Draft

ADVISORY BOARD ON
RADIATION AND WORKER HEALTH

National Institute for Occupational Safety and Health

ARGONNE NATIONAL LABORATORY-WEST SEC-00224 REACTOR PRIORITIZATION FOR EVALUATION OF ORAUT-OTIB-0054 APPLICABILITY

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ABBREVIATIONS AND ACRONYMS

Advisory Board	Advisory Board on Radiation and Worker Health
AFSR	Argonne Fast Source Reactor
ALPR	Argonne Low Power Reactor
ANL-W	Argonne National Laboratory-West
ARA	Auxiliary Reactor Area
ARMF	Advanced Reactivity Measurement Facility Number
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical Facility
BORAX	Boiling Water Reactor Experiment
BWR	boiling water reactor
CET	Critical Experiment Tank
CFA	Central Facilities Area
CFRMF	Coupled Fast Reactivity Measurement Facility
CRCE	Cavity Reactor Critical Experiment
Cs	cesium
D	diameter
DOE	U.S. Department of Energy
EBOR	Experimental Beryllium Oxide Reactor
EBR	Experimental Breeder Reactor
EOCR	Experimental Organic Cooled Reactor
ETR	Engineering Test Reactor
ETRC	Engineering Test Reactor Critical Facility
F	Fahrenheit
FFTF	Fast Flux Test Facility
FRAN	Nuclear Effects Reactor
GCRE	Gas Cooled Reactor Experiment
GW_{th}	gigawatt thermal
H	height
HFEF	Hot Fuel Examination Facility
HOTCE	Hot Critical Experiment

HTRE	Heat Transfer Reactor Experiment
IETF	Initial Engine Test Facility
INL	Idaho National Laboratory
kW_e	kilowatt electric
kW_{th}	kilowatt thermal
LMFBR	Liquid Metal Fast Breeder Reactor
LOFT	Loss of Fluid Test Facility
LPTF	Low Power Test Facility
ML-1	Mobile Low-Power Reactor-1
MTHM	metric ton heavy metal
MTR	Materials Test Reactor
MTU	metric ton uranium
MW	megawatt
MWd	megawatt days
MW_e	megawatt electric
MW_{th}	megawatt thermal
Na	sodium
NaK	sodium-potassium (liquid metal)
NRAD	Neutron Radiography Facility
NRF	Naval Research Facility
NIOSH	National Institute for Occupational Safety and Health
OMRE	Organic Moderated Reactor Experiment
ORAU(T)	Oak Ridge Associated Universities (Team)
ORNL	Oak Ridge National Laboratory
OTIB	ORAUT Technical Information Bulletin
PBF	Power Burst Facility
psig	pounds per square inch gauge
Pu	plutonium
RMF	Reactivity Measurement Facility
SCRCE	Spherical Cavity Reactor Critical Experiment
SEC	Special Exposure Cohort

SEFOR	Southwest Experimental Fast Oxide Reactor
SL-1	Stationary Low-Power Reactor
SNAP	Systems for Nuclear Auxiliary Power
SNAPTRAN	Systems for Nuclear Auxiliary Power Transient
SPERT	Special Power Excursion Reactor Test
Sr	strontium
STPF	Shield Test Pool Facility
STR	Submarine Thermal Reactor
SUSIE	Shield Pool Test Facility Reactor
TAN	Test Area North
THRITS	Thermal Reactor Idaho Test Station
TRA	Test Reactor Area
TREAT	Transient Reactor Test Facility
TRIGA	Training, Research, Isotope General Atomics (reactor)
U	uranium
UO_2	uranium dioxide
WRRTF	Water Reactor Research Test Facility
ZPPR	Zero Power Physics Reactor
ZPR	Zero Power Reactor

1.0 INTRODUCTION

One of the topics presented at the July 23–24, 2014, meeting of the Advisory Board on Radiation and Worker Health (hereafter referred to as the "Advisory Board") in Tampa, Florida, was the Special Exposure Cohort (SEC) Petition SEC-00224 evaluation report for Argonne National Laboratory-West (ANL-W) (NIOSH 2016) prepared by the National Institute for Occupational Safety and Health (NIOSH) and its technical contractor, Oak Ridge Associated Universities (ORAU). At that meeting, the Advisory Board charged SC&A with reviewing the evaluation report.

Part of the review concerns the issues of dose reconstructability and gap analysis. With respect to the latter, inherent in the SEC framework is the assumption that doses can be reconstructed with sufficient accuracy for site areas and time periods that lie *outside* the SEC class definition and that are not being held in reserve for further evaluation by NIOSH. Operations at ANL-W involving radioactive materials were very complex, as many unique nuclear reactors and experiments were built and tested.

A primary tool that NIOSH uses for internal dose reconstruction is the guidance appearing in ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (hereafter referred to as "OTIB-0054") (ORAUT 2015). Except for certain situations, OTIB-0054 assigns fission and activation product intakes for different radioisotopes that are directly tied to an indicator radionuclide (strontium-90 [Sr-90] or cesium-137 [Cs-137]). OTIB-0054 generated nine different representative reactor cases, which are intended to envelope the range of reactor and nuclear fuel types and operating scenarios to which workers might have been exposed.

