NIOSH Proposal for INL and ANL-W Reactor Prioritization for OTIB-0054 Evaluation

White Paper

National Institute for Occupational Safety and Health (NIOSH)

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Division of Compensation Analysis and Support

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Introduction

In March 2016, SC&A released SCA-TR-2016-SEC002, *INL SEC-00219 Reactor Prioritization for Evaluation of ORAU-OTIB-0054 Applicability, Revision 0*. The report was prepared in response to a request from the Advisory Board on Radiation and Worker Health (ABRWH) to screen the rest of the Idaho National Laboratory (INL) reactors not already reviewed with respect to ORAUT-OTIB-0054 to ensure that the prescribed approaches were bounding for INL reactors. ORAUT-OTIB-0054 is a primary tool used for internal dose reconstruction by assigning fission and activation product intakes via ratios tied to an indicator radionuclide.

In SCA-TR-2016-SEC002 Rev. 0, SC&A reviewed all 52 reactors in the history of the INL site and grouped them into high, medium, and low priority categories in regards to performing evaluations on whether OTIB-0054 is bounding for INL reactors. The assignment of a reactor to one of the priority categories considered the following per SC&A:

- reactor design factors such as fuel type, enrichment, cladding, moderator and coolant
- operational modes such as steady-state or periodic
- if the reactor operated within design limits or was intentionally taken outside of those limits (INL was a reactor “testing station”)

From the 52 total reactors the following were removed from consideration:

- 12 Argonne National Laboratory-West (ANL-W) reactors
- 4 Naval Reactor Facility reactors (outside of EEOICPA program)
- 3 Test Area North reactors already evaluated
- 3 Test Reactor Area reactors already evaluated
- 2 reactors cancelled before operations

The 28 INL reactors remaining were grouped into the categories listed in Table 1.

INL Priority Category	Number of Reactors
High	13
Medium	8
Low	7

Table 1: INL reactor priority in SCA-TR-2016-SEC002 Rev 0

For the remainder of this report only the high priority category INL reactors will be addressed as they were identified as the reactors of most concern in regards to OTIB-0054. NIOSH believes that the high priority category recognized in the SC&A report should be addressed first with the medium and low categories reserved for evaluation after completion of the high priority category reactors. The following reactors were identified by SC&A as being in the high priority category. The thirteen high priority INL reactors are presented in the table below:

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Reactor # in SC&A Report	Reactor Name
11	Cavity Reactor Critical Experiment (CRCE)
29	Loss of Fluid Test Facility (LOFT)
31	Mobile Low Power Reactor (ML-1)
35	Organic Moderated Reactor Experiment (OMRE)
36	Power Burst Facility (PBF)
39	Special Power Excursion Reactor Test I (SPERT-I)
40	Special Power Excursion Reactor Test II (SPERT-II)
41	Special Power Excursion Reactor Test IIII (SPERT-III)
42	Special Power Excursion Reactor Test IV (SPERT-IV)
43	Spherical Cavity Reactor Critical Experiment (SCRCE)
46	Systems for Nuclear Auxiliary Power 10A Transient No. 1 (SNAPTRAN-1)
47	Systems for Nuclear Auxiliary Power 10A Transient No. 2 (SNAPTRAN-2)
48	Systems for Nuclear Auxiliary Power 10A Transient No. 3 (SNAPTRAN-3)

Table 2: High priority category INL reactors in SCA-TR-2016-SEC002 Rev 0

On June 10, 2016, revision 1 of SCA-TR-2016-SEC002 was released. The primary change in the revision was a change in the prioritization of INL reactors for OTIB-0054 review. The primary change was the reduction of high priority category reactors from 13 to 7. Below are tables which summarize the difference in the number of reactors categorized for ORAU-OTIB-0054 applicability.

