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**ADVISORY BOARD ON
RADIATION AND WORKER HEALTH**

National Institute for Occupational Safety and Health

**INL SEC-00219 AND ANL-W SEC-00224: SC&A RESPONSE TO
NIOSH REACTOR ANALYSIS PLAN AND CONSOLIDATION
OF ALL REACTOR MODELING COMMENTS**

**Contract No. 211-2014-58081
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Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 2 of 59
--------------------------------------	----------------------------------	--	----------------------------

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

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Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 3 of 59
--------------------------------------	----------------------------------	--	----------------------------

TABLE OF CONTENTS

Abbreviations and Acronyms	4
1.0 Introduction.....	7
2.0 NIOSH Proposal	14
3.0 SC&A Response to the NIOSH Proposal	16
4.0 Test Area North and Test Reactor Area Reactors.....	19
4.1 TAN Reactors and Associated Irradiated and Spent Fuel	19
4.2 TRA Reactors.....	21
5.0 References.....	23
Appendix A: Characterization of Workforce and Exposure Potential for Six Reactors Selected for Analysis at INL/ANL-W.....	25
Appendix B: Evaluation of Applicability of OTIB-0054 to Reconstructing Internal Doses at Test Area North.....	53

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 4 of 59
--------------------------------------	----------------------------------	--	----------------------------

ABBREVIATIONS AND ACRONYMS

ATR	Advanced Test Reactor
Advisory Board	Advisory Board on Radiation and Worker Health
AEC	Atomic Energy Commission
Al	aluminum
ANL-W	Argonne National Laboratory-West
ATR	Advanced Test Reactor
Be	beryllium
BORAX	Boiling Water Reactor Experiment
Bq	becquerel
CPP	Chemical Processing Plant
Cs	cesium
D ₂ O	deuterium oxide (“heavy” water)
DCF	dose conversion factor
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor
EPA	U.S. Environmental Protection Agency
ETR	Engineering Test Reactor
F	Fahrenheit
FGR	Federal Guidance Report
H&S	Health and Safety
H ₂ O	(“light”) water
HTRE	Heat Transfer Reactor Experiment
I	iodine
IET	initial engine test
INL	Idaho National Laboratory
LLI	lower large intestine
LOCA	loss-of-coolant accident
LOFT	Loss of Fluid Test Facility
LWR	light-water reactor
mrem	millirem
ms	millisecond

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Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 5 of 59
--------------------------------------	----------------------------------	--	----------------------------

MTR	Materials Test Reactor
MTU	metric ton uranium
MW	megawatt
MWe	megawatt electric
MWth	megawatt thermal
Na	sodium
NaK	sodium-potassium (liquid metal)
Nb	niobium
NRC	U.S. Nuclear Regulatory Commission
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
Pr	praseodymium
psig	pounds per square inch gage
Pu	plutonium
RaLa	radioactive lanthanum
Ru	ruthenium
SEC	Special Exposure Cohort
SL-1	Stationary Low-Power Reactor
SNAP	Systems for Nuclear Auxiliary Power
SPERT	Special Power Excursion Reactor Test(s)
Sr	strontium
SRDB	Site Research Database
TAN	Test Area North
TBD	technical basis document
TRA	Test Reactor Area
U	uranium
UO ₂	uranium dioxide
WG	work group

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Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 6 of 59
--------------------------------------	----------------------------------	--	----------------------------

wt% weight percent

Y yttrium

Zr zirconium

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 7 of 59
--------------------------------------	----------------------------------	--	----------------------------

1.0 INTRODUCTION

This report consolidates and summarizes SC&A’s recommendations to the Advisory Board on Radiation and Worker Health (Advisory Board) Idaho National Laboratory (INL) Work Group (WG) with respect to prioritizing reactors for further analysis by the National Institute for Occupational Safety and Health (NIOSH) as part of two Special Exposure Cohort (SEC) evaluations. The report includes all reactors at the INL site that were discussed in four different SC&A reports (SC&A 2015a, 2015b, 2016b, 2016c), both those designated as INL reactors and those designated as Argonne National Laboratory-West (ANL-W) reactors. This report also contains SC&A’s responses to a NIOSH report (NIOSH 2016b) that commented on SC&A’s recommendations from two of the four (SC&A 2016b, 2016c) abovementioned SC&A reports and, at the WG’s request, information on occupancy and exposure potential in various reactor areas.

To place this report in context, it is necessary first to discuss some background material on how the reactor prioritization exercise proceeded and evolved in response to WG requests and comments on and discussions about several SC&A and NIOSH reports.

