Safety Margin Evaluations for ATR In-Core Experiments Supporting U-10Mo LEU Fuel Development

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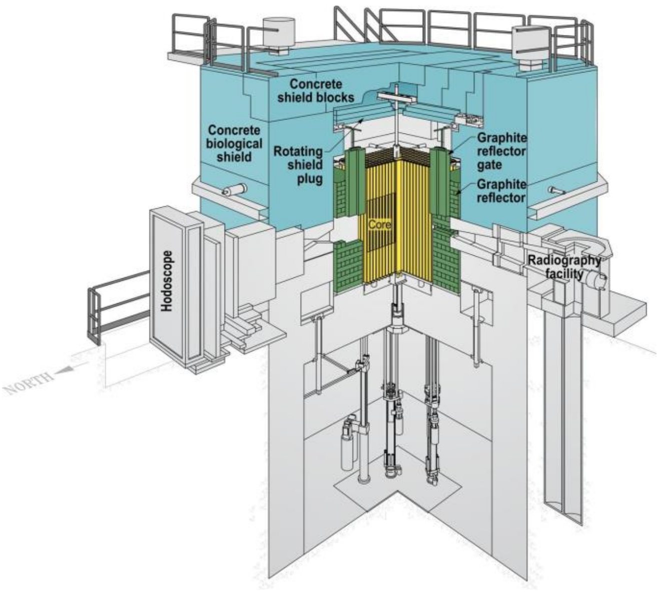
Nuclear Reactor Laboratory

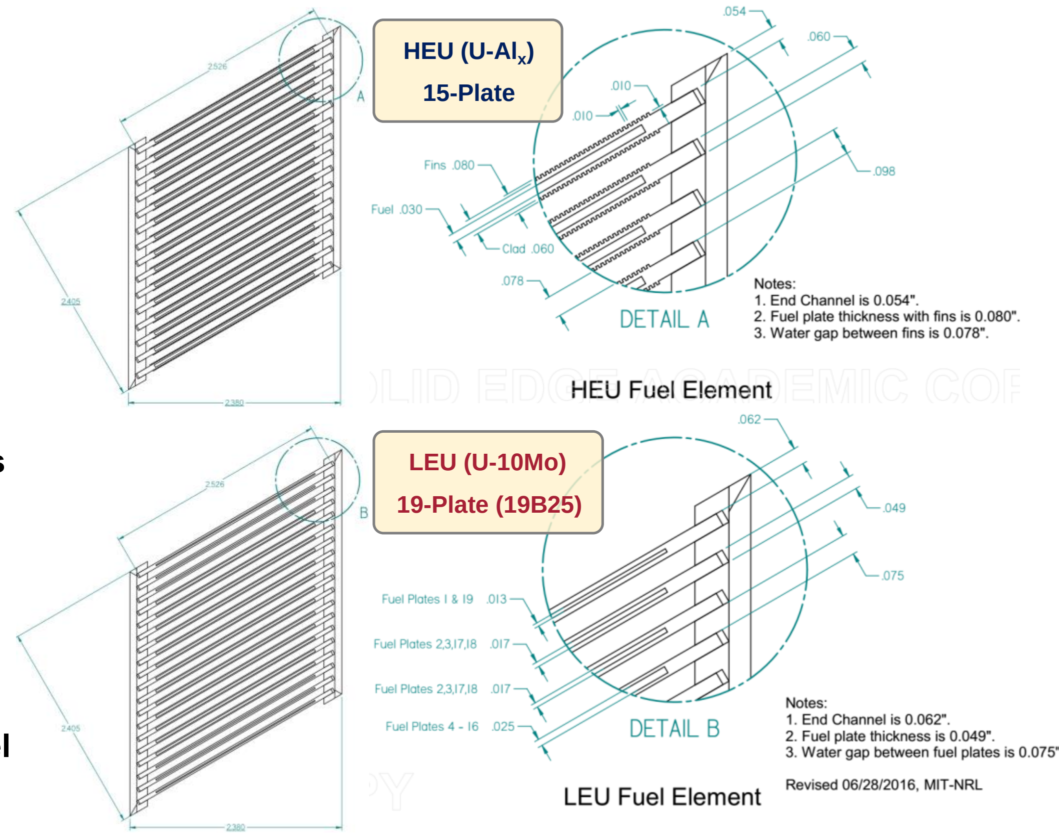
LEU Conversion

- Most U.S. civil-use research and test reactors have been converted from HEU to LEU.
- The remaining ones are five high performance ones (HPRRs), i.e., MITR, MURR, NBSR, ATR (ATRC), and HFIR. (Plus, the to-be-restarted transient test reactor TREAT)
- Reducing the enrichment to 19.75 wt. % U-235 for the HPRRs led to changing the fuel-bearing region of the plates from a dispersion cermet fuel form to an alloy of uranium and 10 wt. % molybdenum, referred to as the U-10Mo monolithic fuel foil.
- This high-density U-10Mo monolithic alloy fuel (15.5 gU/cm^3) is currently under development and qualification.