Priority Category	Revision 0	Revision 1
High	13	7
Medium	8	6
Low	7	15

Table 3: INL reactor priority in SCA-TR-2016-SEC002 Rev 1

Reactor # in SC&A Report	Reactor Name
29	Loss of Fluid Test Facility (LOFT)
35	Organic Moderated Reactor Experiment (OMRE)
36	Power Burst Facility (PBF)
39	Special Power Excursion Reactor Test I (SPERT-I)
40	Special Power Excursion Reactor Test II (SPERT-II)
41	Special Power Excursion Reactor Test IIII (SPERT-III)
42	Special Power Excursion Reactor Test IV (SPERT-IV)

Table 4: High priority category INL reactors in SCA-TR-2016-SEC002 Rev 1

In July 2016, SC&A released SCA-TR-2016-SEC010, *Argonne National Laboratory-West SEC-00224 Reactor Prioritization for Evaluation of ORAU-OTIB-0054 Applicability, Revision 0*. The report evaluated the 12 Argonne National Laboratory-West (ANL-W) reactors that were excluded from SCA-TR-2016-SEC002. The evaluation followed the same criteria as described earlier but also considered four factors “that reflect the scope of the population ‘at risk’ of uncontrolled/unmonitored exposures.” Those four factors are duration of reactor operation, frequency/intensity of operation, approximate number of workers potentially exposure during operation, and incidents/other factors with potential to contribute to the risk of unintended/unprotected exposures.

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ANL-W Priority Category	Number of Reactors
High	7
Medium	1
Low	4

Table 5: ANL-W reactor priority in SCA-TR-2016-SEC010 Rev 0

Again, focusing only on the high priority category the following seven reactors were identified as causing the most concern in regards to OTIB-0054.

Reactor # in SC&A Report	Reactor Name
6	Boiling Water Reactor Experiment No. I (BORAX-I)
7	Boiling Water Reactor Experiment No. II (BORAX-II)
8	Boiling Water Reactor Experiment No. III (BORAX-III)
9	Boiling Water Reactor Experiment No. IV (BORAX-IV)
10	Boiling Water Reactor Experiment No. V (BORAX-V)
17	Experimental Breeder Reactor No. I (EBR-I)
18	Experimental Breeder Reactor No. II (EBR-II)

Table 6: High priority category ANL-W reactors in SCA-TR-2016-SEC010 Rev 0

Discussion

The following narrative provides a review of each of the high priority category reactors for INL and ANL-W. NIOSH believes that combining the reactors from both sites is practical given the completion of both the INL and ANL-W 83.13 evaluation reports.

INL: Loss of Fluid Test Facility (LOFT)

The Loss of Fluid Test Facility (TAN-650) operated from 1973 - 1985. It was part of an international safety program for commercial power reactors. From the Historical Engineering Record – Idaho National Engineering and Environmental Laboratory – Test Area North, HAER No. ID-33-E (SRDB 83751) page 120, "The first nuclear test took place on December 10, 1978." NIOSH proposes that LOFT be removed from consideration for evaluation of OTIB-0054 applicability at this time because nuclear operations began after the INL SEC evaluation period. If the workgroup believes a review is necessary, we suggest that it be categorized along with other findings arising from SCA's review of the site profile, rather than with SEC-00219 findings.

INL: Organic Moderated Reactor Experiment (OMRE)

The Organic Moderated Reactor Experiment operated from 1957 - 1963. It was a concept reactor to demonstrate the technical and economic feasibility of using a liquid hydrocarbon as both coolant and moderator. NIOSH agrees that OMRE should be evaluated due to its unique design and that the Piqua Nuclear Generating Station, a municipal power reactor that was developed based on OMRE, is already part of the Special Exposure Cohort.

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INL: Power Burst Facility (PBF)

The Power Burst Facility operated from 1972 through 1985 and was located on the former site of SPERT-I. It was an extension of the reactor safety studies started with the Special Power Excursion Test reactors. As such, it was utilized for transient testing to produce accident conditions. Fuel failure was one outcome of the PBF testing. PBF utilized an intermediate enriched fuel with a unique ceramic fuel clad in 304 stainless steel. Due to the unique fuel, NIOSH agrees that PBF should be evaluated.

INL: Special Power Excursion Reactor Tests

Four SPERT reactors (I-IV) were constructed and tested between 1955 and 1970 in an effort to understand the mechanisms that resist power increases and terminate runaway power conditions. These reactors were designed and operated beyond their normal safety limits to study a wide range of variables, such as core configuration, coolant flow and reflection, moderation, void, and temperature coefficients. NIOSH proposes that a model for the most extreme experiment from all of the Special Power Excursion Reactor Tests (SPERT), in terms of possible departures from OTIB-0054, be used to represent the "bounding" case to cover all four SPERT reactors that are discussed below.