NIOSH issued its INL SEC Petition SEC-00219 evaluation report (NIOSH 2015a) on March 12, 2015, and a subsequent revision on July 21, 2015 (NIOSH 2015b). NIOSH issued its ANL-W SEC Petition SEC-00224 evaluation report (NIOSH 2016a) on February 18, 2016. In a series of meetings and discussions among the INL WG, SC&A, NIOSH, and NIOSH’s technical contractor, Oak Ridge Associated Universities (ORAU), the WG directed SC&A to review the issue of dose reconstructability for both INL and ANL-W reactors (both sets of reactors and facilities are located within the overall INL site). 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 definition¹ and that are not being held in reserve for further evaluation by NIOSH. Operations at INL and ANL-W 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.

A primary tool NIOSH uses for internal dose reconstruction is the guidance in ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses*, Revision 04, dated August 27, 2015 (ORAUT 2015; hereafter referred to as “OTIB-0054”). 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. SCA-SEC-2015-0074-C, *NIOSH SEC-00219: Test Reactor Area Modeling* (SC&A 2015a), evaluated whether OTIB-0054 is applicable to the three large materials-testing reactors located in the INL Test Reactor Area (TRA): the Materials Test Reactor (MTR), the Engineering Test Reactor (ETR), and the Advanced Test Reactor (ATR). SCA-TR-2015-SEC0074A, *Review of NIOSH Strategy for Reconstructing Internal Doses to Workers at Test Area North* (SC&A 2015b), similarly

¹ It should be noted that the INL SEC is based on internal alpha exposures only. NIOSH plans to reconstruct internal doses from fission and activation products at the Central Processing Plant (CPP) 1963–1974.

Effective Date:	Revision No.	Document No./Description:	Page No.
12/08/2016	0 (Draft)	SCA-TR-2016-SEC012	8 of 59

evaluated whether OTIB-0054 is applicable to the three Heat Transfer Reactor Experiment (HTRE) reactors located in the INL Test Area North (TAN) that supported the Aircraft Nuclear Propulsion program.²

At the November 10, 2015, INL WG meeting, the Advisory Board members directed SC&A to screen INL reactors other than the six already addressed and create a prioritized list of reactors for detailed examination later with respect to OTIB-0054 applicability. Similarly, at its March 23–24, 2016, meeting, the Advisory Board directed SC&A to screen the ANL-W reactors. SC&A responded to these requests in its INL reactor prioritization report (SC&A 2016a) of March 2, 2016, and Revision 1 (SC&A 2016b) of June 10, 2016, that responded to some Advisory Board comments, and in its ANL-W reactor prioritization report (SC&A 2016c) of July 13, 2016.

In prioritizing reactors for further investigation, SC&A focused on the degree to which the abundance of fission and activation products and actinides relative to the abundance of Cs-137 and Sr-90 are thought to bear any resemblance to the mix of radionuclides in OTIB-0054. In addition to OTIB-0054 applicability, the 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 or unprotected exposures

The priority rankings of the INL and ANL-W reactor prioritization reports were divided into three categories: High, Medium, and Low.³ Though based on a substantial amount of research, the rankings were still somewhat subjective because a full analysis would involve detailed and extensive research for each reactor and performing the OTIB-0054 applicability analyses themselves, which would be counter to the limited objectives of the 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 [Al] or steel), moderator (e.g., “light” water [H₂O], “heavy” water [D₂O], or beryllium), and coolant (e.g., H₂O, nitrogen gas, or organic liquid); operational mode (e.g., steady-state, periodic, or pulsed); 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.⁴ These screening criteria were selected because SC&A judged them to be those that would best

² Discussions of the TRA and TAN reactor prioritization recommendations appear in Section 4 of this report.

³ There is also a category for those reactors that are not considered for various reasons in the prioritization process.

⁴ Information to address exposure potential was not readily available at the time the reactor prioritization reports were written but is included here in Appendix A.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 9 of 59
--------------------------------------	----------------------------------	--	----------------------------

indicate 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 INL who worked in the vicinity of these reactors or worked with irradiated fuel from these reactors.

After excluding reactors that either were located in ANL-W (12) or the Naval Reactors Facility (4), were never operated (2), or the TRA and HTRE reactors that were previously studied in SC&A 2015a and 2015b (6), the initial list of 52 reactors on the site was reduced to 28 for INL. Of those remaining, the SC&A INL reactor prioritization report (SC&A 2016b) categorized seven reactors in the “High Priority” class. Table 1 (adapted from Table A.1 of SC&A 2016b) lists the reactors and provides summary information.

