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A Statistical Method for Benchmarking Nuclear Reactor Plant Models, for Use in Simulators, Using the Automated Code Assessment Program

John McCloskey
Richard Smith

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Outline

- Motivation
- Overview of ACAP
- Description of method
- Demonstration of method using examples from LOFT L2-5 and RELAP5-3D data comparisons

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Conventional Validation Methods

- U. S. Nuclear Regulatory Commission (NRC) defines a set of terms for level of agreement: Excellent, Reasonable, Minimal, Insufficient
  - Requires subjectivity
  - Time consuming
- Experimental Uncertainty
  - Can be difficult to determine an uncertainty

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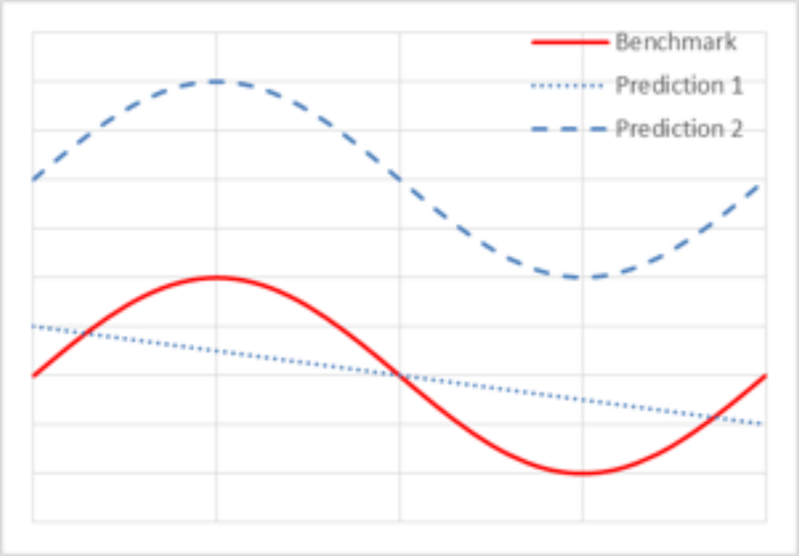
ACAP Overview

- ACAP-Automated Code Assessment Program
  - Developed by Pennsylvania State University under contract by U.S. NRC
  - Runs with a graphical user interface or in batch mode
- Compares nuclear reactor systems code with experimental measurements or a qualified benchmark code
- 0-D, steady state, or transient data
- Data resampling
- Contains 16 statistical metrics
- Figure of Merit (FOM): Statistical level of agreement non-dimensionalized from 0 to 1



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Proposed Method

- Applicable to transient data
- Applicable to nuclear operator training simulator applications
- American Nuclear Society ANSI/ANS-3.5-2009-Nuclear Power Plant Simulators for use in Operator Training and Examination
  - For normal transient evolutions and malfunctions it is required that "any observable change in simulation parameters corresponds in direction to the change expected from actual or best estimate response"
- Quantitative method
- Easy to document
- Automated

ACAP Metrics

- Four metrics are chosen for transient simulator applications
  - Conservative
  - Emphasis on trend errors
  - Automated
    - Avoid scaling, filtering, other inputs

Metrics	
Mean Error	*Index of Agreement*
Standard Deviation of Error	*Cross-Correlation Coefficient*
Mean Square Error	*L_2 Norm of Standard Linear Regression*
Mean Error Magnitude	*L_2 Norm of Standard Linear Regression Constrained Through Origin*
Mean Relative Error	*L_2 Norm of Difference Between Predicted and Perfect Agreement*
Mean Fractional Error	*Percent Validated*
Systematic Mean Square Error	*D'Auria FFT*
Unsystematic Mean Square Error	*Continuous Wavelet Transform*

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Percent Validated (PV)

$$PV \equiv \frac{1}{N}\sum_{i=1}^{N} \max\left[\frac{|EU - |P_i - O_i||}{EU - |P_i - O_i|}, 0\right]$$

