Evaluation of Oak Ridge National Laboratory Health Physics Research Reactor Operation Data for Critical Benchmark Creation

Mathieu Dupont
Ph. D.
Oak Ridge National Laboratory

Project Overview

- FY19-20 US DOE NCSP funded project: NCSP Task IP&D-5
- Use available data from Health Physics Research Reactor (HPRR) operation to create a benchmark report for inclusion in the ICSBEP, as a Criticality Accident Alarm System (CAAS) **shielding** benchmark
- In this talk, focus is given to create a **critical** experiment benchmark. Spoiler alert: It does not look good

The Health Physics Research Reactor

- The HPRR or Fast Burst Reactor (FBR), was designed and built at ORNL in 1961
- Part of the Dosimetry Application Research (DOSAR) facility in ORNL from 1963 to 1987
- Operated for thousands of hours, achieved criticality nearly 10,000 times
- Numerous studies and publications, involving dosimetry, plants radiobiology, radiation alarms, teaching and training

DOSAR Facility, A History of Research Reactors Division (1987)

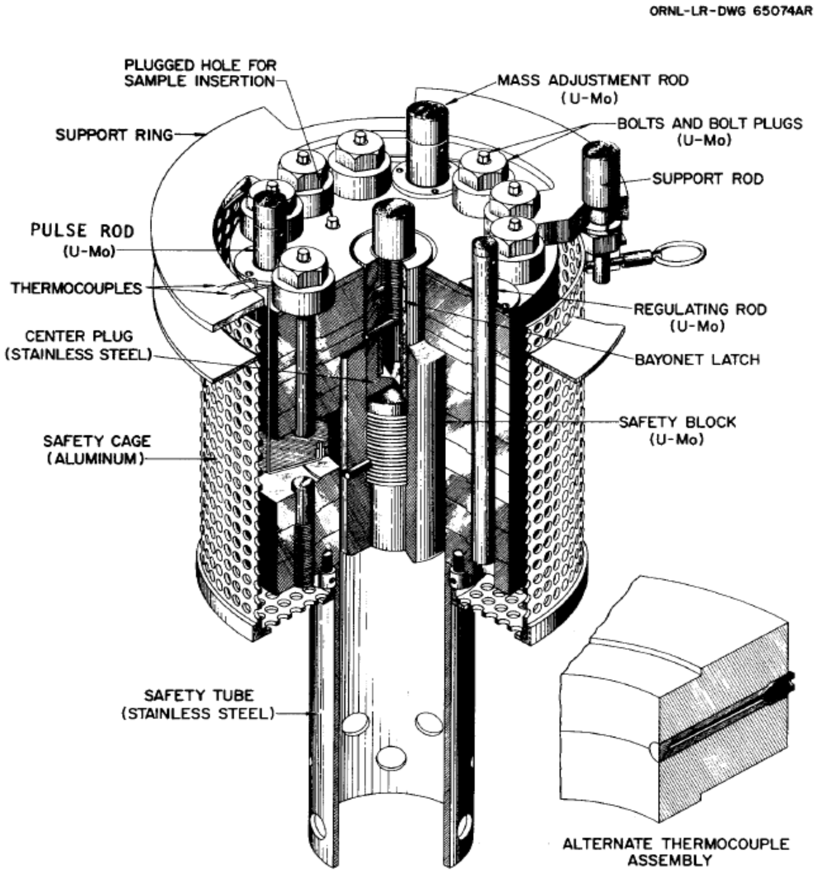
OAK RIDGE National Laboratory

ANS

The Health Physics Research Reactor

- The HPRR is a fast reactor: Unshielded, unmoderated, highly enriched (93.15%) U-Mo alloy (90% U) core

- U-Mo inventory:
  - 11 U-Mo annulus plates
  - 9 U-Mo partially hollow bolts
  - 9 bolt inserts
  - 3 control rods
  - 1 sample irradiation hole
  - 1 safety block (center cylinder)