Table 1. INL Reactors: High Priority Rankings

Reactor Name^(a)	Operation Dates	Summary Description
29. Loss of Fluid Test Facility (LOFT)	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) up to a worst-case, double-ended break in the primary coolant system. The reactor had a maximum power of 50 megawatts thermal (MWth), and the associated components and systems were built as a volumetrically scaled model of a commercial, four-coolant-loop pressurized-water reactor, including its engineered safety features.
35. Organic Moderated Reactor Experiment (OMRE)	1957–1963	The OMRE reactor was part of an Atomic Energy Commission (AEC) program to assess the feasibility and determine the nuclear and engineering technical basis of different reactor concepts in support of a civilian nuclear power industry. OMRE used a waxy liquid hydrocarbon as both coolant and moderator. The relatively low-power (5–10 MWth), critical reactor tested various types and configurations of highly enriched uranium dioxide (UO ₂) fuel elements and gathered performance data on the coolant, as well as nuclear data. The organic coolant was thought to have some advantages over “conventional” coolants because it allows low-pressure operation, solidifies at low temperatures, and does not corrode metals.
36. Power Burst Facility (PBF)	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 commercial nuclear power plant, led to fuel and cladding damage accompanied by the evolution of hydrogen and the release of fission products to the reactor containment. Simulated LOCAs and other severe accident tests were performed in an experimental loop within the reactor core. The PBF could produce very high, short-duration (millisecond [ms]) power excursions that were self-limiting. It could operate at a steady-state power of 20 MWth for a short period of time before initiating a very short super-critical power burst.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 10 of 59
--------------------------------------	----------------------------------	--	-----------------------------

Reactor Name^(a)	Operation Dates	Summary Description
39. Special Power Excursion Reactor Test No. 1 (SPERT-I)	1955–1964	<p>The four reactors in the SPERT program were deliberately subjected to large, rapid reactivity excursions to gather data on coupled neutronic and thermal-hydraulic responses as part of an AEC 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 flows, 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.</p> <p>SPERT-I, the first reactor of the program, was an open-tank, light-water-moderated and reflected reactor, with the uranium fuel enriched up to 93.5%. Some experiments were also conducted with fuel enriched only a few percent to better simulate power reactor fuel. The fuel consisted of plate type uranium and aluminum fuel assemblies in a 4-foot diameter and 14-foot deep carbon steel tank, clad with aluminum. 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 deliberate 2,300 MWth excursion on November 5, 1962, resulted in an explosion that completely melted approximately 8% and partially melted about 35% of the plate-type core and even distorted the reactor vessel. Subsequently, SPERT-I was rebuilt, and low-enriched fuel rods replaced the high-enriched fuel plates. A deliberate 17,400 MWth excursion on November 12, 1963 (with 4% UO₂ fuel rods), and a deliberate 35,000 MWth excursion on April 14, 1964 (with 4% UO₂ fuel rods), tested the resilience of the fuel rods; the latter test damaged some of them. SPERT-I underwent about 1,300 kinetic tests with six different cores in its 10-year lifetime.</p>
40. Special Power Excursion Reactor Test No. 2 (SPERT-II)	1960–1964	<p>SPERT-II continued to investigate transient behavior in a reactor that modelled a commercial reactor. Several different types of fuel assemblies were used, both light and heavy water were tested as moderators and coolants, and different reflectors were also used. Unlike SPERT-I, SPERT-II was placed in a closed pressure vessel and the coolant system was pressurized. Each fuel plate contained a 93.5% enriched uranium-aluminum alloy and was clad in aluminum. Because the reactor operated in the transient, burst mode, with power excursions up to 20 megawatt (MW)-sec, total burnup was small.</p>

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 11 of 59
--------------------------------------	----------------------------------	--	-----------------------------

Reactor Name^(a)	Operation Dates	Summary Description
41. Special Power Excursion Reactor Test No. 3 (SPERT-III)	1958–1968	SPERT-III accommodated the widest variation in several important parameters, such as 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 degrees Fahrenheit (°F) and pressure of 2,500 pounds per square inch gage (psig) also simulated nuclear power plant conditions. The system could produce a maximum of 60 MWth 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 reactor used ordinary water as coolant, moderator, and reflector.
42. Special Power Excursion Reactor Test No. 4 (SPERT-IV)	1962–1970	SPERT-IV also investigated transient reactor behavior to provide neutronic and thermal-hydraulic data applicable especially to large, open pool reactors; the open pool design allowed direct observation of reactor performance under different hydrodynamic conditions. The fuel consisted of a 93.5% enriched uranium-aluminum matrix in a plate-type configuration. 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.

Note:

(a) The reactor numbering scheme is taken from Stacy 2000.

SC&A's ANL-W reactor prioritization report (SC&A 2016c) analyzed the 12 ANL-W reactors in a similar fashion as SC&A 2016b analyzed the 28 INL reactors and categorized seven reactors in the High Priority class. Table 2 (adapted from Attachment 1 of SC&A 2016c) lists the reactors and provides summary information.