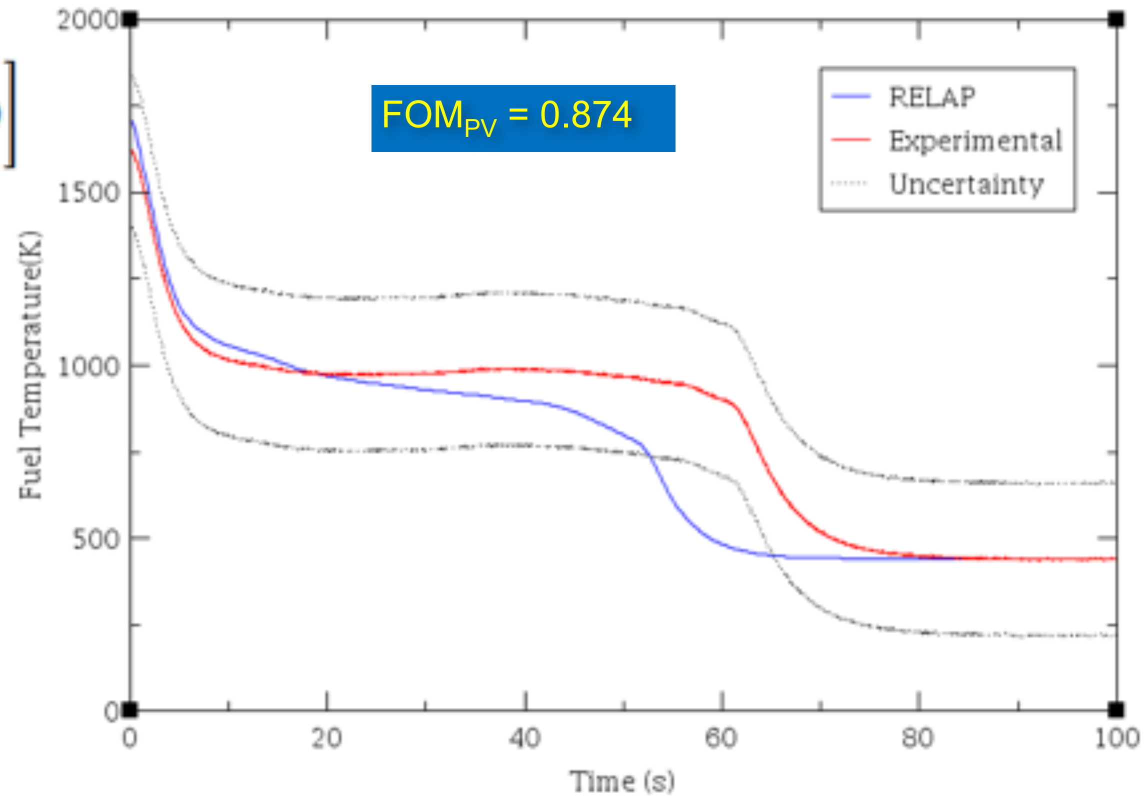
N → Number of Data Points

P_i → Computed Data

O_i → Benchmark Data

EU → Experimental Uncertainty

$$FOM_{PV} \equiv PV$$



RELAP: "Developmental Assessment of RELAP5-3D Version 2.9.3+" INL/EXT-09-15965

Experimental: Loss-Of-Fluid-Test (LOFT) Facility Large Break Loss-Of-Coolant Experiment L2-5

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Experimental Uncertainty

- ANS-3.5 Steady State Requirements:
  - "It shall be demonstrated that the following PWR parameters match reference data within __% of the reference unit instrument loop range."

1% of Range	2% of Range	10% of Range
Temperature (T)-average	*Steam generator feed flow*	*All other parameters*
T-hot	*Reactor coolant system flow*	
T-cold	*Steam generator level*	
Core MWt	*Letdown flow*	
Power range nuclear instrumentation readings	*Charging flow*	
Reactor coolant system pressure	*Steam flow*	
Steam generator pressure	*Turbine first stage pressure*	
Pressurizer level	*MWE*	

