Draft

ADVISORY BOARD ON
RADIATION AND WORKER HEALTH

National Institute for Occupational Safety and Health

INL SEC-00219 REACTOR PRIORITIZATION FOR EVALUATION OF ORAUT-OTIB-0054 APPLICABILITY

Contract No. 211-2014-58081
SCA-TR-2016-SEC002, Revision 0

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March 2016

SC&A, INC.: ***Technical Support for the Advisory Board on Radiation & Worker Health Review of NIOSH Dose Reconstruction Program***

DOCUMENT TITLE:	INL SEC-00219 Reactor Prioritization for Evaluation of ORAUT-OTIB-0054 Applicability
DOCUMENT NUMBER/ DESCRIPTION:	SCA-TR-2016-SEC002
REVISION NO.:	0 (Draft)
SUPERSEDES:	N/A
EFFECTIVE DATE:	March 2, 2016
TASK MANAGER:	John Stiver, MS, CHP
PROJECT MANAGER:	John Stiver, MS, CHP
DOCUMENT REVIEWER(S):	John Stiver

Record of Revisions

Revision Number	Effective Date	Description of Revision
0 (Draft)	03/02/2016	Initial issue

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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
ANP	Aircraft Nuclear Propulsion
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
CET	Critical Experiment Tank
CFA	Central Facilities Area
CFRMF	Coupled Fast Reactivity Measurement Facility
CRCE	Cavity Reactor Critical Experiment
Cs	cesium
D_2O	deuterium oxide
DOE	(U.S.) Department of Energy
EBOR	Experimental Beryllium Oxide Reactor
EBR	Experimental Breeder Reactor
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EOCR	Experimental Organic Cooled Reactor
ESF	engineered safety feature
ETR	Engineering Test Reactor
ETRC	Engineering Test Reactor Critical Facility
F	Fahrenheit
FFTF	Fast Flux Test Facility
FRAN	Nuclear Effects Reactor
Ft	feet, foot
GCRE	Gas Cooled Reactor Experiment

GW	gigawatt
GW_{th}	gigawatt thermal
HOTCE	Hot Critical Experiment
HTGR	high-temperature gas-cooled reactor
HTRE	Heat Transfer Reactor Experiment
IET	Initial Engine Test
IETF	Initial Engine Test Facility
in	inch
INL	Idaho National Laboratory
kW_e	kilowatts electric
kW_{th}	kilowatts thermal
LOCA	loss-of-coolant accident
LOFT	Loss of Fluid Test Facility
LPTF	Low Power Test Facility
LWR	light water reactor
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
MW-hr	megawatt-hour
N/A	not applicable
Na	sodium
NaK	sodium-potassium (liquid metal)
NASA	National Aeronautics and Space Administration
NRAD	Neutron Radiography Facility
NRC	(U.S.) Nuclear Regulatory Commission
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
Pu	plutonium
RMF	Reactivity Measurement Facility
RML	Radiation Measurement Laboratory
SCRCE	Spherical Cavity Reactor Critical Experiment
SEC	Special Exposure Cohort
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
UF_6	uranium hexafluoride
UO_2	uranium dioxide
USC	United States Code
W	watt
WRRTF	Water Reactor Research Test Facility
ZPPR	Zero Power Physics Reactor

1.0 INTRODUCTION

Following a series of meetings and discussions among the Advisory Board on Radiation and Worker Health (hereafter referred to as the "Advisory Board"), SC&A, and the National Institute for Occupational Safety and Health (NIOSH) and its technical contractor, Oak Ridge Associated Universities (ORAU), about NIOSH's March 12, 2015, release of the Special Exposure Cohort (SEC) Petition SEC-00219 Evaluation Report for the Idaho National Laboratory (INL) (NIOSH 2015a),[1] the Advisory Board requested that SC&A initially review two issues as part of a graded approach to assess the complex issues at this complex site: (1) class definition and (2) 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 INL involving radioactive materials were very complex, as many unique nuclear reactors and experiments were built and tested, irradiated nuclear fuel handled and processed, and radioactive waste disposed of. An SC&A report (SC&A 2015a) examined one aspect of the dose reconstructability assumption for one of the several major site areas, the Test Reactor Area (TRA). Another SC&A report (SC&A 2015b) examined dose reconstructability for some of the reactors in another major site area, Test Area North (TAN).

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 reactor and nuclear fuel types and operating scenarios to which workers might have been exposed. SC&A (2015a) evaluated whether OTIB-0054 is applicable to the three large materials-testing reactors located in the TRA: the Materials Test Reactor (MTR), the Engineering Test Reactor (ETR), and the Advanced Test Reactor (ATR). SC&A (2015b) similarly evaluated whether OTIB-0054 is applicable to the three Heat Transfer Reactor Experiment (HTRE) reactors located in TAN that supported the Aircraft Nuclear Propulsion (ANP) program. Subsequently, at the November 10, 2015, INL Work Group meeting, the Advisory Board members directed SC&A to screen reactors other than the six already addressed and create a prioritized list of reactors for detailed examination at a later date with respect to OTIB-0054 applicability; this report is the result of SC&A's screening.

Susan Stacy, in her comprehensive review of the history of INL from inception through 1999, *Proving the Principle* (Stacy 2000), lists in Appendix B the 52 reactors that were built at INL (including two that never operated) and provides a brief summary of each. SC&A, following the practice of NIOSH in its INL reports (e.g., the site profile), uses Stacy's list as a convenient

[1] Revision 1 of the ER was released on July 21, 2015 (NIOSH 2015b).

framework to examine the various reactors. Attachment 1 lists all the reactors and, for each, notes its 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.[2] Though based on a substantial amount of research, the rankings are still somewhat subjective because a full analysis would involve actually performing the detailed OTIB-0054 applicability analyses themselves. The assignment of reactors to priority ranking categories considers reactor design factors, such as the types of fuel, enrichment, cladding, moderator, and coolant; and operational modes, such as steady-state or periodic operations, as common for experimental reactors; and whether the reactor performed within design limits or was deliberately or inadvertently taken outside those limits, such as in tests supporting power reactor safety programs.