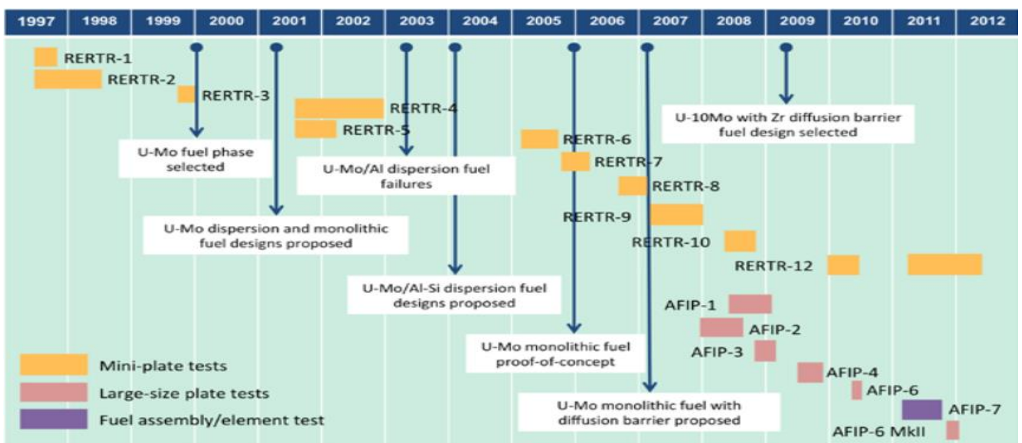
MITR CONVERSION PROGRESS

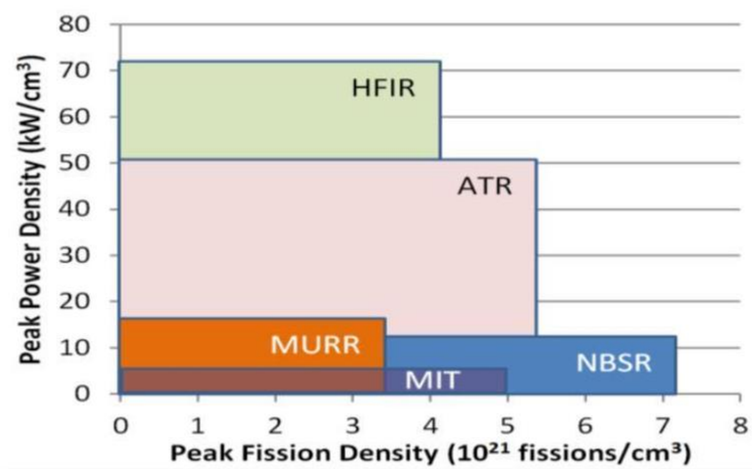
- ✓ **U-10Mo LEU Fuel Design (19B25)**
- ✓ **Conversion Impacts on In-core Experiments**
- ✓ **Preliminary SAR (PSAR) Preparation**
- ✓ **Transition Cores Developments and Analyses**
- ❑ **Start-up Tests Planning**
- ❑ **Development of LEU Fuel Specifications**
- ✓ **Feasibility Study of Rod-type (UZrH) LEU Fuel**

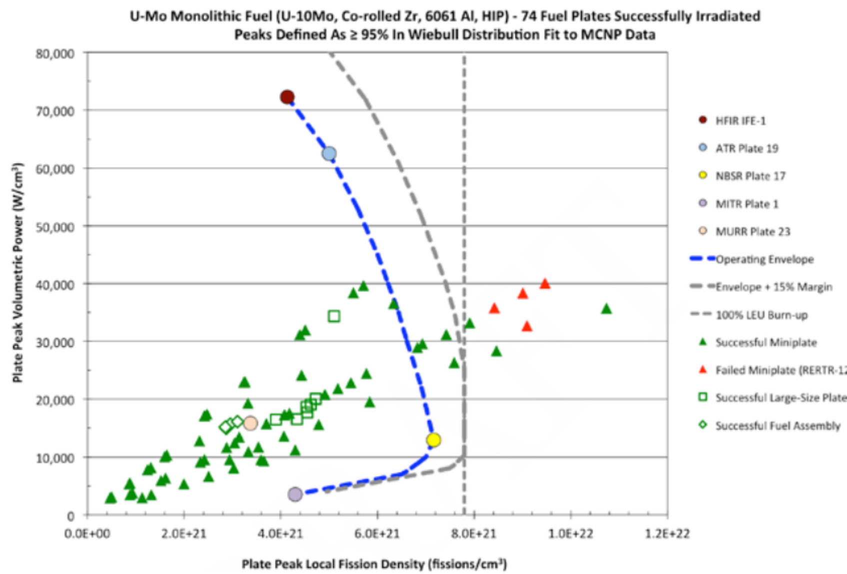


FUEL QUALIFICATION EXPERIMENTS

- The fuel development / qualification efforts for U-Mo monolithic LEU fuel system have been made for more than two decades.
- The final design consists of U-10 (wt. %) Mo fuel foil, a zirconium diffusion barrier layer, and 6061 aluminum cladding.
- Irradiation tests have been completed over a range of fission rates and fission densities. Satisfactory performance is shown.
- **Data gap exists in sufficiently high power range.**

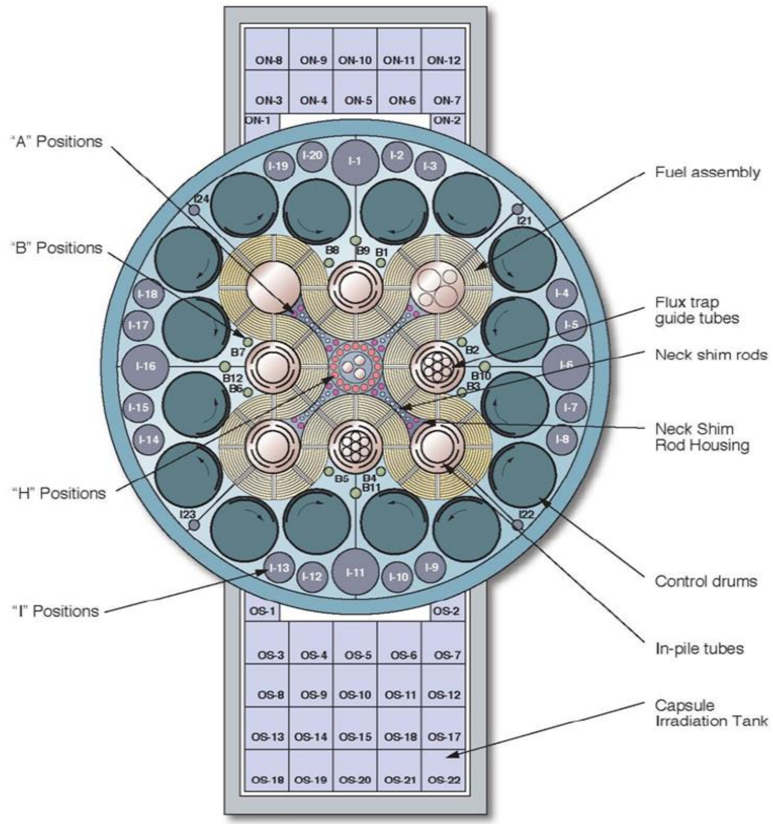






CURRENT RESEARCH BACKGROUND

- Current ATR safety basis (SAR-153) ensures that the plant protection criteria is maintained for all Condition 2 events using Engineering Hot Channel Factors (EHCFs) by verifying that, for Flow Coastdown and Reactivity Insertion Accidents, the Departure from Nucleate Boiling Ratio (DNBR) is **> 2**.
- The basis for this limit is not well defined but may be traced to research reactor licensing based on overly conservative thermal hydraulic criteria.
- This limitation may not be applicable to reactor experiments because the quantity of fissionable material and fission product inventory in experiments is much less than that of the reactor core, and may prevent or limit future tests.
- In particular, fueled experiments may be excluded from irradiation in ATR if the desired fission power **(e.g. high power LEU tests)** cannot be achieved.
- The research will evaluate the DNBR limit using various DNB correlations and consider the impacts of changing the limit to other criteria that have been demonstrated to be a more suitable thermal-hydraulic safety limit for research/test reactor fuel experiments.
- The intent is to maintain an adequate safety margin while simultaneously expanding the ATR capabilities.