INL: Special Power Excursion Reactor Test I (SPERT-I)

Special Power Excursion Reactor Test I operated from June 1955 until the fall of 1964. It was an open-tank, light water-moderated and reflected reactor, originally using 93 % enriched uranium fuel. The reactor tank was four feet in diameter and 14 feet tall and water filled to a level two feet above the core. The SPERT-I tests were characterized by sudden power spikes arrested by shutdown or the self-limiting tendencies of the reactor itself. No provisions for heat removal or coolant circulation through the core were included. Three important tests for SPERT-I were:

1. A deliberate 2,300 MW_t power burst on November 5, 1962 that destroyed the highly-enriched uranium metal plate core and distorted the reactor vessel. SPERT-I was rebuilt and used low-enriched uranium rod fuel afterward.

2. Uranium oxide fuel (low enrichment - 4%) was for a 17,400 MW_t power burst on November 12, 1963.

3. Uranium oxide fuel (low enrichment 4%) was for a 35,000 MW_t power burst on April 14, 1964.

INL: Special Power Excursion Reactor Test II (SPERT-II)

Special Power Excursion Reactor Test II operated from March 1960 until August 1964. It continued the program of investigating the kinetic behavior of heterogeneous, water-moderated, reactor systems begun with SPERT-I. It was a closed, pressurized water reactor with a reactor tank 10 feet in diameter and 24. 5 feet deep. The coolant flow system designed for utilized either light water or heavy water. SPERT-II utilized 93% enriched uranium aluminum clad fuel plates and was not designed to operate at high temperatures and pressures.

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INL: Special Power Excursion Reactor Test III (SPERT-III)

Special Power Excursion Reactor Test III operated from December 1958 until June 1968. It provided the widest practical range of control over temperature, pressure, and coolant flow. Pressures from atmospheric to 2,500 psi, water temperatures from 68° to 670°F, and coolant flow rates ranging from zero to 20,000 gallons per minute with heat-removal capacities up to 60,000 kilowatts for durations of 30 minutes were attainable. SPERT-III utilized 4% enriched uranium stainless steel clad fuel plates.

INL: Special Power Excursion Reactor Test IV (SPERT-IV)

Special Power Excursion Reactor Test IV operated from July 1962 until August 1970. Unlike the other SPERT reactors, it was a twin-pool facility (open-tank) for detailed studies of reactor stability as affected by varying conditions, including forced coolant flow, variable height of water above the core, hydrostatic head, and other hydrodynamic effects. Most specifically, SPERT-IV made detailed studies of phenomena of instability first observed in the 1,300 excursions conducted at SPERT-I. SPERT-III utilized 93% enriched uranium aluminum clad fuel plates.

ANL-W: Boiling Water Reactor Experiment No. I (BORAX-I)

Boiling Water Reactor Experiment No. I operated from July 1953 to July 1954 but only during the summer and fall months because it was not housed in a building. It was the first of a series of five separate reactors designed to investigate the safety of boiling water reactors. Fuel assemblies were made of curved fuel plates containing a U-235-aluminum-alloy core that was clad with aluminum. In July 1954, it was subjected to a final deliberate destructive excursion to determine the inherent safety under extreme conditions. It operated at 1.4 MW_t. NIOSH proposes that BORAX-I be removed from consideration for evaluation of OTIB-0054 applicability, as nuclear operations of this reactor are during an existing SEC time period. NIOSH has already determined it is infeasible to reconstruct the fission product doses from this reactor due to limited to non-existent bioassay.

ANL-W: Boiling Water Reactor Experiment No. II (BORAX-II)

Boiling Water Reactor Experiment No. II operated from October 1954 to March 1955. With BORAX-II, ANL tested new core combinations using different enrichments of U-235 in the fuel plates. It operated at 6.4 MW_t. NIOSH proposes that BORAX-II be removed from consideration for evaluation of OTIB-0054 applicability as nuclear operations of this reactor are during an existing SEC time period. NIOSH has already determined it is infeasible to reconstruct the fission product doses from this reactor due to limited to non-existent bioassay.