It should be noted that the initial list of 52 reactors was quickly reduced by subtracting the three TRA and three TAN reactors already evaluated in SC&A 2015a and SC&A 2015b, respectively, the four reactors at the Naval Reactor Facility (NRF) because that area is outside the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA) program, the 12 reactors at Argonne National Laboratory-West (ANL-W) because that area is not included in the INL SEC-00219, and the two reactors that were canceled before operations. Subtracting these 24 reactors leaves 28 candidate reactors for further study.

[2] There is also a category for those reactors that are not considered in the prioritization process.

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 EEOICPA program.[3] 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 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 ORNL for the U.S. Nuclear Regulatory Commission (NRC). Table 2 (OTIB Table 5-2) lists the parameters and basis selected for each of the cases.

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 Burnup = 152,230 MWd/MTHM	Maximum burnup at nominal power.

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

Case	Parameters	Basis
FFTF 2	Specific power = 163.8 MW/MTHM Irradiation time = 488.3 days Burnup = 79,984 MWd/MTHM	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. The evaluation in this report examines whether OTIB-0054 adequately envelopes the INL reactors listed in Attachment 1. The first step in the process was to identify which of the 52 listed reactors should be excluded from the evaluation: i.e., reactors that have already been examined elsewhere, that were in NRF or ANL-W, or that never operated. Table 3 shows the results of that winnowing process.

Table 3. INL Reactors Excluded From the Prioritization Process

Reactor Category	Reactor Number from Attachment 1
ANL-W (12)	5, 6, 7, 8, 9, 10, 17, 18, 33, 50, 51, 52
NRF (4)	27, 28, 32, 45
Already Evaluated (6)	3, 14, 22, 23, 24, 30
Never Operated (2)	16, 19
Total Excluded Reactors	24

Attachment 1 categorizes the remaining 28 reactors according to three prioritization levels: High, Medium, or Low, depending on a number of factors, such as the fuel properties, power level, and operating characteristics and history. The 24 reactors that are excluded from the prioritization process are included in the last group in Attachment 1 for completeness. It is apparent from looking at the summary descriptions of Attachment 1 that most of the INL reactors were different from the four representative reactors of OTIB-0054, since most of the former were one-of-a-kind experiments that might have utilized different fuels, 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 INL reactors operated in pulsed mode, in which they produced a huge amount of power, but in a very short time interval, before they shut themselves down (due to strongly negative reactivity properties, such as negative void or temperature coefficients). 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.

Notwithstanding the above considerations, SC&A categorized the 28 reactors as a guide to which should be considered first in a more detailed study, such as was done in SC&A 2015a for some reactors in the TRA and in SC&A 2015b for some in TAN. The results of the categorizations of Attachment 1 are summarized in Table 4, which shows that 13 reactors were put in the High category, 8 in the Medium category, and 7 in the Low category.

Table 4. Priority Class Categorization

Priority Class	Reactor Number from Attachment 1
High (13)	11, 29, 31, 35,36, 39, 40, 41, 42, 43, 46, 47, 48
Medium (8)	2, 12, 13, 20, 26, 34,3 8, 49
Low (7)	1, 4, 15, 21, 25, 37, 44
Total Included Reactors	28

4.0 REFERENCES

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

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
HIGH PRIORITY RANKING			
11. Cavity Reactor Critical Experiment (CRCE) TAN - WRRTF – LPTF	1967–early 1970s	CRCE was located in a shielded test cell of the Low Power Test Facility (LPTF) in the Water Reactor Research Test Facility (WRRTF) area of Test Area North (TAN). It was a National Aeronautics and Space Administration (NASA)-sponsored experiment to investigate using nuclear power for space rocket propulsion, where a gaseous core would be suspended in a spherical tank by a fast-moving hydrogen propellant. The hydrogen could be heated to a very high temperature, producing a very high specific impulse that might be used, for example, on a voyage to Mars. The core in the CRCE was uranium hexafluoride (UF_6), operating at the relatively low temperature of about 200° F.	The CRCE, with a gaseous core of UF_6, lies totally outside the envelope of any of the OTIB-0054 representative reactors. Priority Ranking: High
29. Loss of Fluid Test Facility (LOFT) TAN (Bldg. TAN-650)	1973–1985	The LOFT series of 38 nuclear power experiments, sponsored by the U.S. Nuclear Regulatory Commission (NRC), made major contributions to the light water reactor (LWR) safety program for commercial nuclear power plants by simulating system behaviors during a loss-of-coolant accident (LOCA). The reactor, which had a maximum power of 50 MW_{th}, and the associated components and systems, were built as a scale model of a commercial, four-coolant-loop pressurized water reactor and its engineered safety features (ESFs).	The LOFT facilities modeled a commercial LWR but often operated beyond usual limits to explore coupled nuclear and thermal hydraulic behavior in design-basis or beyond-design-basis accident scenarios. Priority Ranking: High

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
		The LOFT facility included five major systems: the reactor system, primary coolant system, blowdown suppression system, emergency core cooling system, and a secondary coolant system. The reactor fuel had the same characteristics as commercial fuel. The tests investigated whether the ESFs activated in a LOCA, sometimes coupled with loss of offsite power, could prevent or mitigate core damage and release of radioactive material. The LOFT reactor was located within a containment building to minimize radioactive releases to the environment. The experiments simulated different LOCAs due to small, medium, or large pipe breaks, including the actual Three Mile Island meltdown scenario of 1979. The resulting nuclear and thermal hydraulics data gave insight into system behavior during design and severe accidents, which could be used both to improve reactor system designs and to improve accident systems modeling codes. The last test, in 1985, which involved the deliberate melting of 100 fuel rods in an experimental fuel bundle, provided data on system performance in the event of a severe accident and measured radioactive releases from the damaged fuel. The test also resulted in the release of radioactivity into the containment building.	