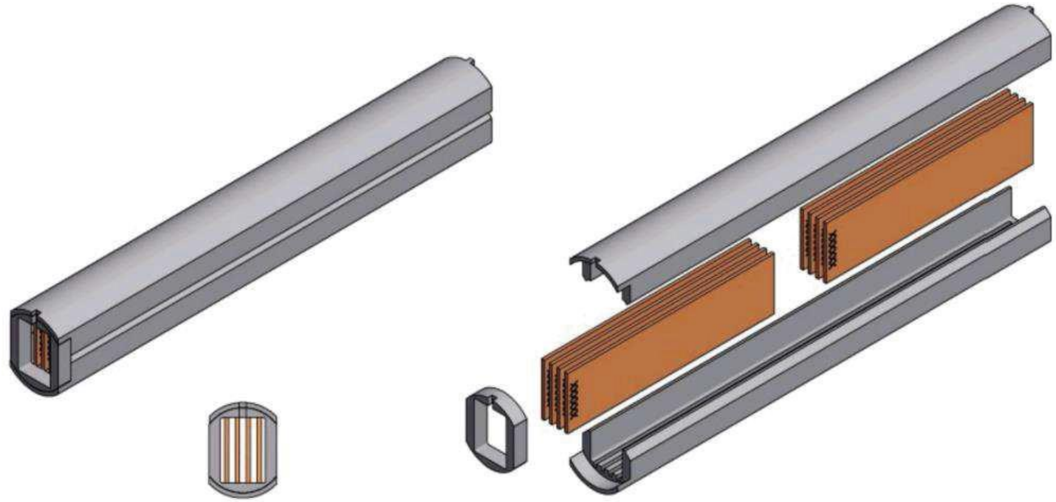


ATR Core and Irradiation Tank*

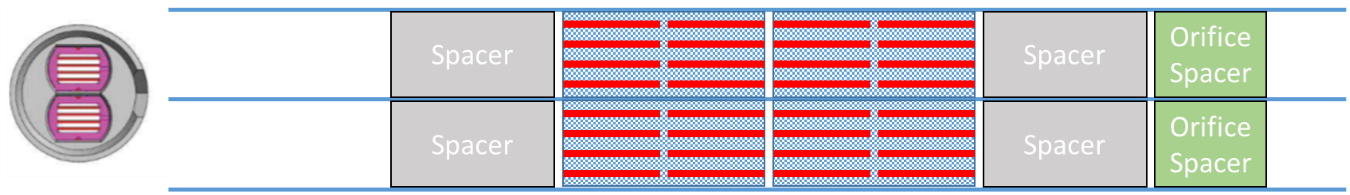
* C. J. Stanley and F. M. Marshall, “Advanced Test Reactor – A National Scientific User Facility,” ICONE-16, 2008.

Experiments Under Consideration

- Mini-Plate-1 (MP-1) High Power Experiment was selected as a test case
- Contains 32 aluminum-clad U-10Mo monolithic fuel plates with a variable orifice spacer that can adjust flow rate
- Before insertion, power limitations were imposed on the experiment to maintain a DNBR > 2.0 during transients, particularly flow coastdown.



MP-1 capsule assembly*



MP-1 SFT configuration*

* C.B. Jensen, et al., “Thermal Analysis of the MP-1 High Power and Medium Power Experiments”, ECAR-2975, INL

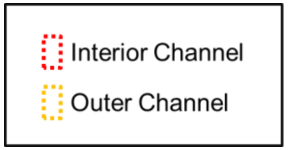
OBJECTIVES

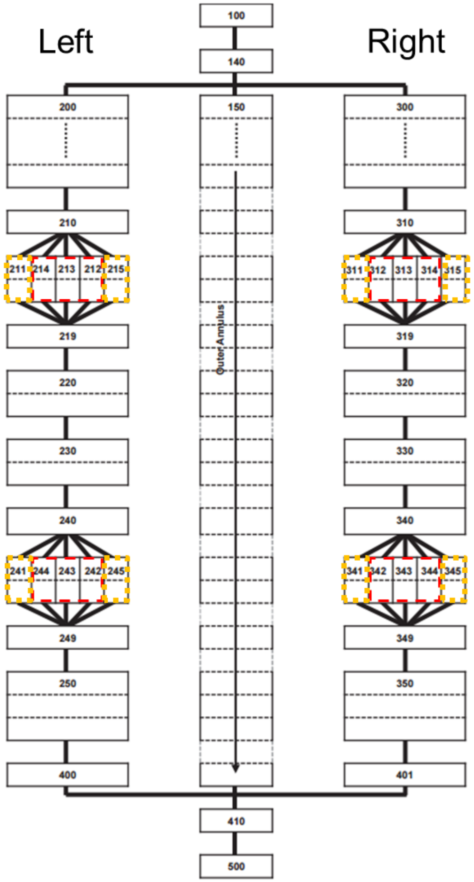
- The research would utilize modern techniques for evaluation of safety margin (e.g., risk-informed safety margin characterization) to adequately quantify the safety margin for ATR experiments.
- An alternative is a BEPU (best-estimate plus uncertainty) statistical approach that maintains 3 sigma from DNBR during condition 2 transients.
- Evaluation of modeling parameters such as DNB correlations and consideration of the uncertainties in those correlations.
- Compare BEPU results of Critical Heat Flux (CHF) with alternative correlations of Onset of Nucleate Boiling (ONB), Onset of Significant Voiding (OSV), and Onset of Flow Instability (OFI).
- Use coupled best-estimated system code (RELAP5-3D) and statistical analysis code DAKOTA and REVAN to perform statistical treatment to thermal-hydraulic parameters, such as power multipliers, dimension tolerance, material properties, and etc.
- It is expected that recommendations can be made with respect to specific criteria that can be established for ATR fuel experiments with adequate safety margin while simultaneously expanding the ATR capabilities.