Before examining each ANL-W reactor in detail and determining whether its operations were bounded by the methodology in OTIB-0054, SC&A prioritizes in this report the reactors for further investigation, where one of the main factors is the degree to which the abundance of fission and activation products and actinides relative to the abundance of Cs-137 and Sr-90 bear any resemblance to the mix of radionuclides in OTIB-0054.[1]

Susan Stacy, in her comprehensive review of the history of the Idaho National Laboratory (INL) from inception through 1999, *Proving the Principle: A History of the Idaho National Engineering and Environmental Laboratory* (Stacy 2000), lists in Appendix B the 52 reactors that were built on the INL site (including two that never operated) and provides a brief summary of each. SC&A, following the practice of NIOSH in its INL/ANL-W reports (e.g., the site profile), uses Stacy's list as a convenient framework to examine the various reactors. Attachment 1 lists all the reactors and, for each covered by the ANL-W SEC petition, notes its

[1] A similar process was followed for the INL reactors, where *INL SEC-00219 Reactor Prioritization for Evaluation of ORAUT-OTIB-0054 Applicability* (SC&A 2016) prioritized them for further examination, *NIOSH SEC-00219: Test Reactor Area Modeling* (SC&A 2015a) examined the dose reconstructability assumption for the major reactors in the Test Reactor Area, and *Review of NIOSH Strategy for Reconstructing Internal Doses to Workers at Test Area* North (SC&A 2015b) examined dose reconstructability for some of the reactors in Test Area North.

operating period, provides a brief summary description, and presents SC&A's screening assessment of its priority ranking with respect to performing a detailed review on whether OTIB-0054 bounds its makeup and operating conditions.

The priority rankings are divided into three categories: High, Medium, and Low. Though based on a substantial amount of research, the rankings are still somewhat subjective because a full analysis would involve detailed and extensive research for each reactor and actually performing the OTIB-0054 applicability analyses themselves, which would go counter to the limited objectives of this screening process. The assignment of reactors to priority ranking categories considers reactor design factors such as the type of fuel (e.g., solid or gaseous, uranium or plutonium-based), enrichment (e.g., low-enriched commercial-type fuel or fully-enriched fuel), cladding (e.g., aluminum or steel), moderator (e.g., light water, heavy water, or beryllium), coolant (e.g., water or liquid metal), operational mode (e.g., steady-state or periodic), length of operation, and whether the reactor performed within design limits or was deliberately or inadvertently taken outside those limits (e.g., in tests supporting power reactor safety programs). Also considered qualitatively is the potential for significant radiation exposure of personnel. These screening criteria were selected because they were judged by SC&A to be those criteria that would be best indicative of the degree to which the default mix of radionuclides in OTIB-0054 might result in an underestimate of the internal doses to workers, or simply result in unrealistic estimates of the internal doses to workers at ANL-W who worked in the vicinity of these reactors or worked with irradiated fuel from these reactors.

In addition to OTIB-0054 applicability, this report also considers the following four factors, to the extent that they are known, that reflect the scope of the population potentially "at risk" of uncontrolled/unmonitored exposures.

- Duration reactor was in operation
- Frequency/intensity of operation
- Where possible, the approximate number of workers potentially exposed during its operation
- Incidents or other factors with potential to contribute to the risk of unintended/unprotected exposures

Of the 52 reactors in the entire INL site, 12 reactors operated at ANL-W; these are listed in Table 3.

2.0 ORAUT-OTIB-0054

After a series of initial runs using the ORIGEN2 isotope generation and depletion code (Croff 1980), OTIB-0054 selects four actual reactors to represent different general categories of reactors that might envelope the wide variety of reactors at the different sites considered in the Energy Employees Occupational Illness Compensation Program Act of 2000 program.[2] The representative reactors are listed in Table 1.

Table 1. ORAUT-OTIB-0054 Representative Reactors

Category	Reactor
High-flux reactors	Advanced Test Reactor (ATR)
Na-cooled fast reactors	Fast Flux Test Facility (FFTF)
Pu production reactors	Hanford N-Reactor
Research reactors	TRIGA with stainless steel cladding

Source: ORAUT 2015.

Multiple ORIGEN-S (ORNL 2015) runs performed by NIOSH produced a total of nine representative cases for the four reactors. ORIGEN-S is a more modern and capable version of ORIGEN and is part of the SCALE code system (ORNL 2015) for nuclear safety analysis and design, developed and maintained by the Oak Ridge National Laboratory (ORNL) for the U.S. Nuclear Regulatory Commission. Table 2 (OTIB-0054 Table 5-2) lists the parameters and basis selected by NIOSH for each of the cases.

[2] ORAUT 2015 should be consulted for the details of the selection process.