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
31. Mobile Low-Power Reactor No. 1 (ML-1) Auxiliary Reactor Area IV (ARA-IV)	1961–1964	The Gas Cooled Reactor Experiment (GCRE) served as the prototype for the U.S. Army's high-temperature, gas-cooled reactor (HTGR), water moderated ML-1, which was located in the ARA-IV area. The ML-1 was truly mobile in that it could be packed up after a 36-hour shutdown, its three skids put on trailers, and moved to a new location. The plant produced 3.4 MW_{th} and, coupled to a compact power generation section, 330 kW_e. The ML-1, built by Aerojet General, went critical on March 31, 1961, and operated until 1964; the Army subsequently ended the development program in 1965. The fuel was 93% enriched contained in 61 fuel elements with 19 pins per element, in an approximately 22" D x 22" H cylindrical core, moderated by light water and cooled by nitrogen.	None of the representative reactor cases of OTIB-0054 are HTGRs. Priority Ranking: High
35. Organic Moderated Reactor Experiment (OMRE) Separate area a few miles east of Central Facilities Area (CFA), between Waste Area Groups 4 and 5.	1957 – 1963	The OMRE reactor, built by Atomics International, was part of an Atomic Energy Commission program to assess the feasibility and determine the nuclear and engineering technical basis of different reactor concepts in support of an emerging civilian nuclear power industry. OMRE used a waxy liquid hydrocarbon (known as "Sanowax") rather than water or a liquid metal as both coolant and moderator. The relatively low-power (5–10 MW_{th}), critical reactor tested various types and configurations of highly enriched uranium dioxide (UO_2) fuel elements.	It is doubtful whether OMRE would be adequately enveloped by any of the OTIB-0054 cases because it used an organic coolant and moderator. Priority Ranking: High

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
36. Power Burst Facility (PBF) Near the SPERT-I site.	1972–1985	The PBF continued the reactor safety program begun with the Special Power Excursion Reactor Test (SPERT) series of facilities but was much larger than the SPERT reactors. Fuel and cladding combinations were varied and tested to failure. Transient testing, including LOCA scenarios that modeled design-basis and severe accident conditions at a nuclear power plant, led to fuel and cladding damage and the subsequent evolution of hydrogen and the release of fission products to the reactor containment. As its name implies, the PBF could produce very high, short-duration (millisecond) power excursions that were self-limiting. It could operate at a steady-state power of 20 MW_e for a short period of time before initiating a very short super-critical power burst. The reactor was water-cooled and uranium-oxide-fueled, with stainless steel cladding. The core consisted of a square array of 121 square cells, 5.85 in × 5.85 in × 60 in, with an active fuel region of about 36 in located in a pressure vessel. Different fuel configurations, compositions, and enrichments were tested. The reactor was water cooled, moderated, and reflected. Some of the experiments used previously irradiated fuel (e.g., 38,000 MWd/MTU) to better simulate accident conditions.	The PBF used a variety of new and previously irradiated fuel and often operated to fuel failure, thus putting it outside the range of the nine representative cases of OTIB-0054. Priority Ranking: High

39. Special Power Excursion Reactor Test No. I (SPERT-I) Separate complex east of CFA	1955–1964	The SPERT series of reactors, of which there were four, were deliberately subjected to large, rapid reactivity excursions in order to gather data on coupled neutronic and thermal-hydraulic responses as part of an Atomic Energy Commission safety assessment program in support of commercial pressurized and boiling water nuclear power plants. The many SPERT experiments, which varied fuel design, core configurations, reflectors, moderators, coolant flow, temperatures, and pressures, supplied data for development and validation of computer codes to simulate reactor dynamics and for establishing safe operating limits. The SPERT series started out with thin, aluminum or stainless steel clad, uranium fuel plates but later transitioned to fuel rods, which were more typical of power reactors. SPERT-I, the first reactor of the series, was an open-tank, light water moderated and reflected reactor, with the uranium fuel enriched to 93.5%. Some experiments were also conducted with fuel enriched only a few percent to better simulate power reactor fuel. The fuel considered of plate type uranium and aluminum fuel assemblies (about 25" long) in a 4-foot diameter and 14-foot deep carbon steel tank, lined with aluminum cladding. Fuel burnup was quite low because the reactor operated in the transient rather than the steady-state mode. While SPERT-I experiments operated outside established design limits, conditions were usually kept below those producing core damage. However, a planned 2,300 MW_{th} excursion on November 5, 1962, destroyed the core and distorted the reactor vessel. Subsequently, SPERT-I was rebuilt, and low-enriched fuel rods replaced the previously used high-enriched fuel plates. A planned 17,400 MW_{th}	The SPERT series modeled a commercial LWR but often operated beyond usual limits to explore coupled nuclear and thermal-hydraulic behavior in design-basis or beyond-design-basis accident scenarios. It also varied fuel and reactor component materials and designs. Priority Ranking: High