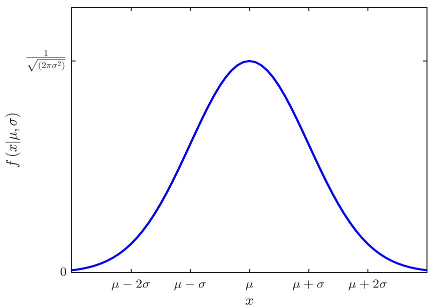
METHODOLOGY

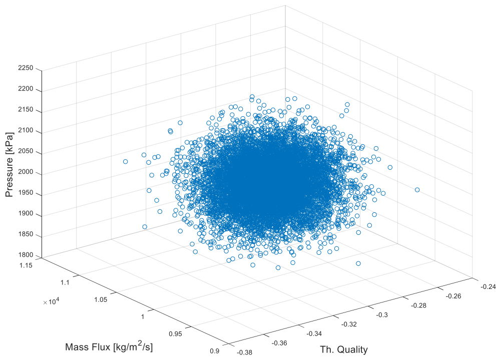
1. Determine independent variables and justify their uncertainties
2. Extract RELAP5 data for each channel and calculate averaged interior/outer channel parameters
3. Generate multivariate sampling distribution
4. Evaluate using TH correlations and post-process data

RELAP5 Model









POWER MULTIPLIERS

	Over-power factor	Power measurement uncertainty	Outer shim flux multiplier	Fissile material loading factor	Plate Power Peaking Factor	P/P_n
	1.277	1.085	1.09	1.15	1.45	
Case 1 Best Estimate						1.00
Case 2						1.82
Case 3 Conservative						2.52

Over-power factor: *Maximum assumed power in lobe divided by nominal*

Power measurement uncertainty: *Epistemic uncertainties associated with instruments*

Outer shim flux multiplier: *Power fluctuation due to full out position of outer shims*

Fissile material loading factor: *Associated with manufacturing of MP-1 plates*

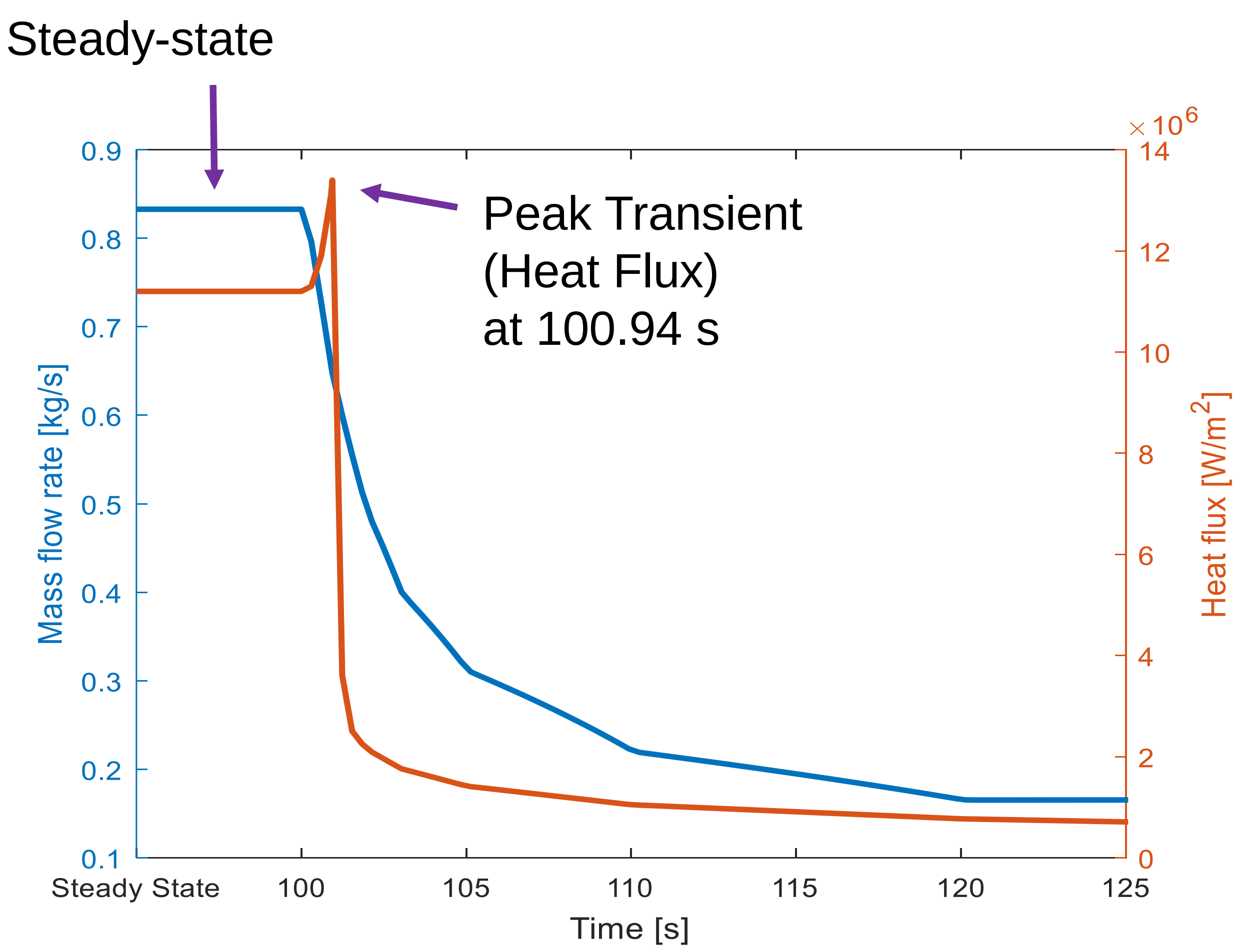
Plate power peaking factor: *Associated with numerical hot spot ratio*

Parameters Sampled

- Uncertainties primarily established through prior reports (ATR-SINDA report from February 1994).
- Values tabulated below are used to generate sampling distributions for T-H correlations

Parameter	Distribution	Distribution Parameters	Justification
Mass flow rate	Normal	$2\sigma/\mu = 0.083$	Coolant flow rate uncertainty noted in ATR-SINDA report
Lobe Power	Normal	$2\sigma/\mu = 0.085$	Power measurement uncertainty is $\approx 8.5\%$.
Pressure	Normal	$2\sigma/\mu = 0.050$	Uncertainty in the pressure drop noted in ATR-SINDA report