Operation Bren, CEX 62-02 (1965)

The Health Physics Research Reactor

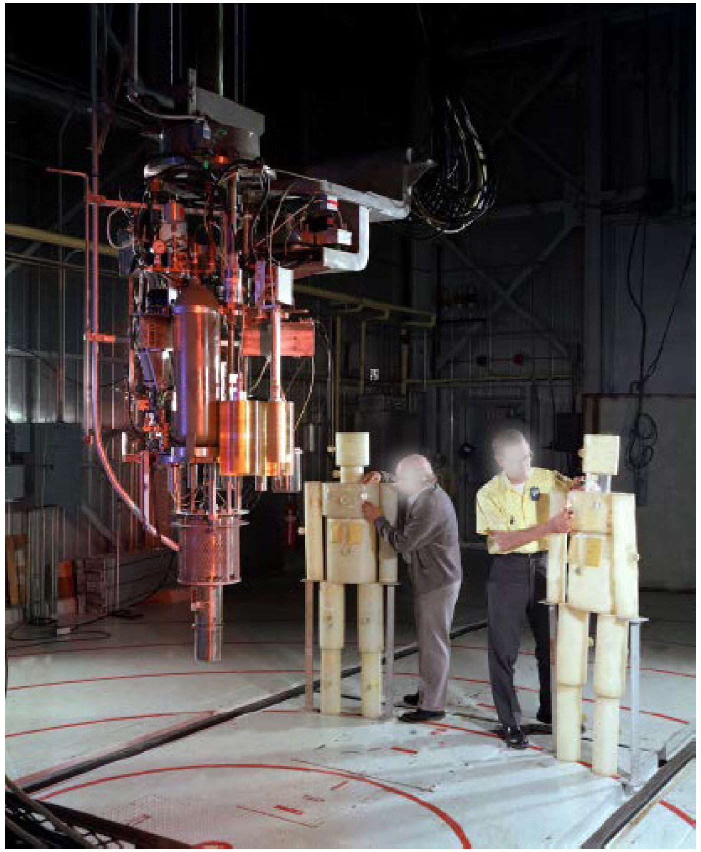


Figure 53: HPRR

A History of Research Reactors Division (1987)

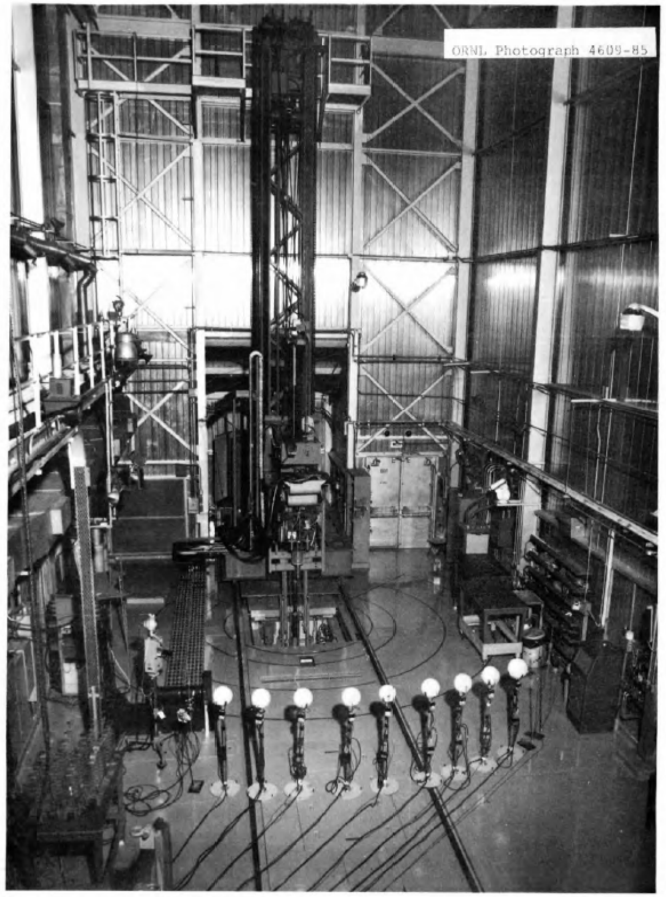


Figure 1. HPRR in experimental position

Health Physics Research Reactor Reference Dosimetry, ORNL-6240 (1987)