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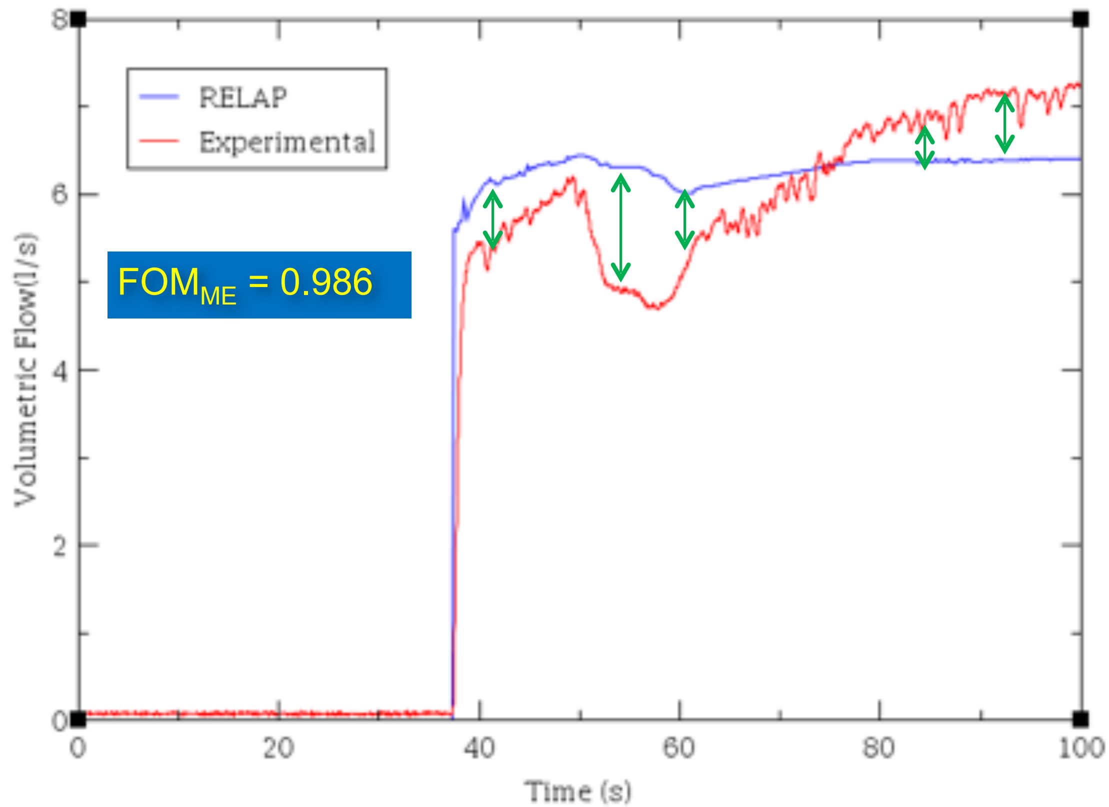
Mean Error (ME)

$$ME^* \equiv \frac{1}{N}\sum_{i=1}^{N}(O_i^* - P_i^*)$$

$$P^* \equiv \frac{P}{|O_{max} - O_{min}|}$$ $P \rightarrow$ Computed Data

$$O^* \equiv \frac{O}{|O_{max} - O_{min}|}$$ $O \rightarrow$ Benchmark Data

$$FOM_{ME} \equiv \frac{1}{|ME^*| + 1}$$



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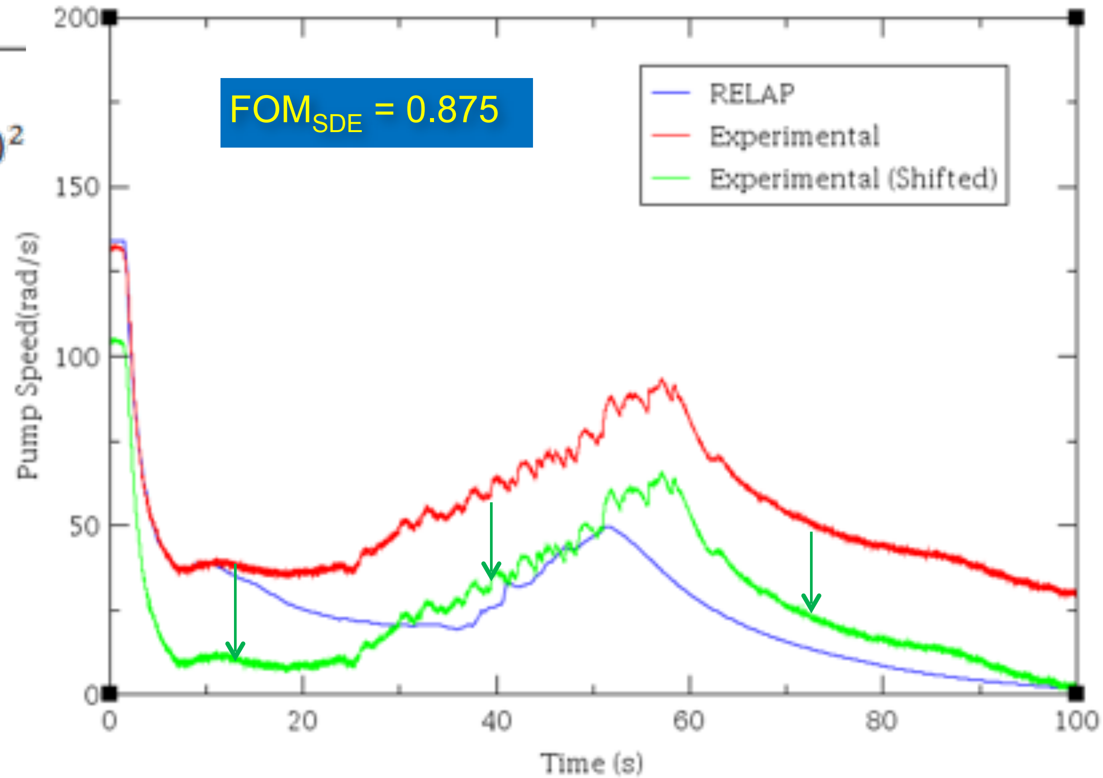
Standard Deviation of Error (σ)