Steady-state and Peak Transient



SUBCOOLED CHF CORRELATIONS

CHF Correlation	Applicability
ATR MP-1 High Power Steady State Conditions	$P \approx 2.10$ Mpa $D = 3.5 - 4.5$ mm $\Gamma \approx 13{,}000$ kg/m²/s $\mathrm{V} \approx 15$ m/s $\mathrm{T} \approx 75$ ºC
Savannah River Laboratory 1974	$p \leq 1.03$ MPa $V \leq 21.3$ m/s $T \leq 82.2$ ºC
Groeneveld Look-up Table 2006	$p = 0.1 - 21$ MPa $D = 3.0 - 25.0$ mm $\Gamma = 0 - 8{,}000$ kg/m²/s (expanded with Kalimullah's work)
Hall-Mudawar 2000	$p = 0.1 - 20$ MPa $D = 0.25 - 15$ mm $\Gamma = 300 - 30{,}000$ kg/m²/s

$$\frac{Q_{cr}}{A} = 188{,}000\,[1 + 0.0515\,V(\Gamma, x_{th}, P)] \cdot [1 + 0.069\,T_{sub}(x_{th}, P)]$$

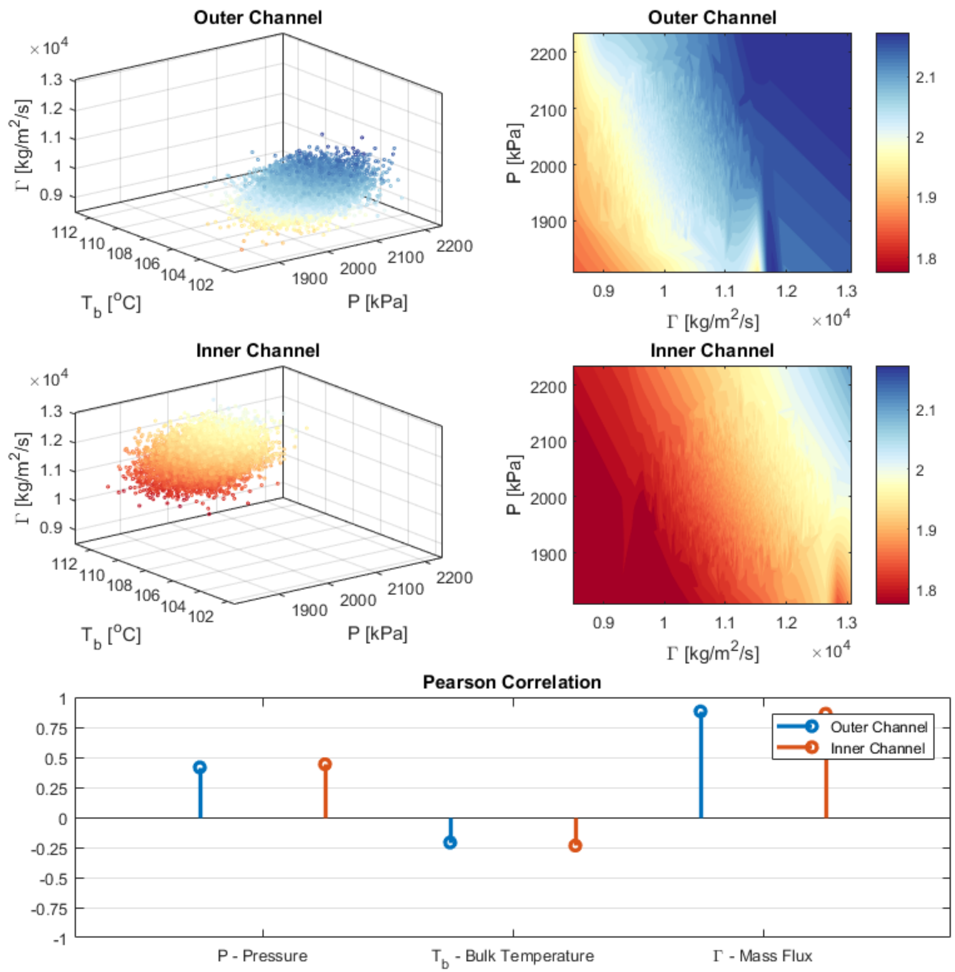
$$q''_{cr} = (q''_{cr})_{LUT} \prod_{n=(1,8)} K_n$$

$$\frac{q''_{cr}}{Gh_{fg}} = C_1 \left(\frac{G^2 D_h}{\rho_f \sigma}\right)^{C_2} \left(\frac{\rho_f}{\rho_g}\right)^{C_3} \left(1 - C_4 \left(\frac{\rho_f}{\rho_g}\right)^{C_5} X_o\right)$$

1. D. H. Knoebel, et. al., "Forced-convection Subcooled Critical Heat Flux," Savannah River Laboratory, 1974.
2. D. Groeneveld, et. al., "The 2006 CHF look-up table," Nuclear Engineering and Design, vol. 237, 2007.
3. Hall and Mudawar, "CHF for water flow in tubes – II. Subcooled CHF correlations," International Journal of heat and Mass Transfer, 2000.

EXAMPLE RESULTS

- 2006 Groeneveld's Critical Heat Flux Look-up Table
- Case 3 (Conservative): $\boldsymbol{P = 2.52 P_n}$
- Peak Transient Condition
- Generated 10^5 normally distributed,
  Latin Hypercube Sampling (LHS) samples
- 3D and 2D contour maps allow further evaluation of DNBR characteristics
- 3σ CHFR:
  - Outer Ch.: 2.041 ± 0.108
  - Inner Ch.: 1.899 ± 0.102
- Parameters ranked with impact on DNBR
  via Pearson Correlation for sensitivity analysis
- Mass flux has the highest impact for this case



CHFR Results Summary

Steady-state

Correlation	Outer Channel Mean	Outer Channel 3σ	Inner Channel Mean	Inner Channel 3σ
$P = 1.00P_n$				
LUT	8.960	0.460	8.678	0.446
SRL	10.170	0.769	10.860	0.824
HM OCC	11.120	0.581	10.760	0.557
$P = 1.82P_n$				
LUT	4.601	0.237	4.403	0.227
SRL	5.394	0.409	5.765	0.436
HM OCC	5.680	0.304	5.422	0.292
$P = 2.52P_n$				
LUT	3.114	0.161	2.952	0.153
SRL	3.799	0.288	4.047	0.305
HM OCC	3.816	0.208	3.600	0.200