OAK RIDGE National Laboratory

ANS

Experiments of Interest

- A lot of experimental data is available, with a varying level of detail
- Three experiments are considered of potential value for a critical benchmark:

1. Sub-critical and critical operation of the HPRR, from University of Tennessee students and/or Senior Reactor Operator training
    - Goal: To show the influence of the position of the control rods on the reactor reactivity
    - Pros: Explicit rod position
    - Cons: Performed in 1974, accuracy is questionable, and core configuration was different

Experiments of Interest

2. Steady-state critical operation of the HPRR, from Steady-state Log Sheets
   - Goal: Irradiation of samples for a longer time and lower intensity than during burst operation
   - Pros: Hundreds of operations, performed not long before reactor decommissioning
   - Cons: General lack of information on some parameters

Experiments of Interest

3. Sub-critical configuration of the HPRR before Burst operation, recorded in Burst Log Sheets
   - Goal: Necessary step before initiation of a burst
   - Pros: Hundreds of operations, performed not long before reactor decommissioning, two separate measurements of subcritical reactivity, different configuration compared to steady-state critical (Burst Rod is fully out)
   - Cons: General lack of information on some parameters

Experiments of Interest

Evaluation Number	1	2	3	4	5	6	7
Origin	Training sheet			Logbook			
Operation Number	1469			1469	2881	2883	2946
Date	4/9/1974			4/9/1974	1/3/1986	1/7/1986	5/29/1986
Height above floor (m)	1			1	1.43	1.4	1.4
Safety Block (in)	-0.135			-0.135	-0.113	-0.116	-0.13
Regulating Rod (in)	0	2.5	8.24	8.24	7	7	7
Mass Adjustment Rod (in)	6.515	6.31	**5.821**	**6**	6.487	6.734	6.227
Burst rod (in)	IN	IN	IN	IN	IN	IN	IN

Evaluated Critical Experiments

Experiments of Interest

Evaluation Number	1	2	3	4
Origin	Training sheet		Logbook	
Operation Number	1469		B1014	B1016
Date	4/9/1974		10/29/1985	12/11/1985
Height above floor (m)	1		1.44	1.4
Safety Block (in)	-0.135		-0.112	-0.115
Regulating Rod (in)	2.5	4.5	0	0
New Regulating Rod (in)	-	-	1.4	1.1
Mass Adjustment Rod (in)	6.515	6.31	3.38	3.84
Burst rod (in)	IN	IN	OUT	OUT
Reactivty 1 (cents)	-4.9	-5.3	-2.8	-2.23
Reactivity 2 (cents)	-	-	-2.75	-2.23

Evaluated Sub-Critical Experiments

Evaluation of Experimental Data

- **A lot of missing and contradictory data:**
  - No uncertainty on U-Mo composition and density
  - U-Mo coating issues
  - Regulating rod is U-Mo or Aluminum
  - Sample irradiation hole plug length
  - Building walls, concrete material composition and dimensions
  - What was actually inside the building during operation
  - **Lack of material and dimension information**

- Uncertainty study performed with SCALE 6.2.4 KENO-VI to determine the influence on those parameters on k_{eff}