$$\sigma^* \equiv \sqrt{VE^*} \equiv \sqrt{\frac{1}{(N-1)}\sum_{i=1}^{N}(O_i^* - P_i^* - ME^*)^2}$$

$$P^* \equiv \frac{P}{|O_{max} - O_{min}|}$$ P → Computed Data

$$O^* \equiv \frac{O}{|O_{max} - O_{min}|}$$ O → Benchmark Data

$$FOM_\sigma \equiv \frac{1}{|\sigma^*| + 1}$$



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Cross-Correlation Coefficient (ρ_{xy})

$$\rho_{xy} \equiv \frac{\sum_{i=1}^{N}(O_i - \overline{O})(P_i - \overline{P})}{\sqrt{\left[\sum_{i=1}^{N}(O_i - \overline{O})^2\right]\left[\sum_{i=1}^{N}(P_i - \overline{P})^2\right]}}$$

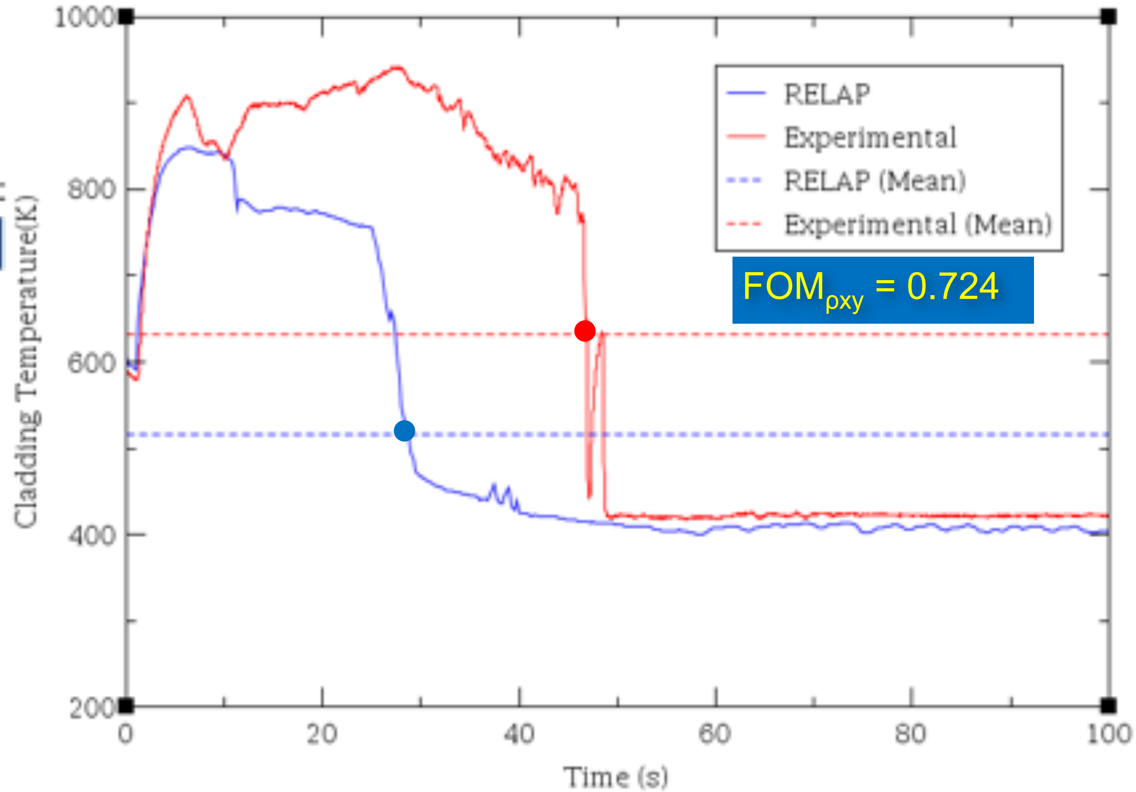
$P \rightarrow$ Computed Data

$O \rightarrow$ Benchmark Data

$\overline{P} \rightarrow$ Average of Computed Data

$\overline{O} \rightarrow$ Average of Benchmark Data

$$\mathrm{FOM}_{\rho_{xy}} \equiv \max(\rho_{xy}, 0)$$



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Weightings for Nuclear Operator Training Simulator Applications

$$\mathrm{FOM_{total}} \equiv \frac{1}{3}\mathrm{FOM_{PV}} + \frac{1}{3}\mathrm{FOM}_{\rho_{xy}} + \frac{1}{6}\mathrm{FOM}_{\sigma} + \frac{1}{6}\mathrm{FOM_{ME}}$$