Peak Transient

Correlation	Outer Channel Mean	Outer Channel 3σ	Inner Channel Mean	Inner Channel 3σ
$P = 1.00P_n$				
LUT	6.441	0.329	6.193	0.319
SRL	7.020	0.531	7.548	0.573
HM OCC	7.921	0.425	7.598	0.407
$P = 1.82P_n$				
LUT	3.144	0.162	2.981	0.154
SRL	3.580	0.271	3.865	0.291
HM OCC	3.803	0.213	3.578	0.202
$P = 2.52P_n$				
LUT	2.041	0.108	1.899	0.102
SRL	2.486	0.189	2.634	0.200
HM OCC	2.413	0.140	2.216	0.131

- Large safety margins are kept for CHF based DNB
- CHFR reaches slightly below **two** using LUT at Peak Transient condition for Case 3, where all conservative power multipliers are adopted.
- We will further cooperate power multipliers into uncertainty analysis.
- We will also evaluate ONB, OSV, and OFI as potential safety limits for ATR fuel experiments

ONB, OSV, and OFI Correlations

- Bergles-Rohsenow correlation* to predict Onset of Nucleate Boiling (ONB)

$$T_{\mathrm{ONB}} = T_{sat} + 0.556\left[\frac{q''}{1082P^{1.156}}\right]^{0.463P^{0.0234}}$$

- Saha-Zuber correlation to predict Onset of Significant Voiding (OSV)

$$T_{sat} - T_{bulk,D} = 154\left(\frac{q''}{Gc_{pl}}\right)$$

- Ledinegg instability criteria** used to determine Onset of Flow Instability (OFI)

$$\dot{m}_{\mathrm{OFI}} = \frac{Q}{Rc_p(T_{sat} - T_{in})}$$

* A. Bergles and W. Rohsenow, "The determination of forced-convection surface-boiling heat transfer.," Journal of Heat Transfer, 1964.

** N. E. Todreas and M. S. Kazimi, Nuclear Systems: Thermal Hydraulic Fundamentals. Vol 1., CRC Press, 2012.

ONB, OSV, and OFI Margins

Steady-state

Correlation	Outer Channel Mean	Outer Channel 3σ	Inner Channel Mean	Inner Channel 3σ
	$P = 1.00P_n$			
ONB [K]	94.82	3.70	96.33	3.68
OSV [K]	140.40	4.74	140.40	4.61
OFIR	12.550	1.610	11.800	1.520
	$P = 1.82P_n$			
ONB [K]	40.80	3.63	43.33	3.62
OSV [K]	119.10	5.84	118.30	5.55
OFIR	7.130	0.522	6.063	0.406
	$P = 2.52P_n$			
ONB [K]	ONB	-	2.72	3.58
OSV [K]	100.30	6.96	99.59	6.58
OFIR	4.950	0.295	4.310	0.213

Peak Transient

Correlation	Outer Channel Mean	Outer Channel 3σ	Inner Channel Mean	Inner Channel 3σ
	$P = 1.00P_n$			
ONB [K]	57.74	3.56	60.43	3.56
OSV [K]	121.80	5.33	121.00	5.10
OFIR	8.69	0.81	7.21	0.59
	$P = 1.82P_n$			
ONB [K]	ONB	-	ONB	-
OSV [K]	88.82	7.55	87.82	6.87
OFIR	4.53	0.23	3.83	0.18
	$P = 2.52P_n$			
ONB [K]	ONB	-	ONB	-
OSV [K]	60.13	9.62	58.76	8.65
OFIR	3.17	0.12	2.63	0.09

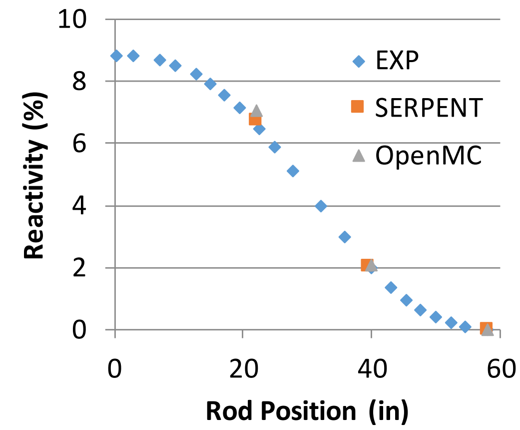
- ONB only occurs at Steady-state for Case 3, where all conservative power multipliers are adopted.
- We will further cooperate power multipliers into uncertainty analysis.
- ONB is not a safety concern during transient.
- Sufficient margins are kept to prevent OSV and OFI at peak flow coastdown transient condition.

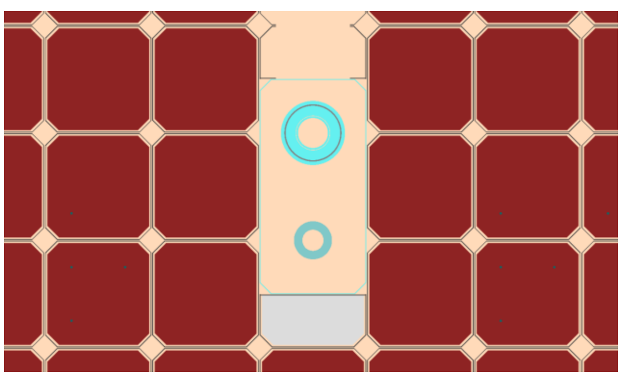
Summary and Future Work

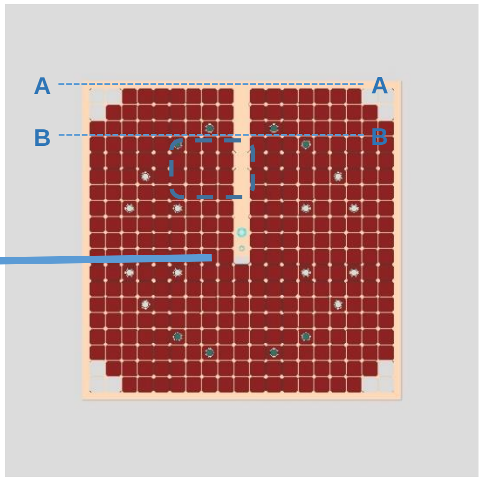
- Intent of the work is to maintain adequate safety margin and expand ATR experimental capabilities, particularly for the MP-1 high power experiment that supporting LEU conversion of US High Performance Research Reactors
- Previous method of determining operating margin for ATR experiments was to use EHCFs (Engineering Hot Channel Factors) and maintain a DNBR > 2 for all Condition 2 transients
- An alternative is a BEPU (best-estimate plus uncertainty) approach that maintains 3 sigma from DNB
- Preliminary assessment of ONB/OSV/OFI/DNB correlations using best-estimate parameters indicate high margin during steady-state and at flow coast-down transient peak condition
- Combination of power multipliers and parameter uncertainties needs additional refining
- Future work involves coupling DAKOTA/RAVEN to RELAP5 and including statistical treatment of other parameters (e.g. power, hydraulic diameter, material properties, etc.)
- In the end, recommendations are expected for more suitable safety basis of ATR fuel experiments