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
		excursion on November 12, 1963 (with 4% uranium oxide fuel rods), and a planned 35,000 MW_{th} excursion on April 14, 1964 (with 4% uranium oxide fuel rods), tested the resilience of the fuel rods; the latter test damaged some of the fuel rods.	
40. Special Power Excursion Reactor Test No. II (SPERT-II) Separate complex east of CFA	1960–1964	SPERT-II construction followed SPERT-III and continued to investigate transient behavior. Several different types of fuel assemblies were used, both light and heavy were tested as moderators and coolants, and different reflectors were also used. Unlike SPERT-I, SPERT-II was placed in a closed pressure vessel. The active length of the flat-plate fuel assemblies were about 24 inches. Each fuel plate consisted of an enriched uranium-aluminum alloy. The fuel was enriched to 93.5% in uranium-235 (U-235). Since the reactor operated in the transient, burst mode, total burnup was small.	The SPERT series modeled a commercial LWR but often operated beyond usual limits to explore coupled nuclear and thermal-hydraulic behavior in design-basis or beyond-design-basis accident scenarios. It also varied fuel and reactor component materials and designs. Priority Ranking: High

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
41. Special Power Excursion Reactor Test No. III (SPERT-III) Separate complex east of CFA	1958–1968	SPERT-III accommodated the widest variation in several important parameters: temperature, pressure, and coolant flow. The core sat in a pressure vessel similar to that used in a commercial nuclear power plant, and the maximum operating temperature of 668 $^{\circ}$F and pressure of 2,500 psig also simulated power plant conditions. The system could produce a maximum of 60 MW_{th} for about 30 min of operating time, limited by the capacity of the heat removal system. The fuel plates contained 4.8% enriched UO_2 clad in stainless steel; the overall core dimensions were about 2 ft diameter × 3 ft height. The reactor used ordinary water as coolant and moderator.	The SPERT series modeled a commercial LWR but often operated beyond usual limits to explore coupled nuclear and thermal-hydraulic behavior in design-basis or beyond-design-basis accident scenarios. It also varied fuel and reactor component materials and designs. Priority Ranking: High
42. Special Power Excursion Reactor Test No. IV (SPERT-IV) Separate complex east of CFA	1962–1970	SPERT-IV also investigated transient reactor behavior to provide neutronic and thermal-hydraulic data applicable especially to large, open pool reactors. The reactor was located in an open tank, and the fuel consisted of a 93.5% enriched uranium-aluminum matrix. The facility utilized a number of different cores and other components and was operated over a wide range of several different parameters. Test scenarios included fuel destruction experiments.	The SPERT series modeled a commercial LWR but often operated beyond usual limits to explore coupled nuclear and thermal-hydraulic behavior in design-basis or beyond-design-basis accident scenarios. It also varied fuel and reactor component materials and designs. Priority Ranking: High

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
43. Spherical Cavity Reactor Critical Experiment (SCRCE) TAN – WRRTF – LPTF	1972–1973	Located in a shielded test cell of the LPTF in the WRRTF of TAN, SCRCE was a NASA-sponsored program to determine the feasibility and explore the neutronics characteristics of a reactor going critical with a core of highly-enriched gaseous uranium (UF_6) and a deuterium oxide (D_2O) reflector and moderator. Heat from the reactor would be used to heat a gas that would propel a space vehicle. The low-power (<100 W) experiment was located in a shielded cell of the LPTF. Several different configurations were measured in the test series.	SCRCE, with a gaseous core of UF_6 and a D_2O reflector, lies totally outside the envelope of any of the OTIB-0054 representative reactors. Priority Ranking: High
46. Systems for Nuclear Auxiliary Power (SNAP) 10A Transient No. 1 (SNAPTRAN–1) TAN – IETF	Early 1960s	The Atomic Energy Commission's Systems for Nuclear Auxiliary Power Transient (SNAPTRAN) program, located at the Initial Engine Test Facility (IETF) of TAN, extended the SPERT reactor safety testing program to aerospace applications by testing the SNAP 10A/2 reactor under extreme conditions. The SNAP 10A/2 reactor was a compact cylinder, with a core about 9 inches in diameter and 13 inches high, with 37 1.212-inch diameter stainless steel-clad fuel rods. The reactors were fully enriched, zirconium hydride-uranium fueled, sodium-potassium (NaK) cooled, and beryllium reflected. SNAPTRAN-1 was subjected to nondestructive, large-transient tests in conditions approaching but not resulting in damage to the fuel. The series of tests investigated the effect on the fuel of large power transients.	The SNAPTRAN series often operated beyond usual limits to explore coupled nuclear and thermal-hydraulic behavior in design-basis or beyond-design-basis accident scenarios. It also varied fuel and reactor component materials and designs. Priority Ranking: High

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
47. Systems for Nuclear Auxiliary Power (SNAP) 10A Transient No. 3 (SNAPTRAN–3) TAN – IETF	1964	SNAPTRAN-3 followed SNAPTRAN-2 and included a destructive test on April 1, 1964, to simulate the accidental fall of a reactor into water or wet earth following an accident on a nuclear-powered aircraft. The test demonstrated that the reactor would destroy itself immediately instead of building up a high inventory of radioactive fission products. The SNAP 10A/2 reactor operated at 30 GW for 1.5 milliseconds.	The SNAPTRAN series often operated beyond usual limits to explore coupled nuclear and thermal-hydraulic behavior in design-basis or beyond-design-basis accident scenarios. It also varied fuel and reactor component materials and designs. Priority Ranking: High
48. Systems for Nuclear Auxiliary Power (SNAP) 10A Transient No. 2 (SNAPTRAN–2) TAN – IETF	1965–1966	SNTAPTRAN-2 followed SNAPTRAN-3. The SNAP 10A/2 reactor was intentionally destroyed on January 11, 1966. It provided information on the dynamic response, fuel behavior, and inherent shutdown mechanisms of these reactors in an open air environment. The reactor produced a peak power of 74 GW for 1.5 milliseconds before it destroyed itself.	The SNAPTRAN series often operated beyond usual limits to explore coupled nuclear and thermal-hydraulic behavior in design-basis or beyond-design-basis accident scenarios. It also varied fuel and reactor component materials and designs. Priority Ranking: High

