thermal Hydraulic Analysis Results

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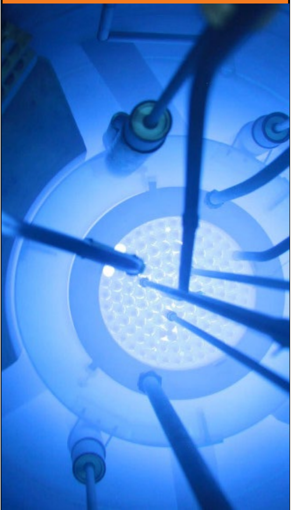
As long as Maximum Power-Per-Element does not exceed 19.85 kW, the DNBR should be greater than 2. The maximum expected power-per-element in the most limiting core configuration was 18.52 kW (at 1.1 MW total core power).

Thus the OSTR cannot depart from nucleate boiling during normal 1 MW operation.



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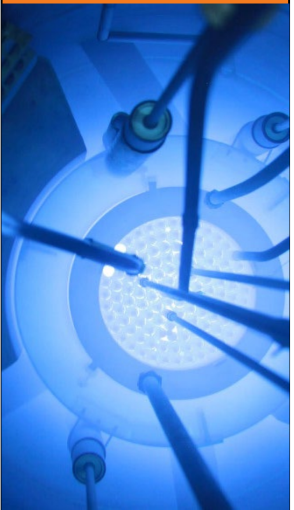

Steady-State Thermal Hydraulic Analysis Results

Even with a Maximum Power-Per-Element of 35 kW (almost twice the calculated maximum power-per-element of 18.52 kW), fuel temperature is not expected to exceed 800°C. This is still significantly lower than the GA-recommended limit of 830°C.

Thus the OSTR cannot expect to experience fuel damage during normal 1 MW operation, and an instrumented fuel element is not needed for temperature information.

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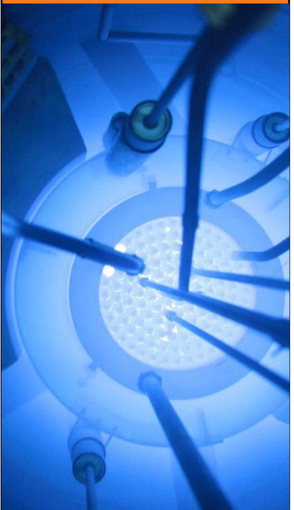
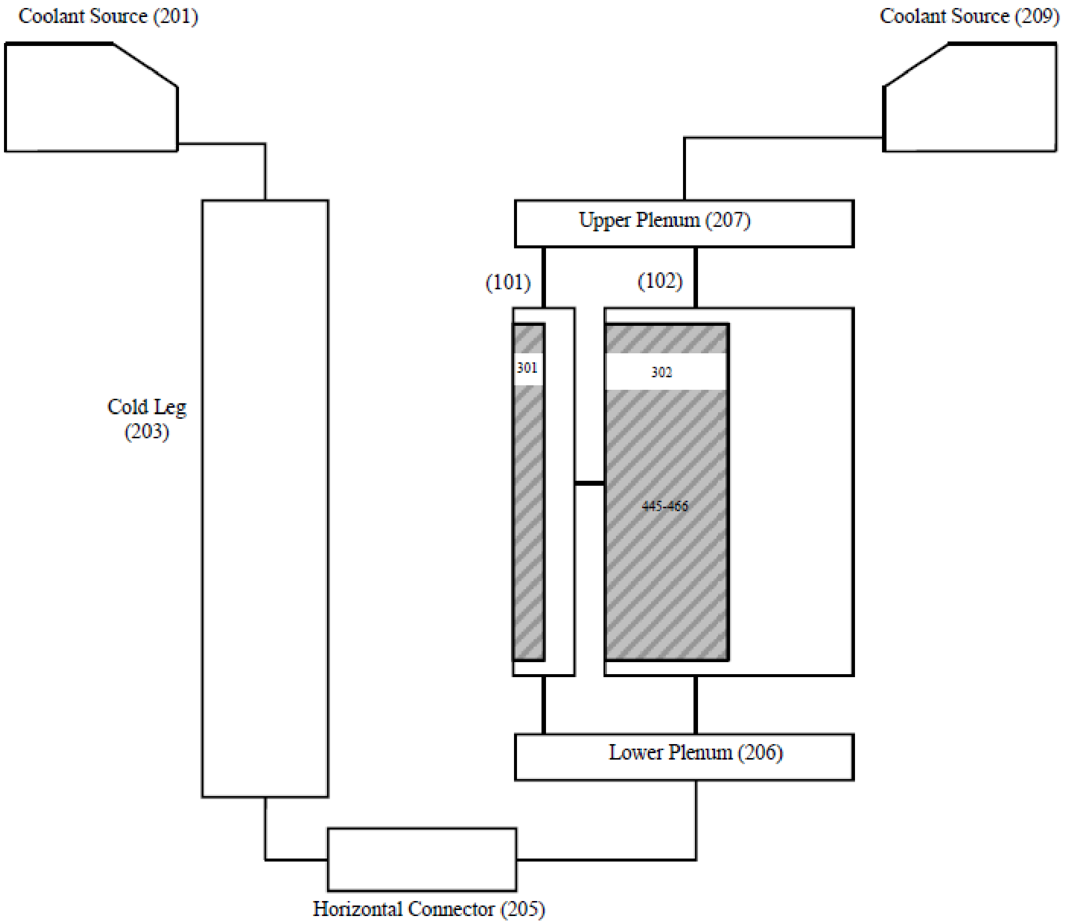