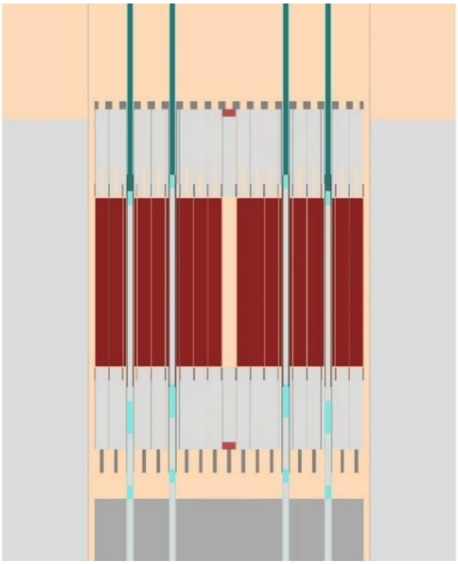
TREAT CONVERSION INVOLVEMENT

- Supporting development of LEU graphite fuel for the Transient Reactor Test (TREAT) Facility at the Idaho National Laboratory (INL)
- TREAT is an air-cooled, graphite reactor capable of pulses up to 18,000 MW
- LEU conversion considering both UO_2 and U_3O_8 particle fuel in graphite matrix
- In-core irradiations in the MITR to reach full lifetime fuel burnup (< 0.1 DPA) at normal (250°C) and maximum (600°C) fuel temperatures
- Zircaloy-encapsulated fuel specimens will undergo post-irradiation examination at MIT and INL





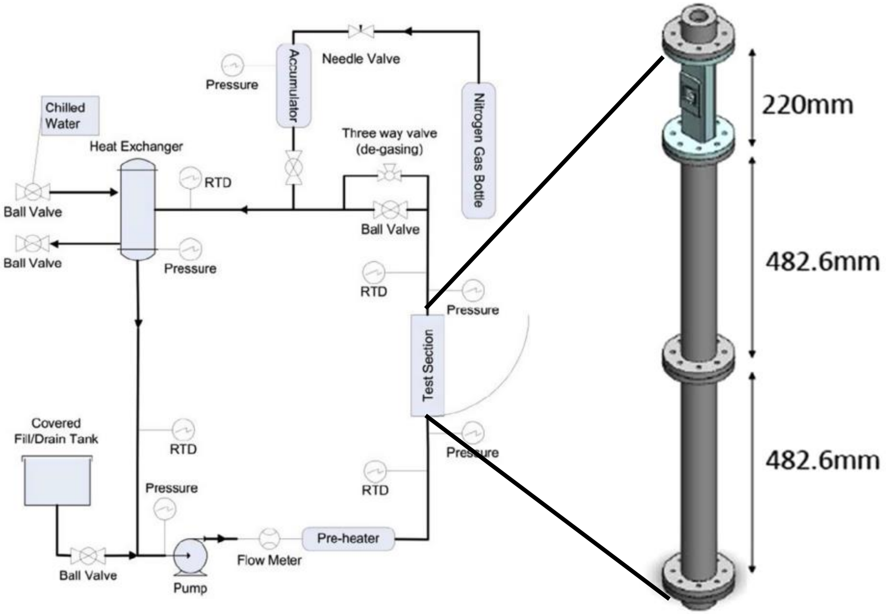




Back-up Slides…

TRANSIENT BOILING LITERATURE

- Transient boiling of water
  - G-Y Su, M. Bucci, T. McKrell, J. Buongiorno, "**Transient boiling of water under exponentially escalating heat inputs. Part I: Pool boiling**," International Journal of Heat & Mass Transfer, 2016.
  - G-Y Su, M. Bucci, T. McKrell, J. Buongiorno, "**Transient boiling of water under exponentially escalating heat inputs. Part II: Flow boiling**," International Journal of Heat & Mass Transfer, 2016.
- Exponential power escalations with periods in the range from 5 to 500 ms, and subcooling of 10, 25 and 75 K were explored. The Reynolds number was varied from 25,000 to 60,000. All experiments conducted at 1 bar.
- Found that for short periods, single phase heat transfer coefficient was $\propto 1/\sqrt{\tau}$



Transient Boiling Literature

- During short periods, it was found that transient conduction becomes more important
- For short periods, single phase heat transfer coefficient was $\propto 1/\sqrt{\tau}$

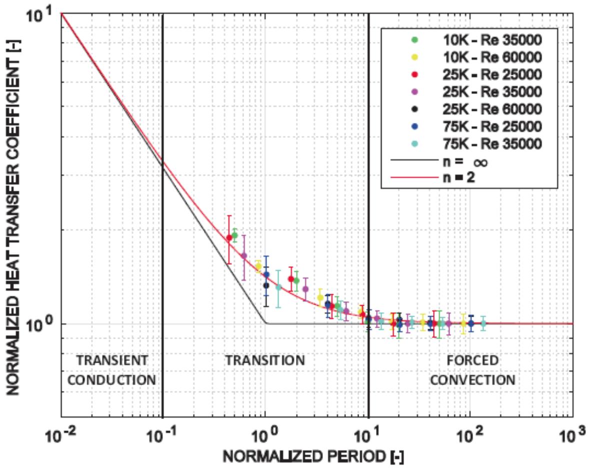


Fig. 9. Normalized heat transfer coefficient vs. normalized period (all tests).