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
MEDIUM PRIORITY RANKING			
2. Advanced Reactivity Measurement Facility No. 2 (ARMF-II). Renamed Coupled Fast Reactivity Measurement Facility (CFRMF) in 1968 TRA (Bldg. TRA-660)	1962–1991	The ARMF-II occupied the opposite end of the tank containing the ARMF-I and was similar to that reactor but had some additional features to improve its accuracy. In addition to measuring the nuclear properties of reactor fuels and materials, the reactor was also used for neutron radiography. The ARMF-II was modified in 1968 to enable it to measure the nuclear characteristics of fast reactor fuels and materials; it was then renamed the Coupled Fast Reactivity Measurement Facility (CFRMF).	The reactor used MTR/ETR/ATR-type fuel but operated only periodically and at a low power level, resulting in low burnup of the fuel. Hence, it might not be enveloped by any of the three ATR cases of OTIB-0054, but modeling it with one of the ATR cases might be claimant favorable, since the ATR cases would have a greater buildup of longer-lived radionuclides. However, when operating with fast reactor fuels (as the CFRMF) it might not be adequately be enveloped by either an ATR case or a Fast Reactor Test Facility (FFTF) (fast reactor) case. Priority Ranking: Medium
12. Coupled Fast Reactivity Measurement Facility (CFRMF). Formerly named Advanced Reactivity Measurement Facility No. 2 (ARMF-II) TRA (Bldg. TRA-660)	1968–1991	When the ARMF-II reactor was modified in 1968, it was given a new name, the CFRMF. A section of the core was modified to produce a region of high-energy neutron flux that was useful in comparing calculated and observed results. This tool provided physics information about the behavior of fast (i.e., unmoderated) neutrons. Physicists studied differential cross-sections and tested calculational methods. The CFRMF contributed to the development of fast neutron reactors.	The CFRMF operated only periodically and at a low power level, resulting in low burnup of the fuel. When running as a fast reactor, it might not be adequately enveloped by either an ATR case or a FFTF (fast reactor) case. Priority Ranking: Medium

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
13. Critical Experiment Tank (CET) TAN – WRRTF – LPTF	1958–1960	Three low-power reactors supported the Aircraft Nuclear Propulsion (ANP) program by testing various components and collecting nuclear physics data: the Critical Experiment Tank (CET), the Hot Critical Experiment (HOTCE), and the Shield Pool Test Facility (STPF) Reactor (SUSIE). They were located in the LPTF, which was part of the WRRTF of TAN. The water-moderated and reflected CET was a low-power reactor that was originally intended to simulate the Heat Transfer Reactor Experiment (HTRE)-1 and HTRE-2 reactors in the ANP program, and to perform critical experiments of HTRE fuel bundles. It was also used as a source of neutrons to calibrate neutron sensors.	It is not clear without a more in-depth investigation whether the CET would be adequately enveloped by any of the OTIB-0054 cases. Priority Ranking: Medium
20. Fast Spectrum Refractory Metals Reactor (710) TAN – WRRTF – LPTF	1962–1968	Located in a shielded test cell of the LPTF in the WRRTF of TAN, this low-power, split-table, critical facility collected data for a proposed fast-spectrum, compact, refractory-metal reactor concept for generating power in space.	It is not clear without a more in-depth study if any of the nine OTIB-0054 cases would envelope a split-table, fast reactor. The power level is low, though. Priority Ranking: Medium
26. Hot Critical Experiment (HOTCE) TAN – WRRTF – LPTF	1958–1961	Three low-power reactors supported the ANP program by testing various components and collecting nuclear physics data: the CET, the HOTCE, and the STPF Reactor (SUSIE). They were located in the LPTF, which was part of the WRRTF of TAN. HOTCE was a low-power, high-temperature critical reactor.	It is not clear without a more in-depth investigation whether the HOTCE would be adequately enveloped by any of the OTIB-0054 cases. Priority Ranking: Medium

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
34. Nuclear Effects Reactor (FRAN) ARA-IV	1968–1970	FRAN was a small, prompt-burst reactor that could go super critical for a short time, producing a copious amount of fast neutron and gamma radiation inside its annular void, where samples would be placed. Expansion of its fuel assembly would quickly lower the reactivity, thereby controlling the excursion. The bare cylindrical assembly was fueled with 93.5% enriched uranium and clad with nickel and cadmium. FRAN was used to test new detector systems and to provide heat transfer and nuclear physics information of materials subjected to intense fast-neutron bombardment. The reactor was moved to the Lawrence Livermore National Laboratory (current name) in 1970.	Note: FRAN appears in Stacy 2000 and other literature that was examined, but not in the INL site profile. Priority Ranking: Medium
38. Shield Test Pool Facility (STPF – SUSIE) TAN – WRRTF – LPTF (Bldg. TAN-646)	Early 1960s	Three low-power reactors supported the ANP program by testing various components and collecting nuclear physics data: the CET, the HOTCE, and the STPF Reactor (SUSIE). They were located in the LPTF, which was part of the WRRTF of TAN. The SUSIE reactor was used for bulk shielding experiments that were performed in support of the ANP Shielding Experimentation Program. The water-moderated, -reflected, and –cooled reactor, situated in a water-filled pool at TAN, could be operated safely, was adaptable to many forms of nuclear research, and was easy to operate at minimum cost. The reactor's nominal power level was 2 MW_{th} from aluminum-clad U-235 fuel. After the ANP program was discontinued in 1961, SUSIE continued in use by other programs.	It is not clear without a more in-depth investigation whether SUSIE would be adequately enveloped by any of the OTIB-0054 cases. Priority Ranking: Medium