Table 2. ORIGEN-S Irradiation Parameters for the Nine Representative Reactor Cases

Case	Parameters	Basis
ATR 1	Specific power = 2,379.1 MW/MTU Irradiation time = 132.27 days Burnup = 314,684 MWd/MTU	Maximum burnup at nominal power.
ATR 2	Specific power = 8,651.2 MW/MTU Irradiation time = 36.4 days Burnup = 314,904 MWd/MTU	Maximum burnup at maximum assembly power.
ATR 3	Specific power = 2,379.1 MW/MTU Irradiation time = 56 days Burnup = 133,230 MWd/MTU	Nominal burnup at nominal power.
FFTF 1	Specific power = 163.8 MW/MTHM Irradiation time = 929.4 days	Maximum burnup at nominal power.
FFTF 2	Specific power = 163.8 MW/MTHM Irradiation time = 488.3 days	Nominal burnup at nominal power.
N Reactor 1	Specific power = 10.4 MW/MTU Irradiation time = 114.2 days Burnup = 1,188 MWd/MTU	Production of weapons-grade plutonium (nominal 6% Pu-240 content) at nominal power.
N Reactor 2	Specific power = 10.4 MW/MTU Irradiation time = 285.6 days Burnup = 2,970 MWd/MTU	Production of fuel-grade plutonium (nominal 12% Pu-240 content) at nominal power.
TRIGA 1	Specific power = 15.57 MWd/MTU Irradiation time = 730.1 days Burnup = 11,368 MWd/MTU	Maximum burnup at nominal power.
TRIGA 2	Specific power = 15.57 MW/MTU Irradiation time = 115.2 days Burnup = 1994 MWd/MTU	Nominal burnup at nominal power.

Source: Reproduced from ORAUT 2015, Table 5-2.

3.0 EVALUATION

NIOSH uses OTIB-0054 to determine internal doses to claimants using indicator radionuclides in cases where only gross beta or gross gamma measurements are available. The nine cases of Table 2 are intended to envelope reactor and nuclear fuel types and operating scenarios to which workers might have been exposed. As discussed in Section 1, the screening process in this report examines whether it seems likely that OTIB-0054 adequately envelopes the ANL-W reactors listed in Attachment 1 and prioritizes for further investigation those that might not be enveloped by the OTIB-0054 methodology. Table 3 lists all 52 reactors alphabetically (which is also numerically), as shown in Stacy (2000). The non-ANL-W reactors are included for convenience in identifying a particular reactor in the overall INL site. The rows for the non-ANL-W reactors are shaded. Priority rankings for the ANL-W reactors from Attachment 1 are also included in the last column.

Table 3. List of all INL Reactors

Reactor Name [a,b,c]	Location	Priority Ranking
1. Advanced Reactivity Measurement Facility No. 1 (ARMF-I)	TRA (Bldg. TRA-660)	—
2. Advanced Reactivity Measurement Facility No. 2 (ARMF-II). Renamed Coupled Fast Reactivity Measurement Facility (CFRMF) in 1968	TRA (Bldg. TRA-660)	—
3. Advanced Test Reactor (ATR)	TRA (Bldg. TRA-670)	—
4. Advanced Test Reactor Critical Facility (ATRC)	TRA (Bldg. TRA-670)	—
5. Argonne Fast Source Reactor (AFSR)	ANL-W	Low
6. Boiling Water Reactor Experiment No. 1 (BORAX-I)	ANL-W	High
7. Boiling Water Reactor Experiment No. 2 (BORAX-II)	ANL-W	High
8. Boiling Water Reactor Experiment No. 3 (BORAX-III)	ANL-W	High
9. Boiling Water Reactor Experiment No. 4 (BORAX-IV)	ANL-W	High
10. Boiling Water Reactor Experiment No. 5 (BORAX-V)	ANL-W	High
11. Cavity Reactor Critical Experiment (CRCE)	TAN - WRRTF – LPTF	—
12. Coupled Fast Reactivity Measurement Facility (CFRMF). Formerly named Advanced Reactivity Measurement Facility No. 2 (ARMF-II)	TRA (Bldg. TRA-660)	—
13. Critical Experiment Tank (CET)	TAN - WRRTF – LPTF	—
14. Engineering Test Reactor (ETR)	TRA (Bldg. TRA-642)	—
15. Engineering Test Reactor Critical Facility (ETRC)	TRA (Bldg. TRA-654)	—
16. Experimental Beryllium Oxide Reactor (EBOR)	TAN - WRRTF – LPTF (Bldg. TAN-646)	—
17. Experimental Breeder Reactor No. I (EBR-I)	ANL-W	High
18. Experimental Breeder Reactor No. II (EBR-II)	ANL-W	High
19. Experimental Organic Cooled Reactor (EOCR)	CFA (vicinity)	—
20. Fast Spectrum Refractory Metals Reactor (710)	TAN - WRRTF – LPTF	—
21. Gas Cooled Reactor Experiment (GCRE)	ARA-III	—
22. Heat Transfer Experiment No. 1 (HTRE-1)	TAN - IETF	—
23. Heat Transfer Experiment No. 2 (HTRE-2)	TAN - IETF	—
24. Heat Transfer Experiment No. 3 (HTRE-3)	TAN - IETF	—
25. High Temperature Marine Propulsion Reactor (630-A)	TAN - WRRTF – LPTF	—
26. Hot Critical Experiment (HOTCE)	TAN - WRRTF – LPTF	—