Table 2. ANL-W Reactors: High Priority Rankings

Reactor Name^(a)	Operation Dates	Summary Description
6. Boiling Water Reactor Experiment No. 1 (BORAX-I)	1953–1954	<p>The BORAX series of reactor experiments tested the feasibility and safety and explored the operating parameters of direct steam production in an LWR. The chronology of the BORAX reactors is:</p> <p>BORAX-I: 7/1953–7/1954 (no surrounding structure)</p> <p>BORAX-II: 10/1954–3/1955 (new facility)</p> <p>BORAX-III: 7/1955–4/1956 (modification of B-II)</p> <p>BORAX-IV: 12/1956–6/1958 (modification of B-III)</p> <p>BORAX-V: 3/1962–8/1964 (new facility).</p> <p><u>BORAX-I</u>: The 1.4 MWth reactor was water moderated and relied on natural water cooling and steam formation to remove heat. It was housed in a 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 stationary low-power reactor (SL-1) plant at INL.</p> <p>The core was built up from a lower grid and consisted of 26 curved plate fuel assemblies of a uranium-235 (U-235)-aluminum alloy clad with aluminum. Each assembly contained 18 fuel plates joined to aluminum side plates. 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.”</p>
7. Boiling Water Reactor Experiment No. 2 (BORAX-II)	1954–1955	<p>BORAX-II, at 6.4 MWth, 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. It was much larger than BORAX-I and designed to be more representative of a future commercial boiling-water reactor; e.g., it operated at a design pressure of 300 psig. It released excess energy as steam, because it had no turbine generator attached. The reactor was water cooled (natural circulation), moderated, and reflected.</p>
8. Boiling Water Reactor Experiment No. 3 (BORAX-III)	1955–1956	<p>The BORAX-II reactor was modified (raising its output to 12 MWth) and renamed BORAX-III after the addition of a 2.5 megawatt electric (MWe) steam turbine-generator to investigate radioactive contamination of the turbine from radioactivity in the primary coolant and to demonstrate production of electricity. The fuel consisted of plate-type fuel elements of a 90% enriched uranium-aluminum alloy with aluminum cladding. By the time it went out of service in 1956, BORAX-II had operated for a total of 1,170 hrs.</p>

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 13 of 59
--------------------------------------	----------------------------------	--	-----------------------------

Reactor Name^(a)	Operation Dates	Summary Description
9. Boiling Water Reactor Experiment No. 4 (BORAX-IV)	1956–1958	BORAX-IV, a modification of BORAX-III, at 20 MWth (and 2.5 MWe) 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. Operating 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.
10. Boiling Water Reactor Experiment No. 5 (BORAX-V)	1962–1964	BORAX-V, a new facility with basically the same configuration as BORAX-IV, but with an integral nuclear superheat system, operated at 40 MWth. The reactor was located 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.
17. Experimental Breeder Reactor No. I (EBR-I)	1951–1963	The EBR-I fast breeder reactor demonstration was the first reactor built at INL. It had a maximum power of 1 MWth 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 addition to producing the first electricity from a nuclear plant, EBR-I also demonstrated the feasibility of fuel production (breeding). In fact, at full power of 1 MWth, 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 either with 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.
18. Experimental Breeder Reactor No. II (EBR-II)	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 MWth, and the EBR-II could supply 20 MWe of electric power to INL facilities. In addition to demonstrating fuel reprocessing and electricity production in a liquid metal fast breeder reactor, EBR-II also performed irradiation, fuel development, and transient stability experiments.

Note:

(a) The reactor numbering scheme is taken from Stacy 2000.

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Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 14 of 59
--------------------------------------	----------------------------------	--	-----------------------------

2.0 NIOSH PROPOSAL

NIOSH responded to SC&A's INL and ANL-W reactor prioritization reports (SC&A 2016b, 2016c) with its *NIOSH Proposal for INL and ANL-W Reactor Prioritization for OTIB-0054 Evaluation*, July 28, 2016 (NIOSH 2016b). The report reviews the SC&A high priority reactor lists and makes eight recommendations that inform how NIOSH would proceed with further reactor modeling work to determine how well OTIB-0054 envelopes the INL and ANL-W reactors. Table 3 reproduces NIOSH's proposal conclusions.

Table 3. NIOSH Proposal Conclusions

No. ^(a)	NIOSH Recommendation ^(b)
1	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.
2	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.
3	NIOSH agrees that the Organic Moderated Reactor Experiment (OMRE) should be evaluated for OTIB-0054 applicability due to its unique moderator and coolant.
4	NIOSH agrees that the Power Burst Facility (PBF) should be evaluated for OTIB-0054 applicability due to the use of ceramic fuel.
5	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.
6	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.
7	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, It is not clear to SC&A that the Mark II core material was ever handled outside the SEC period, which would necessitate modeling it.
8	NIOSH agrees that the Experimental Breeder Reactor No. II should be evaluated for OTIB-0054 applicability.

Source: NIOSH 2016b.

Notes:

(a) Numbering added here reflects the order in which each point appeared in the original bulleted list in NIOSH 2016b.

(b) Recommendations copied from the NIOSH 2016b Conclusions section.