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
49. Thermal Reactor Idaho Test Station (THRITS) TAN – WRRTF – LPTF	1964	THRITS, located in an LPTF cell, was a low-power reactor with a split-table core where the two halves were brought together to make a critical assembly. Operators mocked up reactor design concepts for thermal and fast neutron reactor systems to obtain basic physics and design data for such concepts. The THRITS fuel consisted of polyethylene interspersed with enriched uranium foils, producing a thermal neutron spectrum.	It is not clear without a more in-depth study if any of the nine OTIB-0054 cases would envelope a split-table reactor. The power level is low, though. Priority Ranking: Medium
LOW PRIORITY RANKING			
1. Advanced Reactivity Measurement Facility No. 1 (ARMF-I) TRA (Bldg. TRA-660)	1960–1974	The ARMF-I was a very small, water pool reactor, operating up to 100 kW_{th} on highly enriched uranium, located in a water tank near the MTR. It was used to determine nuclear properties, such as reactivity, of small samples placed in it. The reactor replaced the Reactivity Measurement Facility (RMF).	The reactor used MTR/ETR/ATR-type fuel but operated only periodically and at a low power level, resulting in low burnup of the fuel. Hence, it might not be enveloped by any of the three ATR cases of OTIB-0054, but modeling it with one of the ATR cases might be claimant favorable, since the ATR cases would have a greater buildup of longer-lived radionuclides. Priority Ranking: Low
4. Advanced Test Reactor Critical Facility (ATRC) TRA (Bldg. TRA-670)	1964–present	The low power, highly enriched ATRC performs functions for the ATR similar to those of the Engineering Test Reactor Critical Facility (ETRC) reactor for the MTR. The full-sized reactor tests fuel and experiment configurations destined for the ATR, but at low power levels. The ATRC is designed to operate at 5 kW_{th} steady state but is usually operated at about 500 W_{th}.	The reactor uses MTR/ETR/ATR-type fuel but operates only periodically and at a low power level, resulting in low burnup of the fuel. Hence, it might not be enveloped by any of the three ATR cases of OTIB-0054, but modeling it with one of the ATR cases might be claimant favorable, since the ATR cases would have a greater buildup of longer-lived radionuclides. Priority Ranking: Low

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
15. Engineering Test Reactor Critical Facility (ETRC) TRA (Bldg. TRA-654)	1957–1982	The ETRC was operated as a full-scale, low-power version of the ETR that was used to determine the nuclear characteristics of experiments that would later be irradiated in the ETR.	The reactor used MTR/ETR/ATR-type fuel but operated only periodically and at a low power level, resulting in low burnup of the fuel. Hence, it might not be enveloped by any of the three ATR cases of OTIB-0054, but modeling it with one of the ATR cases might be claimant favorable, since the ATR cases would have a greater buildup of longer-lived radionuclides. Priority Ranking: Low
21. Gas Cooled Reactor Experiment (GCRE) ARA-III	1960–1961	The GCRE, located at the ARA-III site, was the Army's initial effort at developing a low-power, nitrogen-cooled, water-moderated mobile nuclear power plant. The GCRE was built by Aerojet General Nucleonics, went critical on February 23, 1960, and operated until April 1961. The GCRE generated 2.2 MW_{th} from either plate-type or pin-type fuel; the former was housed in an aluminum calandria and the latter in a stainless steel calandria. The GCRE provided nuclear and engineering data, as well as served as a training platform for Army and civilian personnel.	Priority Ranking: Low
25. High Temperature Marine Propulsion Reactor (630-A) TAN – WRRTF – LPTF	1962–1964	Located in a shielded test cell of the LPTF in the WRRTF of TAN, the 630-A reactor, a low-power critical experiment, explored the feasibility of an air-cooled, water-moderated system for nuclear-powered merchant ships.	Priority Ranking: Low

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
37. Reactivity Measurement Facility (RMF) TRA (Bldg. TRA-603)	1954–1962	The RMF, a water pool reactor with a highly enriched water-cooled, -moderated, and -reflected uranium core (MTR-type fuel) and a power level of 100 or 200 W_{th}, was located in a canal in the basement of the MTR building. It was used to assay new and spent fuel assemblies for the MTR and ETR and to measure nuclear properties. It was replaced by the ARMF-I, which became operational in 1964.	The reactor used MTR/ETR/ATR-type fuel but operated only periodically and at a low power level, resulting in low burnup of the fuel. Hence, it might not be enveloped by any of the three ATR cases of OTIB-0054, but modeling it with one of the ATR cases might be claimant favorable, since the ATR cases would have a greater buildup of longer-lived radionuclides. Priority Ranking: Low
44. Stationary Low-Power Reactor (Earlier name: Argonne Low Power Reactor) (SL-1, ALPR) ARA-II	1958–1961	The SL-1 was a low-power boiling water reactor, with a design power of 3 MW_{th}. It was designed by Argonne National Laboratory, based on Boiling Water Reactor Experiment (BORAX) experience, as a prototype of a reactor that could be used by the Army in geographically remote locations where fossil fuel was difficult to obtain, such as the Arctic. The fuel, which was 93% enriched, was in the form of aluminum-uranium alloy plates, and light water served as coolant and moderator. A criticality accident occurred on January 3, 1961, due to the central control rod being manually withdrawn too far in an attempted startup after a maintenance period. The withdrawal of the control rod, which has a large reactivity worth, caused the power level to spike at about 20 GW in a few milliseconds. This resulted in a violent steam explosion and core meltdown that killed three workers in the building at the time and released fission products in the building and to the atmosphere, as well as substantial contamination to the area around the building; the reactor building	The ATR cases might adequately envelope the SL-1 (other than its meltdown). Priority Ranking: Low