Reactor Name [a,b,c]	Location	Priority Ranking
27. Large Ship Reactor A (A1W-A)	NRF	—
28. Large Ship Reactor B A1W-B	NRF	—
29. Loss of Fluid Test Facility (LOFT)	TAN (Bldg. TAN-650)	—
30. Materials Test Reactor (MTR)	TRA (Bldg. TRA-603)	—
31. Mobile Low-Power Reactor No. 1 (ML-1)	ARA-IV	—
32. Natural Circulation Reactor (S5G)	NRF	—
33. Neutron Radiography Facility (NRAD)	ANL-W	Low
34. Nuclear Effects Reactor (FRAN)	ARA-IV	—
35. Organic Moderated Reactor Experiment (OMRE)	East of CFA, between Waste Area Groups 4 and 5	—
36. Power Burst Facility (PBF)	Near the SPERT-I site.	—
37. Reactivity Measurement Facility (RMF)	TRA (Bldg. TRA-603)	—
38. Shield Test Pool Facility (STPF - SUSIE)	TAN - WRRTF – LPTF (Bldg. TAN-646)	—
39. Special Power Excursion Reactor Test No. I (SPERT-I)	Separate complex east of CFA	—
40. Special Power Excursion Reactor Test No. II (SPERT-II)	Separate complex east of CFA	—
41. Special Power Excursion Reactor Test No. III (SPERT-III)	Separate complex east of CFA	—
42. Special Power Excursion Reactor Test No. IV (SPERT-IV)	Separate complex east of CFA	—
43. Spherical Cavity Reactor Critical Experiment (SCRCE)	TAN – WRRTF - LPTF	—
44. Stationary Low-Power Reactor (Earlier name: Argonne Low Power Reactor) (SL-1, ALPR)	ARA-II	—
45. Submarine Thermal Reactor (S1W, STR). Also known as the Submarine Prototype Reactor.	NRF	—
46. Systems for Nuclear Auxiliary Power (SNAP) 10A Transient No. 1 (SNAPTRAN–1)	TAN - IETF	—
47. Systems for Nuclear Auxiliary Power (SNAP) 10A Transient No. 3 (SNAPTRAN–3)	TAN - IETF	—
48. Systems for Nuclear Auxiliary Power (SNAP) 10A Transient No. 2 (SNAPTRAN–2)	TAN – IETF	—
49. Thermal Reactor Idaho Test Station (THRITS)	TAN – WRRTF - LPTF	—
50. Transient Reactor Test Facility (TREAT)	ANL-W	Medium
51. Zero Power Physics Reactor (Earlier name: Zero Power Plutonium Reactor) ZPPR	ANL-W	Low
52. Zero Power Reactor No. 3 (ZPR-III)	ANL-W	Low

Notes:
(a) The list of 52 reactors was taken from Stacy 2000.
(b) Location acronyms (current names are used in most cases): ANL-W = Argonne National Laboratory-West; ARA = Auxiliary Reactor Area; CFA = Central Facilities Area; IETF = Initial Engine Test Facility; LPTF = Low Power Test Facility; NRF = Naval Reactor Facility; TAN = Test Area North; TRA = Test Reactor Area; WRRTF = Water Reactor Research Test Facility.
(c) Shaded rows indicate non-ANL-W reactors at the INL site.

Attachment 1 categorizes the 12 ANL-W reactors according to three prioritization levels: High, Medium, or Low, as discussed in Section 1. It is apparent from looking at the summary descriptions of Attachment 1 that all of the ANL-W reactors were different from the four representative reactors of OTIB-0054 to a significant degree, since most of the former were one-of-a-kind experiments that might have utilized different fuel types (e.g., fissile materials, chemical forms, cladding, and physical arrangements), blankets (e.g., to breed more fuel), moderators, and coolants, and were often deliberately or inadvertently operated beyond design limits, sometimes to failure. In addition, rather than operate at more-or-less steady-state conditions for some length of time, some ANL-W reactors operated in pulsed mode, in which they produced a huge amount of power in a very short time interval before they shut themselves down (due to strongly negative reactivity properties, such as negative void, temperature, or expansion coefficients), or in intermittent mode, where a series of experiments were run, then the reactor shut down and possibly modified until the next series of experiments. Fuel burnups were frequently considerably lower than for the representative reactors of OTIB-0054, so that the long-lived decay products did not have the chance to build up in the fuel, resulting in different isotopic ratios than in the OTIB-0054 reactors. In addition, many of the individual ANL-W reactors experimented with different fuels, core configurations, coolants, moderators, etc. over their operating lifetimes.