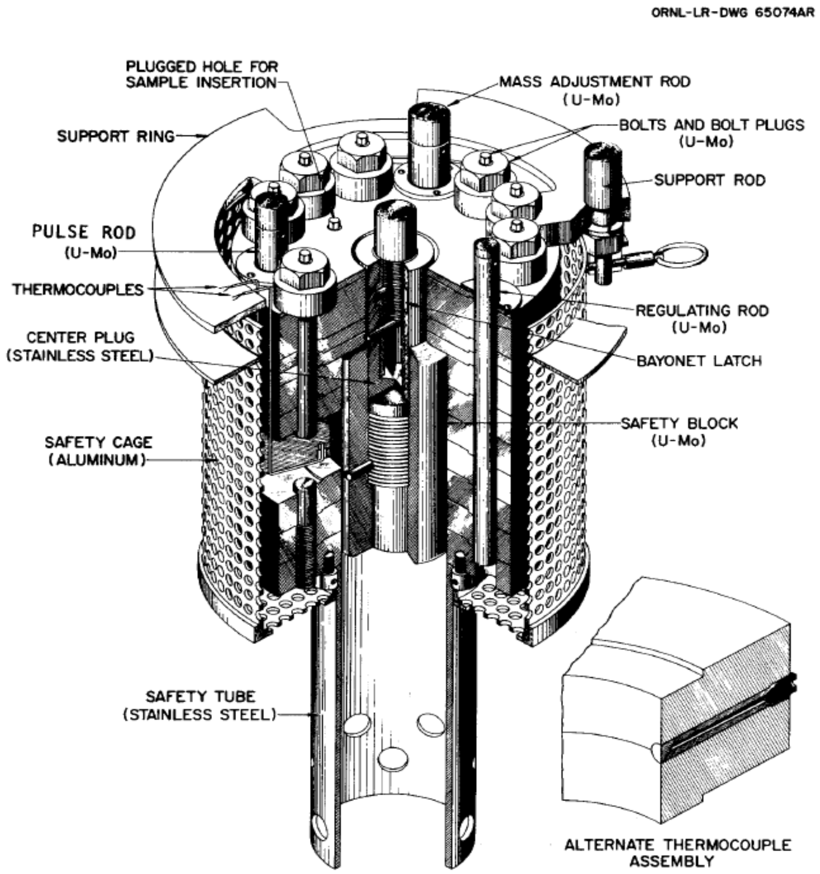
Evaluation of Experimental Data

- **Observations:**
  - Very high uncertainty, ~3.8% relative on k_{eff}
  - Main contributor is fuel alloy density, evaluated with the ICSBEP guide to express uncertainties guidelines
  - Low uncertainty due to the rod position