The NIOSH report ends with a recommendation for the six reactors (or series of reactors) to model with detailed analyses using the ORIGEN (Croff 1980) and SCALE (ORNL 2015) system of isotopic buildup and decay codes. Those reactors and the NIOSH-proposed groupings appear here in Table 4.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 15 of 59
--------------------------------------	----------------------------------	--	-----------------------------

Table 4. Reactors for NIOSH Proposed Grouping for OTIB-0054 Applicability Evaluation

Reactor Numbers	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)

3.0 SC&A RESPONSE TO THE NIOSH PROPOSAL

NIOSH presented its response (NIOSH 2016b) to the two SC&A reactor prioritization studies at the August 2, 2016, meeting of the INL/ANL-W WG. The WG members then directed SC&A to comment on the NIOSH proposal with respect to OTIB-0054 adequacy considerations, as well as the potential for personnel exposure at each of the six reactors. Table 5 presents SC&A's assessment of NIOSH's proposal, and Appendix A of this report presents SC&A's detailed findings on the potential for personnel exposure at the six reactors.

As Appendix A shows, the prioritized reactor sites generally employed hundreds of monitored workers, except for the PBF. This facility only appears to have assigned 30 workers during most badging cycles in the available records. Penetrating doses to workers at the prioritized reactor sites were also significant, with some monthly badging cycles averaging on the order of hundreds of millirem (mrem) (see Sections A.2 and A.4 on EBR-II and SPERT, respectively). While external exposure rates are not necessarily directly correlative to internal exposure potential, the magnitude of these accrued external doses is indicative of the source terms being considered. Coupled with the extensive internal dosimetry program at INL for fission products, an adequate characterization of the mix of source term contaminants appears warranted.

Table 5. SC&A Responses to NIOSH Proposal

No. (a)	NIOSH Recommendation ^(b)	SC&A Response
1	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.	SC&A concurs. Whether a reactor is classified as an INL reactor or an ANL-W reactor is immaterial to the reactor modeling work. Treating them together and generating a single report would reduce repetition in NIOSH's report-writing and the Board's and SC&A's reviewing efforts.
2	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.	SC&A recognizes that the first five LOFT experiments were non-nuclear, thermal-hydraulic experiments and that the potential for radiation exposure did not occur until December 1978, which is after the INL SEC-00219 period. SC&A believes that, given the facility's size, long operating history, beyond-design-basis operating scenarios, and potential to have exposed a significant number of personnel, the LOFT reactor merits a more detailed examination with respect to whether it can be adequately modelled by OTIB-0054. Such an examination could be conducted as a site profile exercise.
3	NIOSH agrees that the Organic Moderated Reactor Experiment (OMRE) should be evaluated for OTIB-0054 applicability due to its unique moderator and coolant.	SC&A agrees with NIOSH's characterization with respect to OTIB-0054 and notes (Section A.5) that, based on the limited data available (only for the last year of operation), there appears to be a significant potential for exposure of hundreds of regular workers and visitors.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 17 of 59
--------------------------------------	----------------------------------	--	-----------------------------

No.^(a)	NIOSH Recommendation^(b)	SC&A Response
4	NIOSH agrees that the Power Burst Facility (PBF) should be evaluated for OTIB-0054 applicability due to the use of ceramic fuel.	SC&A agrees with NIOSH's characterization with respect to OTIB-0054 and notes (Section A.3) that, based on the limited data available (only for the first few years of operation), there appears to be a potential for exposure of typically less than 100 regular workers and visitors per month.
5	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.	SC&A disagrees with NIOSH's recommendation. Although the four SPERT reactors were all part of the same series of reactor experiments that subjected the reactor systems to large reactivity excursions, as seen in the summaries of Table 1, they differed significantly from each other and should be examined separately, perhaps by choosing the "worst case" scenario for each reactor. NIOSH should justify in its report its choice, perhaps by performing some preliminary calculations to determine the "bounding case."
6	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.	SC&A concurs with NIOSH's assessment about OTIB-0054 and notes (Appendix A) that individual documentation concerning the workforce at the BORAX-IV experiment in 1958 could not be located and, therefore, is not discussed further.
7	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.	SC&A concurs with NIOSH about OTIB-0054 and expects that the resulting NIOSH report will make a compelling case for which core is bounding. In addition, SC&A notes (Section A.1) that several hundred workers and visitors were present during the period of operation for the MARK IV core.
8	NIOSH agrees that the Experimental Breeder Reactor No. II should be evaluated for OTIB-0054 applicability.	SC&A concurs with NIOSH about OTIB-0054 and notes (Section A.2) that hundreds of workers and visitors could have been exposed each year; in some years, the average worker penetrating doses were greater than 100 mrem.

Notes:

(a) Numbering added here reflects the order in which each point appeared in the original bulleted list in NIOSH 2016b.

(b) Recommendations copied from the NIOSH 2016b Conclusions section.