Notwithstanding the above considerations, SC&A categorized the 12 ANL-W reactors as a guide for determining which reactors should be considered first in a more detailed study, such as was done in SC&A 2015a for some reactors in the Test Reactor Area (TRA) and in SC&A 2015b for some in Test Area North (TAN). The results of the categorizations of Attachment 1 are summarized in Table 4, which shows that seven reactors were put in the High category, one in the Medium category, and four in the Low category. High rankings were assigned in cases where it was thought that the mix of beta emitters from a particular reactor's fuel would be very different than the mix in the OTIB-0054 reactors, such that doses might be materially underestimated. This would not be evident, however, until the products of the relative amount of each radionuclide times its inhalation dose conversion factor, times its release fraction are calculated and added up. This index of risk would be compared to that associated with the mix in OTIB-0054. Only through such a process would it be determined if OTIB-0054 is bounding, but that determination would only be made in a detailed examination, which would be the next step beyond this report's screening assessment, which must rely on heuristic arguments.

The rationale behind the categorizations shown in Table 4 appears in Attachment 1. SC&A recommends the natural groupings of the five Boiling Water Reactor Experiment (BORAX) reactors together and the two Experimental Breeder Reactors (EBRs) together, all in the High group.

Table 4. Priority Class Categorization for ANL-W

Priority Class	Reactor Number from Attachment 1
High (7)	6, 7, 8, 9, 10, 17, 18
Medium (1)	50
Low (4)	5, 33, 51, 52
Total Reactors	12

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ATTACHMENT 1. ANL-W REACTOR PRIORITIZATION WITH RESPECT TO ORAUT-OTIB-0054 APPLICABILITY INVESTIGATION

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
5. Argonne Fast Source Reactor (AFSR) ANL-W/EBR-I Complex (1959–1970)/EBR-II Complex (1970–late 1970s)	1959–late 1970s	The AFSR, a small fast reactor facility, with a design power level of 1 kilowatt thermal (kW_{th}), fueled with high-enriched uranium, and with a blanket of solid depleted uranium with a minimum thickness of eight inches, supplied fast and thermal neutrons to: test, calibrate, and develop radiation detectors for the ANL-W fast reactor program; prepare radioactive metallic foils; check out experimental systems before placement in other reactors; and develop potential fast reactor experiments. The reactor had a 4.5" diameter (D) × 4.25" height (H) cylindrical core of solid high enrichment uranium and a blanket of solid depleted uranium (2,100 kg) over eight inches thick. The core was built up from uranium discs canned in nickel. Approximately 20 kg of U-235 were required to achieve criticality. It had a high density concrete shield and a 4 × 4 × 6 foot graphite thermal column. Controls and safety systems were located in a pit under the reactor. Experimental beam ports included a graphite thermal column and several beam holes. It was divided into an upper and lower section, with a ½" D hole at the centerline. It occupied two different locations: 1959–1970 – the EBR-I Complex in a standalone butler building, and 1970–1997 – the EBR-II Complex in a portion of the ZPPR support wing, tied into ZPPR ventilation system with negative pressure. 4 to8 people worked periodically at the facility in its first location, but significantly more were potentially exposed when it moved to the ZPPR facility. [The text in this cell continues on the next page.]	The AFSR was quite small and low-powered, so, even though it does not appear to be enveloped by the representative reactors of OTIB-0054 (it operated at a much lower power and with a much lower fuel burnup than the Fast Flux Test Facility), it doesn’t warrant a detailed examination at this time. <u>Priority Ranking</u>: Low

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
		References ANL 2016 Amundson 1969 Brunson 1959	
6. Boiling Water Reactor Experiment No. 1 (BORAX-I) ANL-W/EBR-I Complex	1953–1954	Overview: The BORAX series of reactor experiments, beginning with the operation of BORAX-I in 1953, tested the feasibility and safety and explored the operating parameters of direct steam production in a light water reactor. Would the formation of steam bubbles destabilize or stabilize the nuclear reaction? The chronology of the BORAX reactors is: BORAX-I: 7/1953–7/1954 (no surrounding structure) BORAX-II: 10/1954–3/1955 (new facility) BORAX-III: 7/1955–4/1956 (modification of B-II) BORAX-IV: 12/1956–6/1958 (modification of B-III) BORAX-V: 3/1962–8/1964 (new facility) BORAX-I: The 1.4 megawatt thermal (MW_{th}) reactor was fabricated at Argonne National Laboratory-East and shipped to ANL-W, where it was assembled and operated. The reactor was water moderated and relied on natural water cooling and steam formation to remove heat. It was housed in a 10-ft diameter and 20-ft deep tank open to the atmosphere, which resulted in the requirement of a ½ mile exclusion zone when operating and limited operations to the warmer months of the year. BORAX-I was a precursor to the SL-1 plant at INL. The core was built up from a lower grid and consisted of 26 curved plate fuel assemblies of a U-235-aluminum alloy clad with aluminum. Ten non-fuel plugs were also present to define the shape of the core. [The text in this cell continues on the next page.]	The BORAX series investigated the safety of boiling water reactors in five different reactors with a variety of configurations. The reactors were run in both steady-state and in transient modes, with the latter leading to core damage in some cases. Total burnup for each reactor was probably low, since each reactor operated for only a short period of time and then not continuously. It is not apparent whether any of the OTIB-0054 reactors could adequately envelope the BORAX conditions. It appears to make sense to consider the five reactors together in one study. Priority Ranking: High