Pulse Analysis

RELAP’s point-reactor kinetics function can also be used to determine the maximum peak fuel temperature during a pulse in order to determine maximum reactivity insertion.



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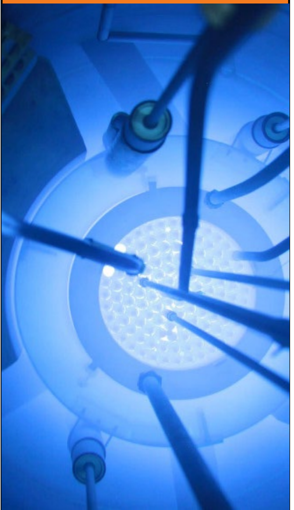
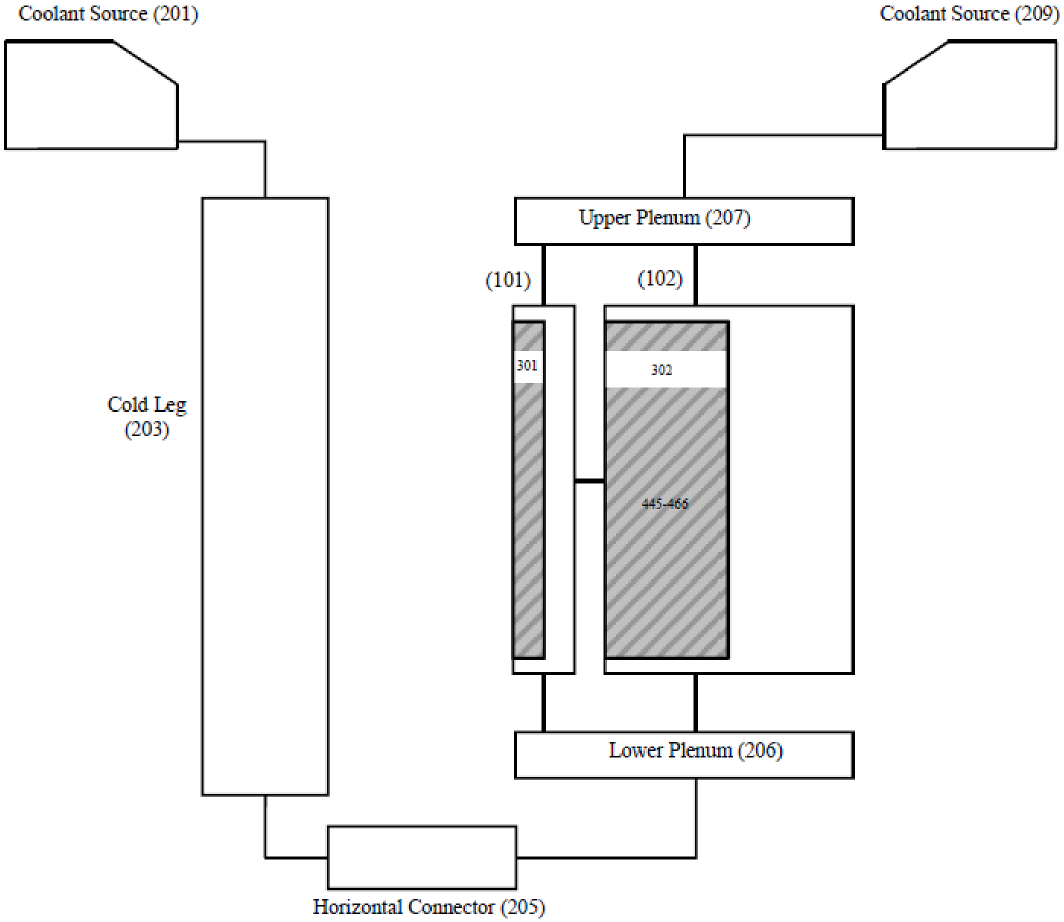