Method	Description	Trend Errors	Magnitude Errors	No Inputs Required	Independent of Benchmarking Range	Translationally Invariant	Applicable to Steady State
Percent Validated (PV)	Percentage that data is within tolerance band	X	X		X	X	X
Cross-Correlation Coefficient (ρ_{xy})	How often data are both above or both below their respective mean	X		X	X	X	
Standard Deviation of Error (σ)	Difference in trend after removing mean error	X		X		X	
Mean Error (ME)	Difference in means		X	X		X	

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FOM Threshold

- Used as an aid to highlight potential problems
- Threshold depends on type of test
- FOMs are not used for pass/fail decisions

FOM Threshold	Type of Test
0.7	*Loss of Coolant Accidents and Steam Line Ruptures*
0.9	*Operational Transients and Non-Leak Accidents*
0.99	*Computer Hardware Changes, Model Changes*, Tool Upgrades, and Operating System Upgrades*

*Model changes not intended to change the benchmark results

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FOM Summary

- Table can be auto-generated
- Can easily identify parameters most likely to exhibit a discrepancy
- Can identify patterns
- Table can be easily updated after model changes and compared with previous results
- Easy to document results in a report

Comparison Between LOFT Loss of Coolant Experiment L2-5 Data with RELAP5-3D Predictions

Parameter	Mean Error	Standard Deviation of Error	Cross-Correlation Coefficient	Percent Validated	Combined FOM
Reactor Pressure	0.9953	0.9888	0.9977	0.8180	0.9359
Steam Generator Pressure	0.9407	0.9478	0.9605	0.9655	0.9567
Pressurizer Liquid Level	0.9896	0.9872	0.9977	0.9587	0.9816
Mass Flow Rate Cold Leg Broken Loop	0.9969	0.9582	0.9442	0.5904	0.8374
Mass Flow Rate Hot Leg Broken Loop	0.9888	0.9624	0.9053	0.9083	0.9297
Mass Flow Rate Hot Leg Intact Loop	0.9927	0.9574	0.5326	0.7151	0.7409
Mass Flow Rate Cold Leg Intact Loop	0.9552	0.9339	0.8037	0.4130	0.7204
Primary Coolant Pump Speed	0.7900	0.8745	0.7428	0.1880	0.5877
Density Cold Leg Broken Loop	0.9890	0.8648	0.6791	0.8523	0.8194
Density Hot Leg Broken Loop	0.9657	0.8975	0.6887	0.6836	0.7679
Density Hot Leg Intact Loop	0.8844	0.8492	0.3824	0.5425	0.5972
Density Cold Leg Intact loop	0.9880	0.7836	0.3261	0.2614	0.4911
Accumulator Liquid Level	0.9983	0.9877	0.9996	1.0000	0.9975
High-Pressure Injection System Flow	0.9906	0.9269	0.9153	0.9596	0.9445
Low-Pressure Injection System Flow	0.9856	0.9203	0.9785	0.9952	0.9756
Primary Coolant Temperature	0.9723	0.9622	0.9941	0.3853	0.7822
Primary Coolant Temperature	0.9304	0.9296	0.9844	0.3781	0.7642
Fuel Centerline Temperature	0.9410	0.9059	0.9075	0.8742	0.9017
Fuel Cladding Temperature	0.8578	0.7986	0.7857	0.7869	0.8002
Fuel Cladding Temperature	0.8199	0.7720	0.7235	0.6440	0.7211
Fuel Cladding Temperature	0.8089	0.7766	0.7609	0.5616	0.7051
Fuel Cladding Temperature	0.8641	0.8359	0.9049	0.5763	0.7771
Fuel Cladding Temperature	0.8467	0.8317	0.8862	0.5230	0.7495
Fuel Cladding Temperature	0.9478	0.9151	0.9808	0.9064	0.9396
Fuel Cladding Temperature	0.9424	0.8575	0.8912	0.8468	0.8794
Fuel Cladding Temperature	0.9793	0.8284	0.7780	0.7270	0.8030
Fuel Cladding Temperature	0.8976	0.8367	0.8581	0.8365	0.8539
Fuel Cladding Temperature	0.9124	0.8561	0.8863	0.9052	0.8919
Fuel Cladding Temperature	0.8847	0.8558	0.8282	0.7956	0.8314
Fuel Cladding Temperature	0.8958	0.8735	0.8528	0.9391	0.8922