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
	[Row continued from previous page; no additional text in this cell.]	Each assembly contained 18 fuel plates joined to aluminum side plates to form units approximately 3 inches square and 2 feet long (active section). Operators initially conducted a series of nondestructive experiments consisting of steady-state boiling, as well as over 70 excursion tests of <25 ms duration, until the reactor was deliberately destroyed in its final experiment on July 22, 1954, through the rapid withdrawal of its control rods. Calculations done prior to the test seriously underestimated the actual damage, where much of the core melted and blew pieces of fuel plates, etc. 200–300 ft away, in a large steam explosion and resulting "geyser." The BORAX-I testing determined that steam formation was an effective and rapid self-limiting feature against short-duration excursions due to the large negative coefficient of reactivity. Thus, it paved the way for the further development of the boiling water reactor (BWR) concept. <u>References</u> ANL 2016	

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
7. Boiling Water Reactor Experiment No. 2 (BORAX-II) ANL-W/EBR-I Complex	1954–1955	BORAX-II, at 6.4 MW_{th}, continued the boiling water reactor testing program beyond BORAX-I. It was contained in a stainless steel pressure vessel located below ground in a concrete-lined pit, and operated with varying enrichments of uranium in its MTR-type fuel plates. It was intentionally destroyed in 1955 by taking it prompt critical (critical from prompt fission neutrons alone without reliance on delayed neutrons, which are released over time from fission products). It was much larger than its predecessor and designed to be more representative of a future commercial BWR; e.g., it operated at a design pressure of 300 pounds per square inch gauge (psig). It released excess energy as steam, since it had no turbine generator attached. The reactor was water cooled (natural circulation), moderated, and reflected. <u>References</u> ANL 2016	The BORAX series investigated the safety of boiling water reactors in five different reactors with a variety of configurations. The reactors were run in both steady-state and in transient modes, with the latter leading to core damage in some cases. Total burnup for each reactor was probably low, since each reactor operated for only a short period of time and then not continuously. It is not apparent whether any of the OTIB-0054 reactors could adequately envelope the BORAX conditions. It appears to make sense to consider the five reactors together in one study. <u>Priority Ranking</u>: High
8. Boiling Water Reactor Experiment No. 3 (BORAX-III) ANL-W/EBR-I Complex	1955–1956	The BORAX-II reactor was modified (raising its output to 12 MW_{th}) and renamed BORAX-III after the addition of a 2.5 megawatt electric (MW_e) steam turbine-generator to investigate radioactive contamination of the turbine from radioactivity in the primary coolant and to demonstrate the production of electricity. The fuel consisted of plate-type fuel elements of a 90% enriched uranium-aluminum alloy with aluminum cladding. For two hours on July 17, 1955, the reactor supplied Arco, Idaho with 500 kilowatt electric (kW_e), the BORAX-III facility with 500 kW_e, and the Central Facilities Area with 1,000 kW_e, successfully demonstrating power production and distribution from a BWR. By the time it went out of service in 1956, BORAX-II had operated for a total of 1,170 hrs. <u>References</u> ANL 2016	The BORAX series investigated the safety of boiling water reactors in five different reactors with a variety of configurations. The reactors were run in both steady-state and in transient modes, with the latter leading to core damage in some cases. Total burnup for each reactor was probably low, since each reactor operated for only a short period of time and then not continuously. It is not apparent whether any of the OTIB-0054 reactors could adequately envelope the BORAX conditions. It appears to make sense to consider the five reactors together in one study. <u>Priority Ranking</u>: High

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
9. Boiling Water Reactor Experiment No. 4 (BORAX-IV) ANL-W/EBR-I Complex	1956–1958	BORAX-IV, a modification of BORAX-III, at 20 MW_{th} (and 2.5 MW_e) and 300 psig primary coolant pressure, tested uranium (U-233 and U-235) and thorium ceramic fuel plates (to allow higher temperature operations than with uranium fuel plates), some of which purposefully contained defects to determine reactor behavior with compromised fuel. The tests released some short-lived radionuclides to the atmosphere. Testing at full power with a large number of fuel elements having cladding defects, it released approximately 4,565 curies of short-lived radionuclides to the atmosphere in March 1958. References ANL 2016	The BORAX series investigated the safety of boiling water reactors in five different reactors with a variety of configurations. The reactors were run in both steady-state and in transient modes, with the latter leading to core damage in some cases. Total burnup for each reactor was probably low, since each reactor operated for only a short period of time and then not continuously. It is not apparent whether any of the OTIB-0054 reactors could adequately envelope the BORAX conditions. It appears to make sense to consider the five reactors together in one study. Priority Ranking: High