Pulse Analysis

Volume 102 is the core average volume and its heat generation information is passed onto the hot channel (Volume 101).



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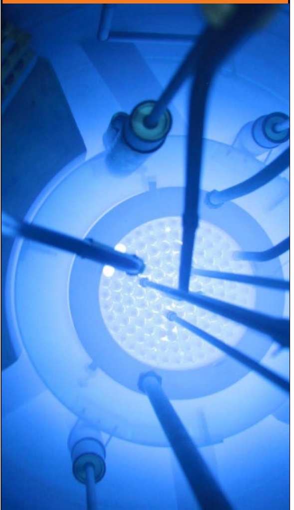

Pulse Analysis

The RELAP version used for all calculations had a known PRK error. OSU developed a PRK model from basic principles to compare results to RELAP as well as a GA benchmark problem. The comparison of the methods was as follows (and show good agreement):

Parameter	GA Paper	RELAP	OSU PRKM
Maximum Power [MW]	20000	20606	21058
Time of Maximum Power [sec]	~0.0207	0.02108	0.02032
Pulse FWHM	~0.00351	0.00464	0.00454
Peak Adiabatic Fuel Temp [°C]	1000	1083.1	880.665
Average Adiabatic Core Temp [°C]	500	492.032	473.249
Core Energy Release After 0.1 sec [MJ]	106	109.47	108.24

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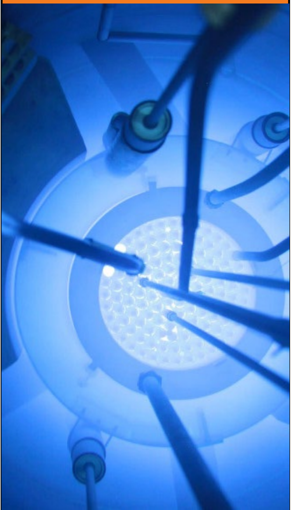
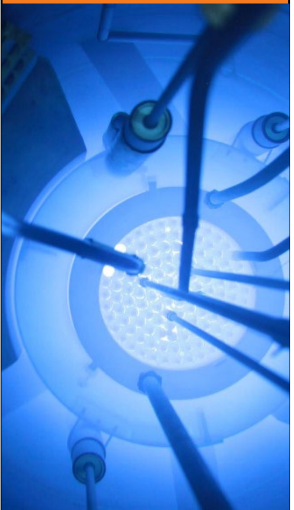


Pulse Analysis

The RELAP PRKM was used to determine pulse characteristics in the limiting LEU MOL ICIT core. The results are as follows:

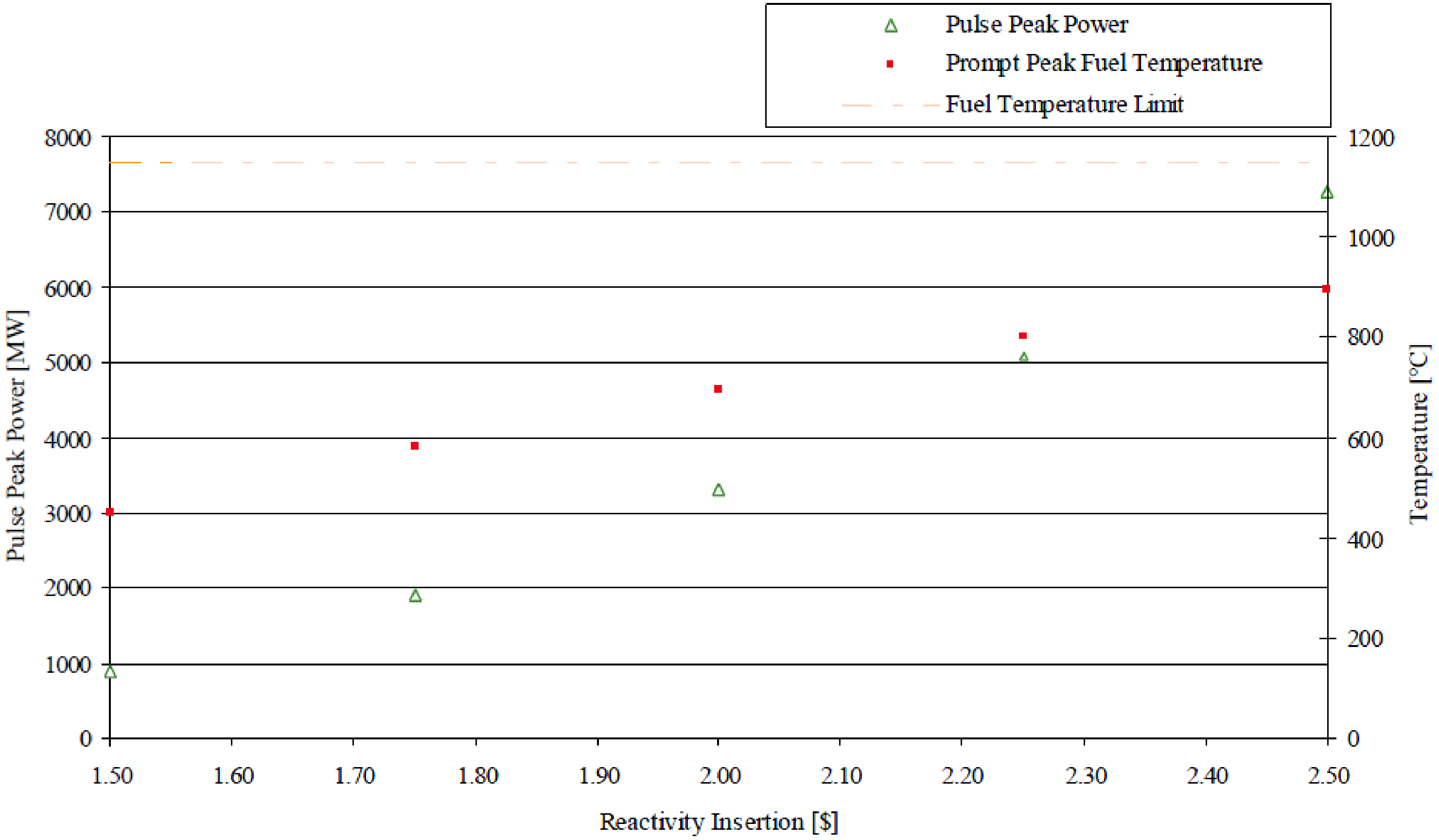
Reactivity Insertion [$]	1.50	1.75	2.00	2.25	2.50
Peak Total Core Power [MW]	875	1910	3316	5087	7270
Prompt Peak Fuel Temperature [°C]	448	582	697	800	894
Maximum Thermocouple Temperature [°C]	375	480	574	657	724

Note that the IFE thermocouple temperature is significantly lower than the actual prompt peak fuel temperature due to the location within the fuel meat.



Pulse Analysis

Prompt peak fuel temperature is linear to reactivity. Interpolation shows that 830°C is exceeded at $2.33. Thus the reactivity limit for OSTR was set at $2.30.