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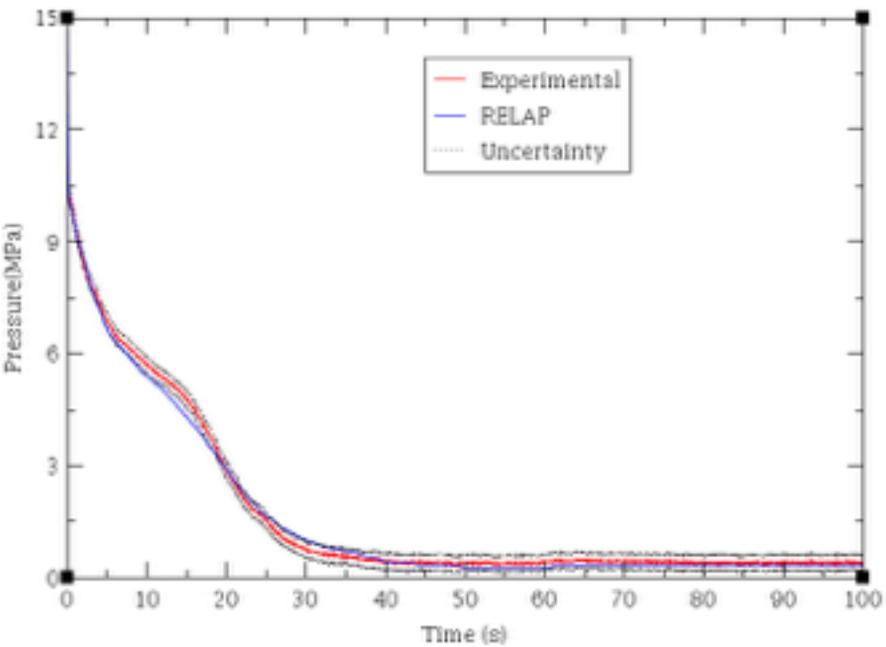
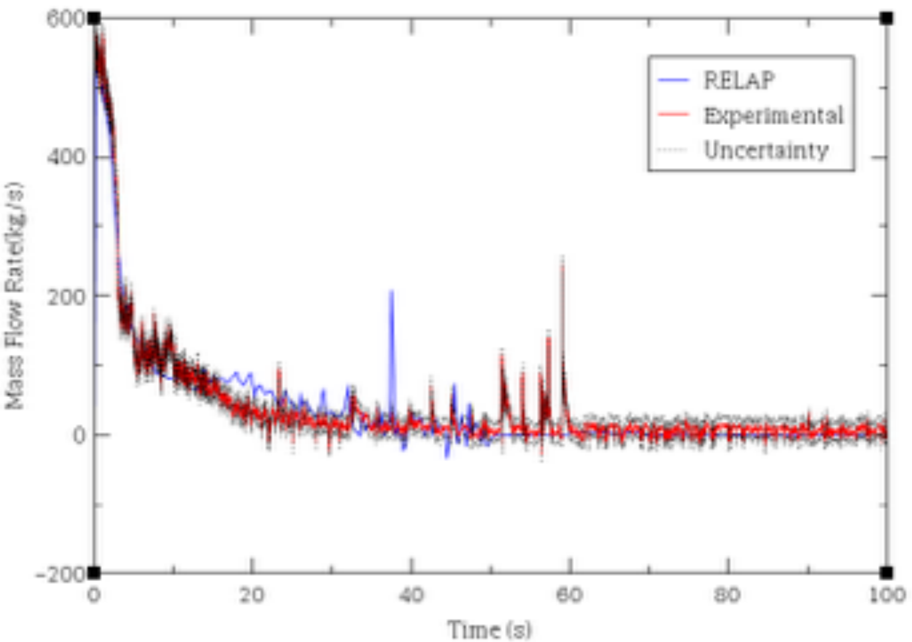
Experimental: Loss-Of-Fluid-Test (LOFT) Facility Large Break Loss-Of-Coolant Experiment L2-5

Samples-High FOMs

Percent Validated	0.818
Mean Error	0.995
Standard Deviation Error	0.989
Cross-Correlation Coefficient	0.998
Combined Figure of Merit	0.936