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Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 18 of 59
--------------------------------------	----------------------------------	--	-----------------------------

Note that SC&A's recommendations for further analysis for TAN and TRA reactors are discussed in Section 4.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 19 of 59
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4.0 TEST AREA NORTH AND TEST REACTOR AREA REACTORS

SC&A verbally presented its draft results, summarized in Table 5 of this report, to an INL WG teleconference on September 8, 2016. At the recommendation of NIOSH and at the direction of the WG, SC&A includes in this section its observations and recommendations pertaining to the TRA and TAN reactors that were the subject of investigations in SC&A's September 28, 2015, review of NIOSH internal dose reconstruction at TRA (SC&A 2015a) and SC&A's September 28, 2015, TAN evaluation report (SC&A 2015b).⁵ Hence, this document now contains discussions of all the INL and ANL-W reactors of interest in the two SEC evaluations.

4.1 TAN REACTORS AND ASSOCIATED IRRADIATED AND SPENT FUEL

One of the topics addressed in SC&A's September 28, 2015, TAN evaluation report (SC&A 2015b) is the applicability of OTIB-0054 and Tables 5-22 and 5-23 of the INL technical basis document (TBD) (NIOSH 2010) to the performance of internal dose reconstruction for facilities that handled and stored spent and irradiated fuel at TAN. The spent and irradiated fuel from the HTREs was of particular interest because the reactor, fuel, and operational combinations that underpin the OTIB-0054 methodology reflect situations in which burnup often occurred over protracted periods of time (hundreds of days) and the fuel maintained its integrity during burnup. In contrast, the fuel at HTRE had very short burnup times and the reactors operated at high temperatures, allowing the fuel to melt. In addition, the HTRE spent and irradiated fuel employed highly enriched uranium, as opposed to the enrichment levels of the uranium in the fuel used to derive the mix of fission and activation products provided as default mixes in OTIB-0054 and TBD Tables 5-22 and 5-23.

To provide additional information to address these potential concerns, preliminary analyses reported in SC&A 2015b included generic ORIGEN-ARP⁶ runs that SC&A had made, where the isotopic mixtures of fission and activation products were compared at different lengths of continuous operation at 20 MW (low power) and at 200 MW (high power): 20 hours and 200 days for both power levels, and an additional 20 day-run for the high-power case.⁷ These scenarios were intended to provide a general representation of the operating conditions of the HTRE tests at INL (e.g., HTRE-1 was operated at 20 MW for 150.8 hours). The results of the ORIGEN runs were used to help assess the suitability of using OTIB-0054 reactor configurations and operating conditions to represent the TAN reactor operations.

The first set of calculations assumed a high power level (200 MW) and derived the mix of fission products in the fuel relative to Sr-90 after a 10-day cool-down period, which was assumed instead of the longer cool-down periods in Table 7-3a of OTIB-0054 to minimize the relative importance of the longer-lived fission products. Only the ATR1 case of Table 7-3a was investigated, because SC&A believes the results would also apply (well enough to inform the

⁵ Note that while NIOSH commented on the two SC&A reactor prioritization reports (SC&A 2016b, 2016c) in the proposal report (NIOSH 2016b), it never commented on the SC&A TRA (SC&A 2015a) and TAN (SC&A 2015b) reactor reports.

⁶ ORIGEN-ARP is a version of the ORIGEN code contained in the SCALE system of nuclear codes (ORNL 2015).

⁷ It is understood that power levels for the TAN reactors refer to "thermal power," since none of the reactors produced electricity.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 20 of 59
--------------------------------------	----------------------------------	--	-----------------------------

investigation) to the other reactor types in OTIB-0054 in the preliminary analysis that was performed. The second set of calculations assumed a low power level (20 MW).

For each of the two assumed power levels, SC&A ran ORIGEN for different fuel burnup times. The 200-day cases are indicative of the long burnup times used to derive the relative mixes of radionuclides in Tables 7-3a through 7-3i (the nine characteristic reactor cases) of OTIB-0054, while the shorter burnup times (20 days and 20 hours) are typical of many of the TAN experiments. The SC&A analysis assumed a fuel cool-down period of 10 days, and then multiplied the relative amounts of fission products produced by selected organ dose conversion factors (DCFs), which yielded a relative “index of harm” for each fission product. SC&A then summed the indices of harm for the fission products for each of the burnup durations. The detailed results of these calculations are provided in Tables 1 and 2 of SC&A 2015b. In summary, the results of these investigations revealed the following for fission products:

- For the high power level (200 MW), the indices of harm for the 20-day burnup and the 20-hour burnup were comparable to, slightly higher than, or slightly lower than (i.e., claimant favorable), the 200-day burnup for all organs of concern, except for the thyroid, for which the relative index of harm was substantively higher (i.e., 8.29).
- For the low power level (20 MW), the derived indices of harm for the 20-hour burnup compared to the 200-day burnup for all organs of concern were not claimant favorable.

Given the complexity of this subject, SC&A took additional steps in this report to evaluate the degree to which OTIB-0054 is, in fact, applicable or claimant favorable to workers at TAN facilities, keeping in mind that OTIB-0054 is used to reconstruct the intake rates of individual radionuclides by TAN workers who were routinely monitored on a gross beta/gamma urine bioassay program. The results of this additional investigation appear in Appendix B.