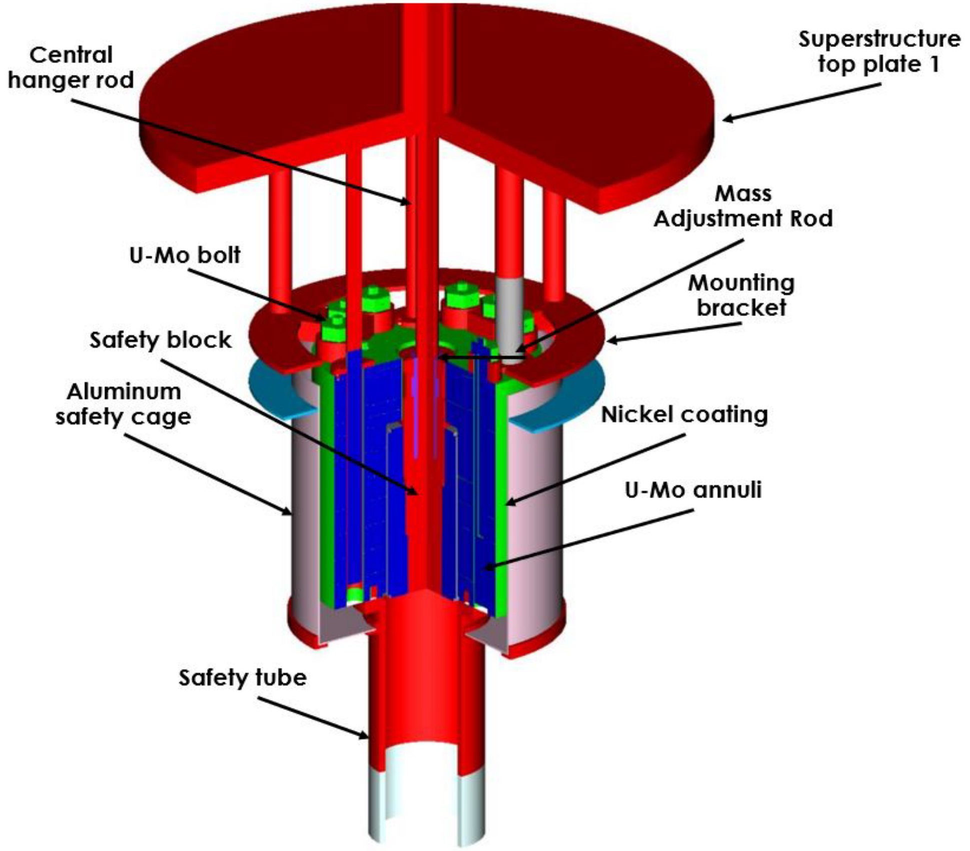
Element	k_{eff} Uncertainty (pcm)
Burst Rod position	4
Mass Adjustment Rod position	100
Regulating Rod position	40
Safety Block position	749
Fuel Uranium content	142
Fuel Molybdenum content	Negligible
Fuel alloy density (g/cm^3)	3668
Fuel ^{235}U content	139
Core elements Stainless Steel 304 Chromium content	Negligible
Core elements Stainless Steel 304 Nickel content	Negligible
Core elements Stainless Steel 304 density (g/cm3)	538
Thermocouple presence	Negligible
Coating presence	300
Regulating Rod is aluminum rod	Negligible
Reactor height position	Negligible
Aluminum safety cage presence	113
Sample irradiation plug height	61
Sum in quadrature	**3803**

Estimated Experimental Uncertainties

Benchmark Model Overview



Operation Bren, CEX 62-02 (1965)



Overview of the detailed benchmark model made in SCALE, front right quarter

Oak Ridge National Laboratory

ANS

Sample Calculations Results

Reactor State	Configuration Number	k_{eff}				
		Expected	Uncertainty	Calculated	Uncertainty	Relative difference (%)
Critical	1	1.00000	0.03798	1.01385	0.00010	1.4
	2	1.00000	0.03798	1.01331	0.00010	1.3
	3	1.00000	0.03798	1.01029	0.00018	1.0
	4	1.00000	0.03798	1.00958	0.00017	0.9
	5	1.00000	0.03798	1.00951	0.00021	0.9
	6	1.00000	0.03798	1.00948	0.00018	0.9
	7	1.00000	0.03798	1.00988	0.00021	1.0
Sub-Critical	1	0.99966	0.03797	1.01288	0.00010	1.3
	2	0.99964	0.03797	1.01150	0.00010	1.2
	3	0.99981	0.03797	1.01229	0.00016	1.2
	4	0.99985	0.03797	1.01166	0.00019	1.2

Sample Calculation Results and Comparison to Expected Values

Sample Calculations Results

Reactor State	Configuration Number	k_{eff}				
		Expected	Uncertainty	Calculated	Uncertainty	Relative difference (%)
Critical	1	1.00000	0.03798	1.01385	0.00010	1.4
	2	1.00000	0.03798	1.01331	0.00010	1.3
	3	1.00000	0.03798	1.01029	0.00018	1.0
	4	1.00000	0.03798	1.00958	0.00017	0.9
	5	1.00000	0.03798	1.00951	0.00021	0.9
	6	1.00000	0.03798	1.00948	0.00018	0.9
	7	1.00000	0.03798	1.00988	0.00021	1.0
Sub-Critical	1	**0.99966**	0.03797	1.01288	0.00010	1.3
	2	**0.99964**	0.03797	1.01150	0.00010	1.2
	3	**0.99981**	0.03797	1.01229	0.00016	1.2
	4	**0.99985**	0.03797	1.01166	0.00019	1.2