Percent Validated	**0.590**
Mean Error	0.997
Standard Deviation Error	0.958
Cross-Correlation Coefficient	0.944
Combined Figure of Merit	0.837





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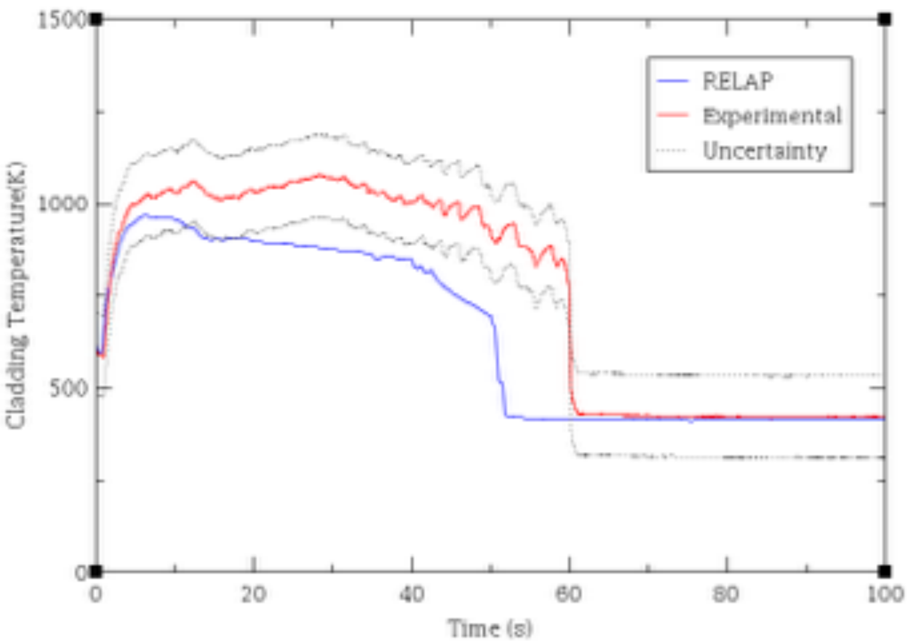
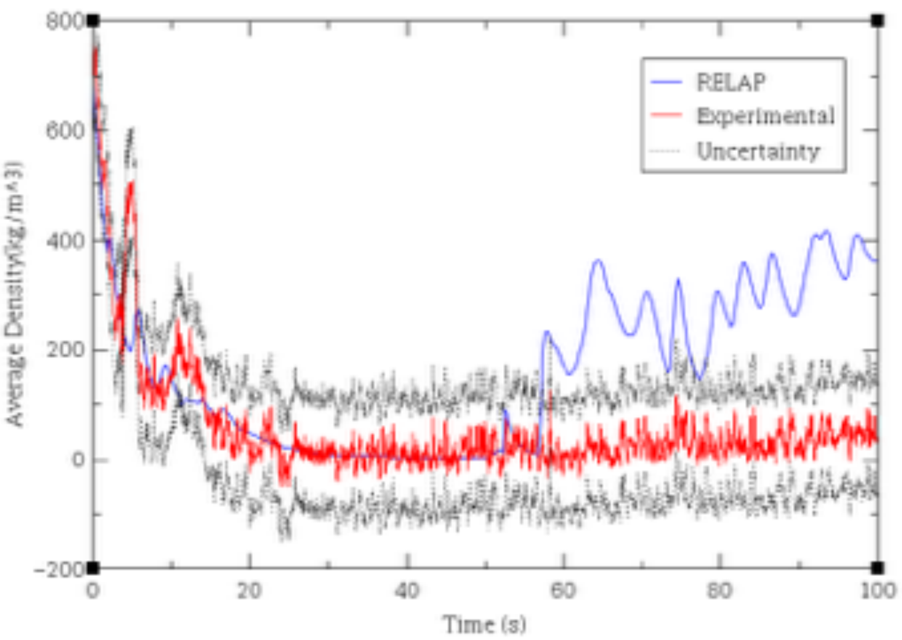
Experimental: Loss-Of-Fluid-Test (LOFT) Facility Large Break Loss-Of-Coolant Experiment L2-5

Samples-Low FOMs

Percent Validated	**0.523**
Mean Error	0.847
Standard Deviation Error	0.832
Cross-Correlation Coefficient	0.886
Combined Figure of Merit	0.750

Percent Validated	**0.543**
Mean Error	0.884
Standard Deviation Error	0.849
Cross-Correlation Coefficient	**0.382**
Combined Figure of Merit	**0.597**