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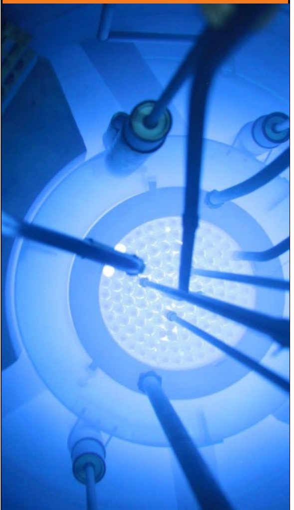


USGS Technical Specifications

USGS is a very similar reactor to OSTR. It also has pulsing capability and does not have any technical specification requirements for instrumented fuel elements.

1) LSSS is based on power level.
2) No IFE-based LCOs nor IFE scram requirements.
3) Pulse limits based on preventing fuel from exceeding 830°C, which was derived from analysis, no IFE required.

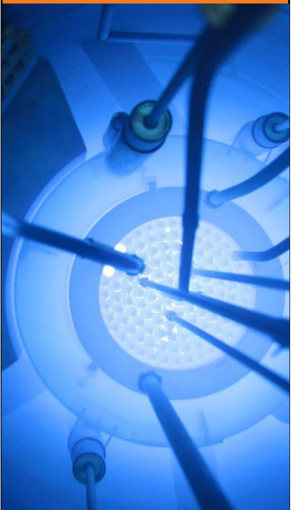
Summary

IFEs can be faulty and thereby cause a reactor to remain shutdown. OSTR nearly experienced a shutdown due to its IFE failures.

As long as the OSTR is operated within the Technical Specification limit of 1.1 MW, analysis shows that temperature limits cannot be exceeded, thereby making the IFE redundant and unnecessary.

IFEs offer no safety function as they cannot cause a scram faster than a reactivity excursion.

While IFEs are an interesting tool for information, they are ultimately limiting on operation and an unnecessary expense.



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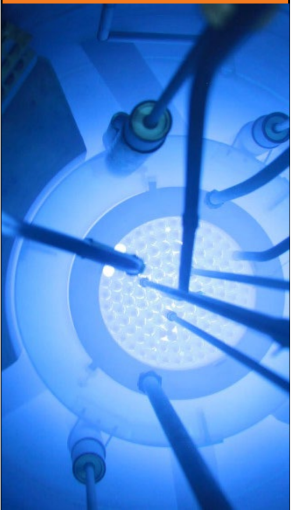
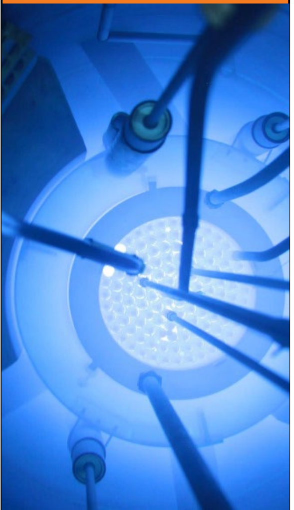
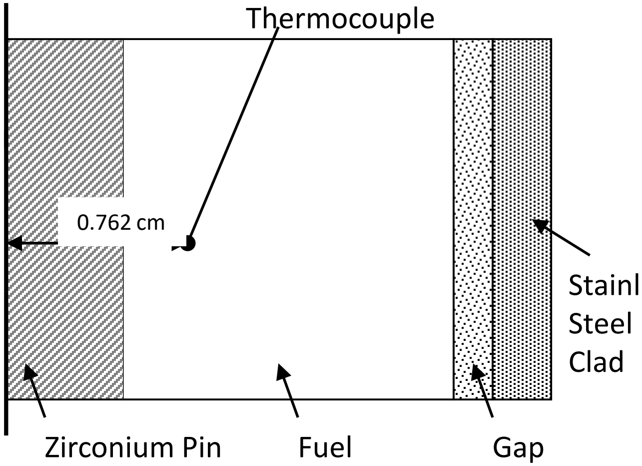


Questions?

OSU Radiation Center



IFE inspection performed on 7/3/18 showed no apparent damage or swelling on IFE or on any surrounding elements



Cross Sectional View
Instrumentation Fuel Element