Sample Calculation Results and Comparison to Expected Values

Sample Calculations Results

Reactor State	Configuration Number	k_{eff}				
		Expected	Uncertainty	Calculated	Uncertainty	Relative difference (%)
Critical	1	**1.00000**	0.03798	**1.01385**	0.00010	1.4
	2	**1.00000**	0.03798	**1.01331**	0.00010	1.3
	3	**1.00000**	0.03798	**1.01029**	0.00018	1.0
	4	**1.00000**	0.03798	**1.00958**	0.00017	0.9
	5	**1.00000**	0.03798	**1.00951**	0.00021	0.9
	6	**1.00000**	0.03798	**1.00948**	0.00018	0.9
	7	**1.00000**	0.03798	**1.00988**	0.00021	1.0
Sub-Critical	1	**0.99966**	0.03797	**1.01288**	0.00010	1.3
	2	**0.99964**	0.03797	**1.01150**	0.00010	1.2
	3	**0.99981**	0.03797	**1.01229**	0.00016	1.2
	4	**0.99985**	0.03797	**1.01166**	0.00019	1.2

Sample Calculation Results and Comparison to Expected Values

Sample Calculations Results

Reactor State	Configuration Number	k_{eff}				
		Expected	Uncertainty	Calculated	Uncertainty	Relative difference (%)
Critical	1	1.00000	0.03798	1.01385	0.00010	**1.4**
	2	1.00000	0.03798	1.01331	0.00010	**1.3**
	3	1.00000	0.03798	1.01029	0.00018	**1.0**
	4	1.00000	0.03798	1.00958	0.00017	**0.9**
	5	1.00000	0.03798	1.00951	0.00021	**0.9**
	6	1.00000	0.03798	1.00948	0.00018	**0.9**
	7	1.00000	0.03798	1.00988	0.00021	**1.0**
Sub-Critical	1	0.99966	0.03797	1.01288	0.00010	**1.3**
	2	0.99964	0.03797	1.01150	0.00010	**1.2**
	3	0.99981	0.03797	1.01229	0.00016	**1.2**
	4	0.99985	0.03797	1.01166	0.00019	**1.2**

Sample Calculation Results and Comparison to Expected Values

Conclusion

- A real information preservation and dissemination work, a lot of legacy content was found and used
- Abundance of uncertainty, discrepancy, contradictory information
- Yet, a detailed, functional SCALE model was built, and experimental data was evaluated for the creation of a critical benchmark
- The estimated experimental uncertainty is about 3800 pcm, very high
- The relative difference between expected and calculated keff values is about 1.5 %, also very high
- It is concluded that a good quality critical benchmark worthy of the ICSBEP standards cannot be created from HPRR data in the present conditions
- Locating the HPRR fuel to obtain an uncertainty on the density would solve the biggest issue

Last Words

- This work serves as a reminder for all of us to always record all information related to experimental work:
  - Dimensions
  - Material composition
  - Configuration of the room

- HPRR data is also currently being considered for the creation of a shielding benchmark
  - A first evaluation was submitted to the ICSBEP TRG in 2021
  - The evaluation is being updated and will be submitted again in 2022 for a 2023 publication in the handbook
  - References on the shielding evaluation:
    - M. N. Dupont, C. Celik, “Evaluation of Oak Ridge National Laboratory Health Physics Research Reactor Operation Data for Criticality Accident Alarm System Benchmark Creation,” Transactions of American Nuclear Society, 125, 1137-1140 (2021).
    - M. N. Dupont, E. M. Saylor, “Sulfur Pellets Responses to a Bare and Steel Reflected Pulse of the Oak Ridge National Laboratory Health Physics Research Reactor,” Oak Ridge National Laboratory, ORNL/TM-2020/1731 (2020).

This work was supported by the Nuclear Criticality Safety Program, funded and managed by the National Nuclear Security Administration for the Department of Energy.

Thank you for your attention

OAK RIDGE National Laboratory

ANS®