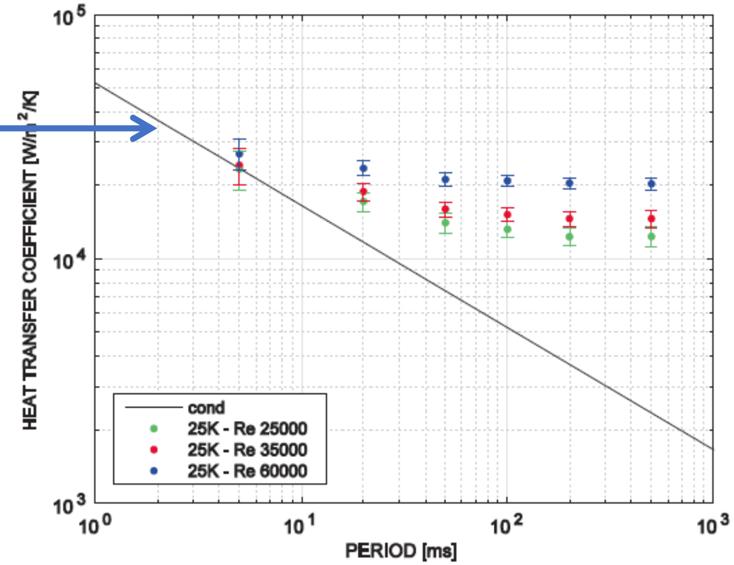
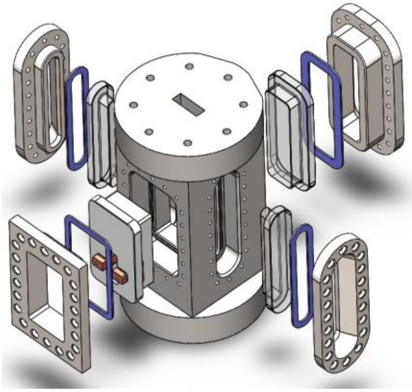
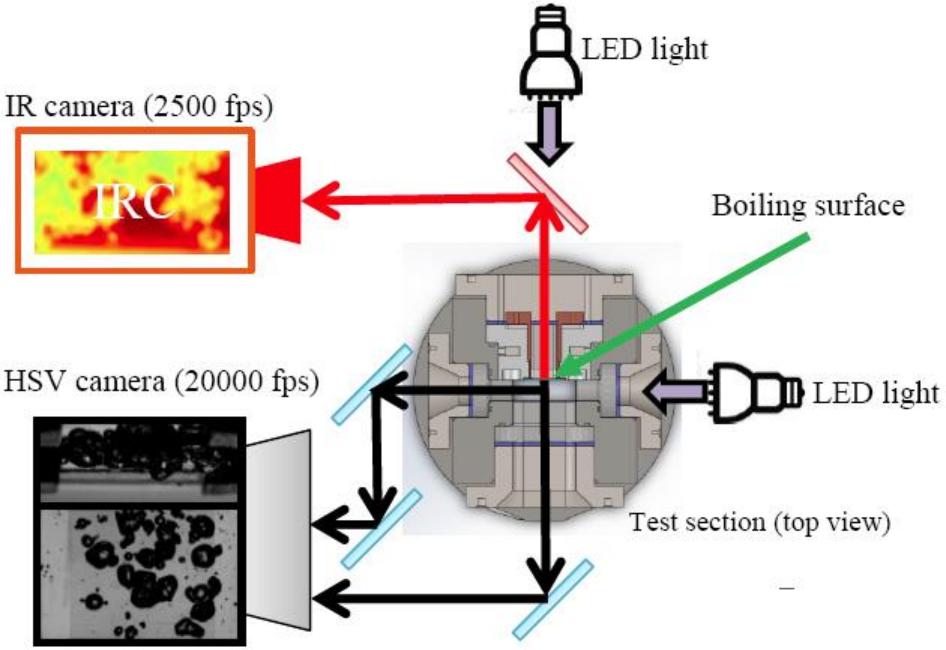


Fig. 6. Single-phase heat transfer coefficient vs. period for different Reynolds number (tests at 25 K subcooling).

Transient CHF Literature

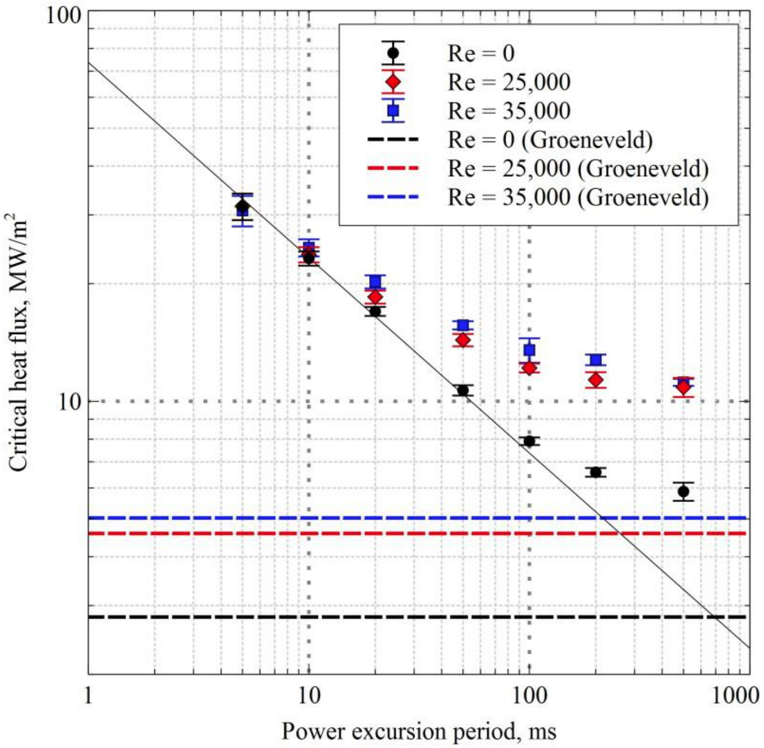
- Transient CHF under exponentially escalating heat
  - A. Kossolapov, T. McKrell, M. Bucci, J. Buongiorno, “**Transient flow boiling CHF under exponentially escalating heat inputs.**” 9th World Conference on Experimental Heat Transfer, Fluid Mechanics and Thermodynamics, Brazil.





Transient CHF Literature

- It was, again, found that for short periods (< 10 ms), CHF $\propto 1/\sqrt{\tau}$
- It was also found that CHF is independent of Re for short periods



APPLICATION TO MP-1 EXPERIMENT

- MP-1 coastdown transient power curve taken from RELAP5 input deck
- Exponential fit for power excursion yields a period of 5.688 s
- $\frac{1}{\sqrt{\tau}} \approx 0.42$