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
		was fabricated out of steel plate and was not designed as a containment structure.	
EXCLUDED FROM PRIORITIZATION PROCESS (ANL-W, NRF, ALREADY EVALUATED, OR NEVER OPERATED)			
3. Advanced Test Reactor (ATR) TRA (Bldg. TRA-670)	1967–present	ORAUT-OTIB-0054 uses the ATR as a surrogate for high-flux reactors (see Table 1), as might be encountered in materials testing or experimental reactors. The ATR, which has a design power level of 250 MW_{th}, is the latest and largest of three materials testing reactors at INL (the other two are the MTR and the ETR, both of which no longer operate). The ATR is a pressurized, light-water moderated, beryllium-reflected reactor, using highly enriched uranium fuel (93.15% nominal enrichment) arranged in an unusual curved plate configuration, and employing a unique design of rotating beryllium cylinder shells as the primary reactivity control mechanism. The reactor's four-lobed design supports nine main test spaces, or loops.	Reviewed in SC&A report (SC&A 2015a): NIOSH SEC-00219: Test Reactor Area Modeling, SCA-SEC-2015-0074-C, Revision 0, September 28, 2015: "*ORAUT-OTIB-0054 explicitly models the ATR, so it is expected that any workers exposed to ATR fuel during reactor operations or when the fuel is out of the reactor, but before the isotopic constituents might have been separated, would be adequately treated by the methodology in the OTIB in order to determine internal exposures. SC&A in its investigations did not find instances of the ATR operating outside of its design envelope.*" Priority Ranking: N/A
5. Argonne Fast Source Reactor (AFSR) ANL-W	1959–late 70s	The AFSR, with a design power level of 1 kW_{th}, fueled with high-enriched uranium, and with a blanket of solid depleted uranium, supplied neutrons to test and develop equipment (e.g., radiation detectors) for the ANL-W fast reactor program.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
6. Boiling Water Reactor Experiment No. 1 (BORAX-I) ANL-W	1953–1954	The BORAX series of reactor experiments, beginning with the operation of BORAX-I in 1953 (1.4 MW_{th}), tested the feasibility and safety and explored the operating parameters of direct steam production in an LWR. The final, planned destructive test of BORAX-I in 1954 melted the core, destroyed the reactor, and released fuel and fission products.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A
7. Boiling Water Reactor Experiment No. 2 (BORAX-II) ANL-W	1954–1955	BORAX-II, at 6 MW_{th}, continued the boiling water reactor testing program. It operated with varying enrichments of uranium in its fuel plates. It was intentionally destroyed in 1955 by taking it prompt critical.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A
8. Boiling Water Reactor Experiment No. 3 (BORAX-III) ANL-W	1955–1956	BORAX-III, at 15 MW_{th}, was connected to a 2 MW_e steam turbine-generator to investigate radioactive contamination of the turbine and to demonstrate the production of electricity.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A
9. Boiling Water Reactor Experiment No. 4 (BORAX-IV) ANL-W	1956–1958	BORAX-IV, at 20 MW_{th}, tested uranium and thorium ceramic 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.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A
10. Boiling Water Reactor Experiment No. 5 (BORAX-V) ANL-W	1962–1964	BORAX-V, at 40 MW_{th}, added an integral, nuclear superheat system to the BORAX experiments to improve thermal efficiency of the boiling water reactor plant	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
14. Engineering Test Reactor (ETR) TRA (Bldg. TRA-642)	1957–1981	The ETR, which was larger and had higher flux than the MTR and was located next to it, was designed to take advantage of information gathered and lessons learned from its predecessor materials test reactor. It operated at a maximum power level of 175 MW_{th}. It eliminated the beam holes of the MTR, with all experiments taking place within the core.	Reviewed in SC&A report (SC&A 2015a): NIOSH SEC-00219: Test Reactor Area Modeling, SCA-SEC-2015-0074-C, Revision 0, September 28, 2015: "*the OTIB-0054 methodology, which explicitly modes the ATR, should also adequately envelope the ETR in considering external exposures.*" Priority Ranking: N/A
16. Experimental Beryllium Oxide Reactor (EBOR) TAN – WRRTF – LPTF (Bldg. TAN-646)	Never operated	Construction began in May 1963, but the project was canceled in 1966 before construction was complete.	Never operated Priority Ranking: N/A
17. Experimental Breeder Reactor No. I (EBR-I) ANL-W	1951–1963	The EBR-I fast breeder reactor demonstration was the first reactor built at INL. It had a maximum power of 1 MW_{th} from its enriched uranium core, cooled by the liquid metal NaK, 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. About half of the core inadvertently melted in 1955 during a coolant flow test.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
18. Experimental Breeder Reactor No. II (EBR-II) ANL-W	1961–1994	The EBR-II continued fast neutron breeder reactor development at ANL-W, including onsite reprocessing of spent fuel into new fuel pins. The core, with 67% enriched U-235 fuel, sat in a tank of liquid sodium (Na) coolant, had a closed-loop Na coolant system, and produced steam in a tertiary system. The reactor was unmoderated. The entire system was placed in a 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.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A
19. Experimental Organic Cooled Reactor (EOCR) CFA (vicinity)	Never operated	The EOCR was intended to test the organic-coolant concept beyond the Organic Moderated Reactor Experiment (OMRE) but was placed in standby in 1962 and never operated.	Never operated. Priority Ranking: N/A