ANL-W: Boiling Water Reactor Experiment No. III (BORAX-III)

Boiling Water Reactor Experiment No. III operated from July 1955 to April 1956. In 1955, ANL replaced the reactor core and added a turbine-generator to BORAX-II to test power distribution, and whether turbine contamination would be a significant problem in a boiling water reactor. The revised reactor configuration was designated BORAX-III. The core consisted of 90% enriched uranium aluminum alloyed fuel plates clad in aluminum. It operated at 12 MW_t. NIOSH proposes that BORAX-III be removed from consideration for evaluation of OTIB-0054 applicability as nuclear operations of this reactor are during an existing SEC time period. NIOSH has already determined it is infeasible to reconstruct the fission product doses from this reactor due to limited to non-existent bioassay.

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ANL-W: Boiling Water Reactor Experiment No. IV (BORAX-IV)

Boiling Water Reactor Experiment No. IV operated from July 1956 to June 1958. ANL-W replaced the BORAX-III core with a core of a maximum seventy-two ceramic uranium-thorium oxide fuel elements to test and demonstrate the feasibility of stable operation with a fuel that: (1) could operate at higher temperatures than the uranium core; and (2) was considered less reactive with water coolant in case of cladding rupture. It operated at 20 MW_t. NIOSH agrees that BORAX-IV should be evaluated for OTIB-0054 applicability due to the use of uranium-thorium oxide fuel during the operational period.

ANL-W: Boiling Water Reactor Experiment No. V (BORAX-V)

Boiling Water Reactor Experiment No. V operated from March 1962 to August 1964. It had essentially the same arrangement as BORAX-IV but had a nuclear superheat system added to determine the safety aspects and potential steam cycle efficiency increases. It operated at 40 MW_t. NIOSH proposes that BORAX-V be removed from consideration for evaluation of OTIB-0054 applicability since the core is the same as BORAX IV and the BORAX V primary function was to evaluate steam superheating.

ANL-W: Experimental Breeder Reactor No. I (EBR-I)

Experimental Breeder Reactor No. I operated from 1951 to 1963. It was an unmoderated and heterogeneous design with a thermal output of approximately 1 MW_t at full power. To permit the efficient removal of heat and to minimize neutron moderation a sodium-potassium (NaK) coolant was used. There were four cores utilized during operation:

1. Mark I (1951 – 1954): This core was 93% enriched unalloyed uranium in stainless-steel jackets.

2. Mark II (1954 – 1955): This core was 98% enriched uranium with a 2% zirconium alloy. On November 29, 1955, this core experienced a partial meltdown.

3. Mark III (1957-1962): This core was also composed of 98% enriched uranium with a 2% zirconium alloy and was used to prove that there was nothing intrinsically unsafe about a breeder reactor after the partial meltdown of the Mark II core.

4. Mark IV (1962 – 1963): This core was composed of 98.6% plutonium, 1.25% aluminum, and some impurities. The fuel was fabricated and canned at Rocky Flats in a joint effort with Argonne National Laboratory (Lemont, IL).

NIOSH proposes that the most bounding of the last two EBR-I cores be used. The first two cores are during the current SEC time period, where NIOSH has already determined it is infeasible to reconstruct internal doses to fission products. While it is initially believed the plutonium core will likely be bounding, some preliminary modeling would need to be performed on both cores to confirm this.

ANL-W: Experimental Breeder Reactor No. II (EBR-II)

Experimental Breeder Reactor No. II operated from September 1961 to September 1994. EBR-II was an unmoderated, sodium-cooled, fast-neutron power reactor, which was designed to produce 62.5 MW_t.

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Until September 1970, EBR-II consisted of a high-enriched uranium core surrounded on all sides by a blanket of depleted uranium. In September 1970, some of the EBR-II Reactor's depleted-uranium blanket subassemblies were replaced with stainless-steel reflector subassemblies. It utilized 67% enriched uranium fuel pins. Since the EBR-II fuel loading is significantly different from the Fast Flux Test Facility modeled in ORAU-OTIB-0054, NIOSH agrees that EBR-II should be evaluated for OTIB-0054 applicability.