For actinide activation products (the results are in Tables 3 and 4 of SC&A 2015b), SC&A found that, for all cases analyzed, the ratios of the inventories of all actinide activation products to the inventories of Cs-137 and Sr-90 were grossly overestimated compared to the ratios in Tables 5-22 and 5-23 of the TBD. SC&A’s scoping analyses stopped at this point.

SC&A recommends that NIOSH continue these types of investigations to better understand the applicability and limitations of OTIB-0054 and Tables 5-22 and 5-23 of the TBD applied to reconstructing internal doses to workers at TAN involved in irradiated and spent fuel operations, where the power levels and burnup durations were significantly different from those upon which the isotopic mixes were derived in OTIB-0054 and Tables 5-22 and 5-23 of the TBD.⁸

Although SC&A was unable to locate specific information concerning workforce and exposure potential for HTRE personnel during the years of its operation, SC&A compiled data for the TAN area in the years immediately following the operation of the HTRE. This information is described in Section A.6 of Appendix A. For the years for which information was available, there were generally 250–300 regular badges and 50–150 “other” badges issued in the TAN area

⁸ It should be noted that a similar situation might also apply to other reactors in the INL complex.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 21 of 59
--------------------------------------	----------------------------------	--	-----------------------------

per badging cycle. The maximum penetrating dose for TAN was generally between 200 and 800 mrem per badging cycle, with the highest value (1.15 rem) during a 2-week period in 1965.

4.2 TRA REACTORS

SC&A's September 28, 2015, review of NIOSH internal dose reconstruction at TRA (SC&A 2015a) examined the three major TRA reactors and their operations to determine if they are adequately enveloped by OTIB-0054 methodology and data: the MTR (1952–1970), the ETR (1957–1981), and the ATR (1967–present). All are material testing reactors with similar designs, but with size, power level, and capabilities increasing from the first to the last; the maximum power level was 40 MW for the MTR, 175 MW for the ETR, and 250 MW for the ATR.⁹ Their high-flux capability allows for the accelerated simulation of long-term irradiation of reactor materials. Each is a pressurized, light-water-moderated, beryllium-reflected reactor, primarily using highly enriched uranium fuel (93.15% nominal enrichment) arranged in an unusual curved plate configuration. The primary reactivity control mechanism consists of a unique design of rotating beryllium cylinders with hafnium (a very effective thermal neutron absorber) shells.

SC&A 2015a found that OTIB-0054 adequately enveloped the three reactors (the OTIB explicitly models the ATR) for uranium fuel operations but noted that the MTR also ran for a period with plutonium fuel. Although the MTR was initially intended to use uranium fuel, in 1958 it became the first reactor run with a plutonium-239 (Pu-239) core. The MTR-Phoenix experiment was a demonstration project for a potential high-power, compact reactor to convert the fertile Pu-240 to the fissile Pu-241 through neutron capture, thereby extending the lifetime of the fuel and avoiding loading the core with excessive reactivity and neutron absorbers at startup, as the Pu-240 neutron absorber was slowly replaced by the fissile Pu-241. The experiment used curved fuel plates similar in configuration to the standard MTR fuel plates, where each Al-clad plate contained 21 weight percent (wt%) Pu and 79 wt% Al. The reactor reached initial power on January 28, 1970, and operated at power levels up to 24 MW until April 23, 1970, at which time it had accumulated a burnup of 923 megawatt days. This compared to the standard U-235 core, which operated at 40 MW.

It is not clear which, if any, of the nine OTIB-0054 representative reactor cases would adequately envelope the MTR with plutonium fuel. Although the MTR plutonium fuel operations used fuel plates physically similar to those used in the uranium fuel, and the rest of the reactor configuration was not significantly modified, the nuclear properties of plutonium (e.g., cross sections) differ from those of uranium, and the fission product abundance distribution and core neutron spectrum (and, hence, activation product yield) would be different. How much different, and whether the differences would be radiologically significant, would require detailed comparative ORIGEN runs, which were not done for the SC&A 2015a report. The issue of whether OTIB-0054 adequately envelopes the MTR when fueled with plutonium merits further investigation and discussion by NIOSH.

⁹ It is understood that power levels for the TRA reactors refer to “thermal power,” since none of the reactors produced electricity.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 22 of 59
--------------------------------------	----------------------------------	--	-----------------------------

Another “unusual” operating scenario for the MTR was the radioactive lanthanum (RaLa) extraction campaign from 1956 through 1963. The fuel was irradiated for only 17 days, then quickly removed and dissolved to recover the short half-life fission product, which was used at Los Alamos National Laboratory to test implosion technologies (explosive lensing) to compress plutonium pits. This campaign does not have to be considered here because, as stated in Section 3.0 of OTIB-0054, “It [OTIB-0054] does not apply to operations involving decay times shorter than 10 days (e.g., early radioactive lanthanum (RaLa) processing).”