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
10. Boiling Water Reactor Experiment No. 5 (BORAX-V) ANL-W/EBR-I Complex	1962–1964	BORAX-V, a new facility, at 40 MW_{th}, added an integral nuclear superheat system to the BORAX experiments to improve thermal efficiency of the boiling water reactor plant and to explore the feasibility of superheating the steam. The latter raised the temperature of saturated steam from 489 °F in the core to 850 °F of dry steam and raised the pressure to 600 psig in the superheater. The reactor system had basically the same configuration as BORAX-IV. The reactor was placed in a cylindrical carbon steel pressure vessel with ellipsoidal heads that was clad with stainless steel in the interior. The reactor was water moderated and water cooled, with the superheater section cooled by steam. Cooling could be accomplished by natural or forced convection. The BORAX-V experiments also demonstrated that only negligible amounts of radioactivity travelled to the turbine generator section of the facility. <u>References</u> ANL 2016	The BORAX series investigated the safety of boiling water reactors in five different reactors with a variety of configurations. The reactors were run in both steady-state and in transient modes, with the latter leading to core damage in some cases. Total burnup for each reactor was probably low, since each reactor operated for only a short period of time and then not continuously. It is not apparent whether any of the OTIB-0054 reactors could adequately envelope the BORAX conditions. It appears to make sense to consider the five reactors together in one study. <u>Priority Ranking</u>: High

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
17. Experimental Breeder Reactor No. I (EBR-I) ANL-W/EBR-I Complex	1951–1963	The EBR-I fast breeder reactor demonstration was the first reactor built at INL and is a Registered National Historic Monument. It had a maximum power of 1 MW_{th} from its graphite-reflected, unmoderated, enriched uranium core. The reactor system was cooled by the liquid metal NaK (primary and secondary coolant systems) and surrounded by a U-238 breeding blanket. In 1951, it became the first reactor to generate electricity from an attached steam turbine-generator, which lit a few lightbulbs adjacent to the reactor. The EBR-I also demonstrated the feasibility of breeding plutonium in a reactor; i.e., creating more nuclear fuel than it consumed in order to supplement the scarce supplies of uranium then found in ore bodies (large deposits were subsequently found worldwide). In addition to producing the first electricity from a nuclear plant, EBR-I also demonstrated the feasibility of fuel production (breeding), whereby fissile Pu-239 was produced in the fertile U-238 blanket. In fact, at full power of 1 MW_{th}, 16% of the total power was generated in the blanket. Breeding ratios increased from 1.00 in the first core to 1.27 in the last core. The reactor also had several beam holes for instrumentation and irradiation experiments. The first three of the EBR-I's four core loadings had highly enriched (94% U-235) uranium fuel clad with either stainless steel or aluminum. On November 29, 1955, the reactor suffered a 40% to 50% inadvertent core meltdown due to operator error with control rods during an experiment. References INL 2016a McFarlane 2016	While OTIB-0054 models the FFTF, a fast reactor, and the Hanford N-Reactor, a plutonium production reactor, neither appears able to adequately envelope the EBR reactor conditions of much lower power and only intermittent operations. Priority Ranking: High

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
18. Experimental Breeder Reactor No. II (EBR-II) ANL-W/EBR-II Complex	1961–1994	The EBR-II, similar to, but a large scale-up from, EBR-I, continued fast neutron breeder reactor development at ANL-W, including onsite reprocessing of spent fuel into new fuel pins, demonstrating the feasibility of a closed fuel cycle. The unmoderated core, with 67% enriched U-235 fuel, sat in a tank of 90,000 gallons of liquid sodium (Na) primary coolant, had a closed-loop Na secondary coolant system, and produced steam in a tertiary system. The entire system was placed in a large containment building. The maximum power level was 62.5 MW_{th}, and the EBR-II could supply 20 MW_e of electric power to INL facilities. In addition to demonstrating fuel reprocessing and electricity production in a liquid metal fast breeder reactor (LMFBR), EBR-II also performed irradiation, fuel development, and transient stability experiments. References INL 2016a McFarlane 2016	While OTIB-0054 models the FFTF, a fast reactor, and the Hanford N-Reactor, a plutonium production reactor, neither appears able to adequately envelope the EBR reactor conditions of much lower power and only intermittent operations. Priority Ranking: High
33. Neutron Radiography Facility (NRAD) ANL-W/EBR-II Complex	1977–present	NRAD is a 250-kW_{th}, uranium-fueled, light water moderated and cooled pool-type, steady-state General Atomics TRIGA reactor, located beneath the main hot cell of the Hot Fuel Examination Facility (HFEF). It is used primarily for non-destructive neutron radiography of irradiated and unirradiated fuel and materials. It has limited experimental space inside the core, but its one beamline was supplemented by a second in 1982; both have a direct line of sight to the core. References Craft et al. 2015	OTIB-0054 explicitly models a TRIGA reactor, of which NRAD is a typical example. Hence, it would be expected that OTIB-0054 would adequately envelope the NRAD reactor. Priority Ranking: Low