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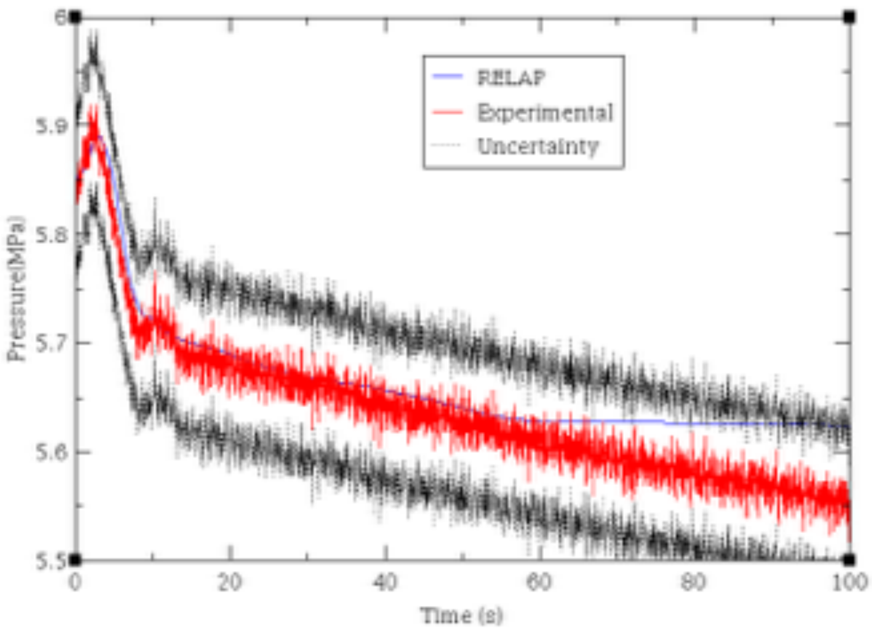
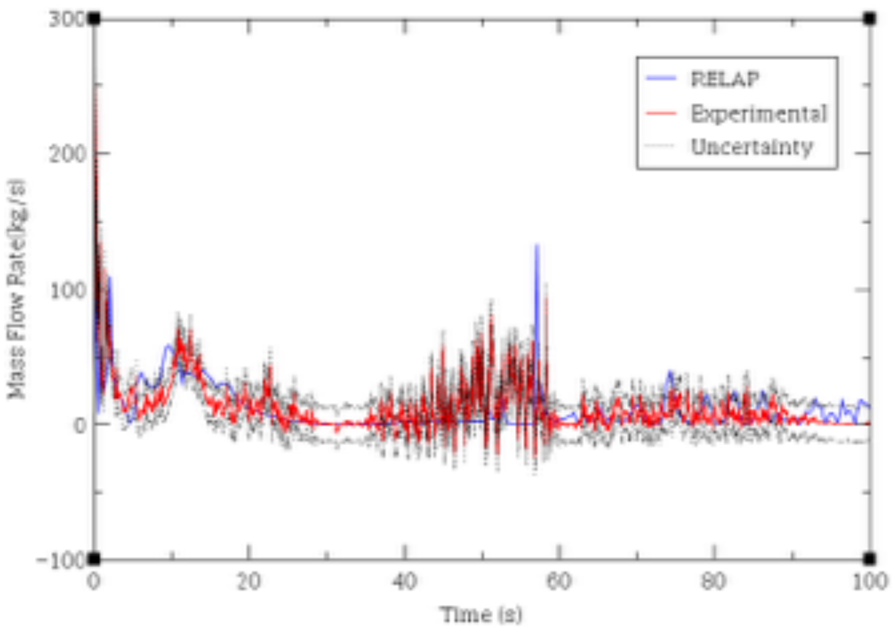
Samples-Data Noise

Percent Validated	0.966
Mean Error	0.941
Standard Deviation Error	0.948
Cross-Correlation Coefficient	0.961
Combined Figure of Merit	0.957

Percent Validated	0.715
Mean Error	0.993
Standard Deviation Error	0.957
Cross-Correlation Coefficient	**0.533**
Combined Figure of Merit	0.741





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Conclusions

- ACAP was used to aid in the validation of nuclear reactor plant models
  - Quantitative
  - Automated
  - Conservative
  - Not used for pass or fail decisions
- Four metrics chosen for transient simulator applications
  - Percent Validated
  - Mean Error
  - Standard Deviation Error
  - Cross-Correlation Coefficient
- Examples shown comparing LOFT L2-5 experimental data with a RELAP5-3D model.

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