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
22. Heat Transfer Experiment No. 1 (HTRE-1) TAN – IETF	1955–1959	The three HTRE reactors (designed by General Electric) and the associated 26 Initial Engine Tests (IETs), some of which were non-nuclear, were an important part of the Air Force's ANP program. They explored the characteristics of direct-cycle heat transfer engineering applied to a turbojet engine intended for an aircraft. Fuel utilized highly enriched uranium in the form of very thin concentric ribbons to maximize heat transfer from the compact reactors. The reactors were located in the IETF of TAN and, due to the nature of the HTRE design, released substantial amounts of radionuclides to the environment through a 150-ft exhaust stack. HTRE-1 used enriched metallic nickel-chromium-oxide fuel elements and was water moderated and cooled. The reactor produced 20 MW_{th} and was operated for 150.8 hours at full power.	SC&A (2015b) evaluated the three HTRE reactors and concluded that OTIB-0054 does not appear to adequately envelope them. <u>Priority Ranking</u>: N/A
23. Heat Transfer Experiment No. 2 (HTRE-2) TAN – IETF	1957–1961	The HTRE-2 core was similar to the HTRE-1 core, but with provisions for operating as a materials testing reactor for experimental fuel sections introduced into a hexagonal center hole. The reactor operated for 1,299 hours at power levels up to 14 MW_{th} and temperatures of 2,800 ˚F for extended periods. Total burnup amounted to about 5,000 MW-hr.	SC&A (2015b) evaluated the three HTRE reactors and concluded that OTIB-0054 does not appear to adequately envelope them. <u>Priority Ranking</u>: N/A

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
24. Heat Transfer Experiment No. 3 (HTRE-3) TAN – IETF	1958–1960	The HTRE-3 was built in a full-scale aircraft reactor configuration, where the reactor and other components were arranged horizontally to simulate their locations in an actual airplane. The reactor was water cooled and water moderated, with uranium fuel and nickel-chromium cladding. The hydrided zirconium moderated was air cooled. The reactor underwent an unplanned nuclear excursion on November 18, 1958, releasing radionuclides up the stack and depositing contamination on the site from fallout. The system operated for a total of 126 hours.	SC&A (2015b) evaluated the three HTRE reactors and concluded that OTIB-0054 does not appear to adequately envelope them. Priority Ranking: N/A
27. Large Ship Reactor A (A1W-A) NRF	1958–1994	The large ship reactor facility (A1W) contained two Westinghouse pressurized water reactors (designated A and B), placed inside a section of a steel hull, that were prototypes for those installed in the *USS Enterprise* aircraft carrier.	NRF is not included in the EEOICPA program. Priority Ranking: N/A
28. Large Ship Reactor B A1W-B NRF	1959–1987	[See A1W-A description]	NRF is not included in the EEOICPA program. Priority Ranking: N/A

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
30. Materials Test Reactor (MTR) TRA (Bldg. TRA-603)	1952–1970	The MTR was the first materials test reactor at INL (it was succeeded by the ETR and the currently operating ATR). It was cooled and moderated with light water and used Al-clad, curved plate, highly enriched uranium fuel most of the time. Beryllium, graphite, and light-water neutron reflectors surrounded the relatively small core. Experiments were conducted external to the core, utilizing neutron fluxes exiting through beam holes. Its maximum power level increase from 30 MW_{th} to 40 MW_{th} in 1955. In 1958, it became the first reactor to operate with a plutonium-239 (Pu-239) core at up to 30 MW_{th} in a demonstration project.	Reviewed in SC&A report (SC&A 2015a): NIOSH SEC-00219: Test Reactor Area Modeling, SCA-SEC-2015-0074-C, Revision 0, September 28, 2015: "*the OTIB-0054 methodology, which explicitly models the ATR, should also adequately envelope the MTR in considering internal exposures when the latter reactor operated with uranium fuel*." "*The MTR also ran for a period of time with plutonium rather than uranium fuel, and at a lower power level and lower burnup. It is not clear which, if any, of the nine OTIB-0054 representative cases…would adequately envelope this situation*." Priority Ranking: N/A
32. Natural Circulation Reactor (S5G) NRF	1965–1995	The S5G, designed by General Electric, was the prototype of a pressurized water reactor for *USS Narwhal*. It could operate in a natural circulation cooling mode, which greatly reduced the noise produced, thereby making it more difficult to detect the submarine.	NRF is not included in the EEOICPA program. Priority Ranking: N/A
33. Neutron Radiography Facility (NRAD) ANL-W	1977–present	NRAD is a 250-kW_{th}, uranium-fueled, water pool-type, steady-state General Atomics TRIGA reactor. It is used primarily for non-destructive neutron radiography of irradiated fuel specimens.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A

Reactor Name & Location[a,b]	Operation Dates	Summary Description	Comments/Priority Ranking
45. Submarine Thermal Reactor (S1W, STR). Also known as the Submarine Prototype Reactor NRF	1953–1989	The Westinghouse S1W (Ship 1, Westinghouse) reactor was a prototype of the reactor that was installed in the *USS Nautilus* submarine. The S1W was located in a simulated submarine hull section and was used for testing and training purposes.	NRF is not included in the EEOICPA program. Priority Ranking: N/A
50. Transient Reactor Test Facility (TREAT) ANL-W	1959–1994	The TREAT experiment continued research and development for fast breeder reactors by simulating accident conditions leading to fuel damage. It was an air-cooled, graphite-moderated, uranium-fueled reactor, operated at a steady power level of 100 kW_{th}, and could produce pulses of up to 19 GW_{th} for experimental purposes. The fuel consisted of a 93% enriched graphite-uranium matrix, with zirconium clad fuel assemblies.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A
51. Zero Power Physics Reactor (Earlier name: Zero Power Plutonium Reactor) (ZPPR) ANL-W	1969–1992	ZPPR was a very low-power, air-cooled, split-table critical facility that was used to mock up other fast neutron spectrum reactors. 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.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A
52. Zero Power Reactor No. 3 (ZPR-III) ANL-W	1955–1970	ZPPR-III was the smallest of the ZPPR series of split-table critical facilities.	ANL-W is not included in the SEC-0219 definition. Priority Ranking: N/A

[a] The list and numbering scheme of the 52 INL 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; NRF = Naval Reactor Facility; LPTF = Low Power Test Facility; TAN = Test Area North; TRA = Test Reactor Area; WRRTF = Water Reactor Research Test Facility.