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
50. Transient Reactor Test Facility (TREAT) ANL-W/EBR-II Complex	1959–1994	The TREAT experiment continued research and development for fast breeder reactors by simulating excursion and accident conditions leading to fuel damage. It was an air-cooled, graphite-moderated and reflected, uranium-fueled, thermal spectrum reactor, operating at a steady power level of up to about 120 kW_{th}. It could also produce pulses of up to 19 gigawatt thermal (GW_{th}) for experimental purposes, relying on a very large negative temperature coefficient of reactivity to then shut it down safely. The fuel consisted of a 93% enriched graphite-uranium dioxide (UO_2) matrix, with zirconium clad fuel assemblies. The core was approximately 6-feet diameter x 4-feet high, and contained a 19 x 19 array of 4 in x 4 in fuel and reflector assemblies. Several different sodium loops were introduced at different times into a portion of the core to perform fuel melt experiments from a simulated loss-of-coolant flow accident and transient overpower accidents. Over the years, the facility tested LMFBR-type oxide fuel elements in support of the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant safety analysis and licensing (the Clinch River plant was cancelled before it was built). Authorized personnel were permitted free access to the Reactor Building during shutdown, critical operation, or steady-state operation. In addition to its fast breeder reactor support mission, TREAT was also used for light water and space reactor fuel testing and nondestructive testing through neutron radiography (examining specimens up to 15 feet long). In its lifetime, the reactor performed 6,604 reactor startups and 2,884 transient irradiations. <u>References</u> Bumgardner 2014 INL 2016b	TREAT was an experimental reactor, testing different fuels in steady state and pulsed mode. While the OTIB-0054 TRIGA reactor also operates in steady state and pulsed mode, TREAT had a considerably different core and was graphite rather than water moderated. Further study would have to be done to determine if any of the reactors of OTIB-0054 would adequately model TREAT. <u>Priority Ranking</u>: Medium

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
51. Zero Power Physics Reactor (Earlier name: Zero Power Plutonium Reactor) (ZPPR) ANL-W/EBR-II Complex	1969–1992	ZPPR was a very low-power, air-cooled, split-table (one table movable and the other stationary) critical facility that was used to mock up other fast neutron spectrum reactors in support of the overall breeder reactor program. It was the largest in the series of split-table assemblies that were designed to support the plutonium-fueled liquid metal fast breeder reactor program by testing the neutronics of reactors of many different sizes, compositions, and shapes. The split-table design facilitated rapid and easy access to the center of core and ensured criticality safety when in the open position. Each half of the stainless-steel honeycomb structure was 14 ft × 14 ft × 5 ft. Reactor materials were stacked in drawers, which were then inserted into matrix tubes of the reactor structure. Fuel material consisted of high enriched uranium and/or plutonium, and the reactor was cooled by liquid sodium. The initial fuel loading had a total fissile mass (Pu-239, Pu-241, and U-235) of 363 kg (4/18/69). ZPPR Assembly 2 had 1,066 kg Pu-239 and Pu-241. References Lawroski 1972	The FFTF modeled in OTIB-0054 is a sodium-cooled fast reactor, and the Hanford N-Reactor bred plutonium. However, ZPPR's core and operating conditions were considerably different than either of the OTIB reactors. ZPPR's power level was quite low, reducing the priority of performing a more detailed investigation. Priority Ranking: Low

Reactor Name & Location [a]	Operation Dates	Summary Description [b]	Comments/Priority Ranking
52. Zero Power Reactor No. 3 (ZPR-III) ANL-W/EBR-I Complex	1955–1970	ZPR-III, which preceded ZPPR, was the smallest of the ZPPR series of split-table critical facilities. The split-table design facilitated rapid and easy access to the center of core and ensured criticality safety when in the open position. It established the feasibility of a sodium-cooled, unmoderated, fast breeder reactor. It was also used to determine the accuracy of predicted mass geometries and critical measurements for fast reactor core designs, including those for Fermi, Rapsodie, and the Southwest Experimental Fast Oxide Reactor (SEFOR). In all, there were a total of 63 different mockups ("assemblies"), with some existing for only short time periods and others for hundreds of experiments. Initially, ZPR-II used uranium fuel (coated with Teflon), but plutonium fuel (coated with stainless steel was introduced later, as was mixed oxide fuel. Fuels were either metallic, oxide, or carbide compositions. In addition, the facility switched from no cooling system to a simple air cooling system after a few years. References ANL 2016 Davey 1964	The FFTF modeled in OTIB-0054 is a sodium-cooled fast reactor, and the Hanford N-Reactor bred plutonium. However, ZPR-III's core and operating conditions were considerably different than either of the OTIB reactors. ZPR-III's power level was quite low, reducing the priority of performing a more detailed investigation. Priority Ranking: Low

Notes:

(a) The list and numbering scheme of the reactors are taken from Stacy 2000.

(b) The primary sources of information for the summary descriptions are: Stacy 2000, ORAUT 2010, and NIOSH 2016. Other significant sources specific to particular reactors are listed at the end of each reactor description.