Conclusions

- NIOSH proposes merging the INL and ANL-W high priority category reactors for evaluation of OTIB-0054 applicability. NIOSH also proposes that after the evaluation of the high priority category reactors is completed, any concerns regarding the medium and low priority category reactors can then be addressed.
- NIOSH proposes that the Loss of Fluid Test Facility (LOFT) be removed from consideration for evaluation of OTIB-0054 applicability at this time, due to nuclear operations not commencing until December 1978.
- NIOSH agrees that the Organic Moderated Reactor Experiment (OMRE) should be evaluated for OTIB-0054 applicability due to its unique moderator and coolant.
- NIOSH agrees that the Power Burst Facility (PBF) should be evaluated for OTIB-0054 applicability due to the use of ceramic fuel.
- NIOSH proposes that a model for the most extreme experiment from all of the Special Power Excursion Reactor Tests (SPERT), in terms of possible departures from OTIB-0054, be used to represent the "bounding" case to cover all four SPERT reactors.
- Boiling Water Reactor Experiment (BORAX) No. I, II, and III all ceased operations before the end of the approved SEC period for ANL-W. NIOSH proposes BORAX I-III be removed from consideration for evaluation of OTIB-0054 applicability as their operating years are covered by the SEC period when bioassay data is known to be incomplete and an infeasibility to reconstruct doses has already been established. NIOSH agrees that BORAX-IV should be evaluated for OTIB-0054 applicability due to the use of uranium-thorium oxide fuel. NIOSH proposes that BORAX-V be removed from consideration for evaluation of OTIB-0054 applicability since its primary function was to evaluate steam superheating with essentially the same configuration as BORAX-IV.
- NIOSH proposes that the most bounding of the last two EBR-I cores be used. While it is initially believed the plutonium core would be bounding, some preliminary modeling would need to be performed on all four cores to confirm this.
- NIOSH agrees that the Experimental Breeder Reactor No. II should be evaluated for OTIB-0054 applicability.

NIOSH recommends the review for the high priority category INL and ANL-W reactors for OTIB-0054 applicability be categorized as presented below based on the reasons provided.

Reactor #s for NIOSH-Proposed Grouping for OTIB-0054 Applicability Evaluation	Reactor Name
35	Organic Moderated Reactor Experiment (OMRE)
36	Power Burst Facility (PBF)
39, 40, 41, 42	Special Power Excursion Reactor Tests I-IV (SPERT I-IV))
9	Boiling Water Reactor Experiment No. IV (BORAX-IV)
17	Experimental Breeder Reactor No. 1 (EBR-I) Core 4
18	Experimental Breeder Reactor No. II (EBR-II)

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References

SC&A 2016. *INL SEC-00219 Reactor Prioritization for Evaluation of ORAUT-OTIB-0054 Applicability,* SCA-TR-2016-SEC002, Revision 0, March 2016

SC&A 2016. *INL SEC-00219 Reactor Prioritization for Evaluation of ORAUT-OTIB-0054 Applicability,* SCA-TR-2016-SEC002, Revision 1, June 2016

SC&A 2015. *Argonne National Laboratory-West SEC-00224 Reactor Prioritization for Evaluation of ORAUT-OTIB-0054 Applicability*, SCA-TR-2016-SEC010, Revision 0, July 2016

SC&A 2015. *NIOSH SEC-00219 Test Reactor Area Modeling*, SCA-SEC-201509974A, Revision 0, September 2015

SC&A 2015. *Review of NIOSH Strategy for Reconstructing Internal Doses to Workers at Test Area North,* SCA-TR-2015-SEC0074A, Revision 0, September 2015

ORAUT 2015. *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses,* ORAU-OTB-0054, Revision 4, August 2015

NIOSH 2015. *SEC Petition Evaluation Report – Petition SEC-00219 – Idaho National Laboratory Revision 01,* July 2015

NIOSH 2016. *SEC Petition Evaluation Report – Petition SEC-00224 – Argonne National Laboratory – West, Revision 00,* February 2016

Historical Engineering Record – Idaho National Engineering and Environmental Laboratory – Test Area North, HAER No. ID-33-E (SRDB 83751)

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