SC&A compiled available data on the workforce and exposure potential in the TRA area, summarized in Section A.7 of Appendix A. In general, the TRA area issued approximately 1,000 regular badges per exchange period from mid-1958 through mid-1967, with spikes greater than 4,000 badges for some periods. In addition, most badging periods also issued several hundred “other” badges, which are issued to visitors or non-routine workers. Construction workers were issued separate dosimeters aside from the “other” category. In general, the number of construction dosimeters issued ranged between 100 and 200 per badging period.

Maximum penetrating exposures were approximately 250 mrem for weekly badging exchanges (1956–1958), 500–1,000 mrem for bi-weekly exchanges (1959–1969), and averaged about 1,000 mrem for monthly badging cycles (1970–1974). Construction worker exposures were generally in the 100–200 mrem range. Several badging cycles for construction workers showed no accrued penetrating external dose.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 23 of 59
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Effective Date:	Revision No.	Document No./Description:	Page No.
12/08/2016	0 (Draft)	SCA-TR-2016-SEC012	24 of 59

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Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 25 of 59
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APPENDIX A: CHARACTERIZATION OF WORKFORCE AND EXPOSURE POTENTIAL FOR SIX REACTORS SELECTED FOR ANALYSIS AT INL/ANL-W

Prepared by
Bob Barton, SC&A

The National Institute for Occupational Safety and Health (NIOSH) presented its response (NIOSH 2016b) to SC&A's reactor prioritization reports (SC&A 2016a, 2016b, 2016c) during the Idaho National Laboratory (INL)/Argonne National Laboratory-West (ANL-W) Work Group (WG) meeting on August 2, 2016, and recommended that six reactors be considered for the initial analyses:

- (1) Experimental Breeder Reactor No. I (EBR-I) MARK IV: 1962–1963
- (2) Experimental Breeder Reactor No. II (EBR-II): 1961–1994
- (3) Power Burst Facility (PBF): 1972–1985
- (4) Boiling Water Reactor Experiment No. 4 (BORAX-IV): 1958
- (5) Special Power Excursion Reactor Tests (SPERT): 1958–1970
- (6) Organic Moderated Reactor Experiment (OMRE): 1958–1963

At the August 2016 WG meeting, the members asked SC&A to research and compile data on the number and exposure potential of workers at these six reactor facilities during relevant time periods. Subsequently, at the September 8, 2016, meeting, the WG asked SC&A to perform additional data compilation for the TRA and TAN areas. SC&A conducted a thorough search of available documents on the Site Research Database (SRDB) for both INL and ANL-W sites. The documents used to compile the available data in this appendix are listed in Section A.8. It should be noted that individual documentation about the workforce at the BORAX-IV experiment in 1958 could not be located and, therefore, is not discussed further in this appendix.

SC&A encountered several different formats of Health and Safety (H&S) reports containing information about the number of individuals and visitors present, the number of regular versus non-routine badges, exposure distributions among the monitored workforce, and average penetrating exposures by time period. SC&A observed that the different formats often contained unclear and/or conflicting information. Additionally, some reports covered time periods that overlapped the badging periods covered in other sources, which obscured the actual worker and dose totals. SC&A used its judgment in interpreting the differing reporting practices to obtain the summary totals displayed in the figures of this appendix and considers this assessment to be a reasonably accurate characterization of total workforce and exposure potential at each reactor site.

In interpreting the reporting practices reflected in the available documentation, it is important to note that for the purposes of this analysis:

- “Regular” refers to individuals who routinely worked and were monitored in a given area.

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- “Visitor” refers to individuals who did not routinely work in an area but were assigned to the area for a given badging period.
- “Other” refers to the category of dosimeters issued to visitors, as well as any non-routine, health physics-specific, and/or incident-driven temporary badges issued per area.

The distinction between “visitor” and “other” is simply that some routine (or “regular”) workers appear to have also been issued non-routine and/or incident badges, which are in addition to the routine worker totals. However, it appears the incidence of such additional badging is likely minimal, and so “other” is considered a fair representation of the visitors entering a reactor area.

A.1. EXPERIMENTAL BREEDER REACTOR I – MARK IV CORE (1962–1963)

Figure A.1-1 plots the number of “regular” individuals reported in available H&S documentation from 1963 to 1965. SC&A was not able to find specific information for EBR-I for 1962. In general, there were approximately 150–200 workers during the period of operation for the MARK IV core, with a spike in September of 1963 in which both regular workers and visitors increased markedly. Although the MARK IV experimentation ended in December 1963, the number of regular workers and visitors was also compiled for 1964 and 1965, as there may have been exposure to residual materials present at EBR-I after the last experiment.

Figure A.1-1. Number of Regular Individuals and Visitors to the EBR-I Area: 1963–1965

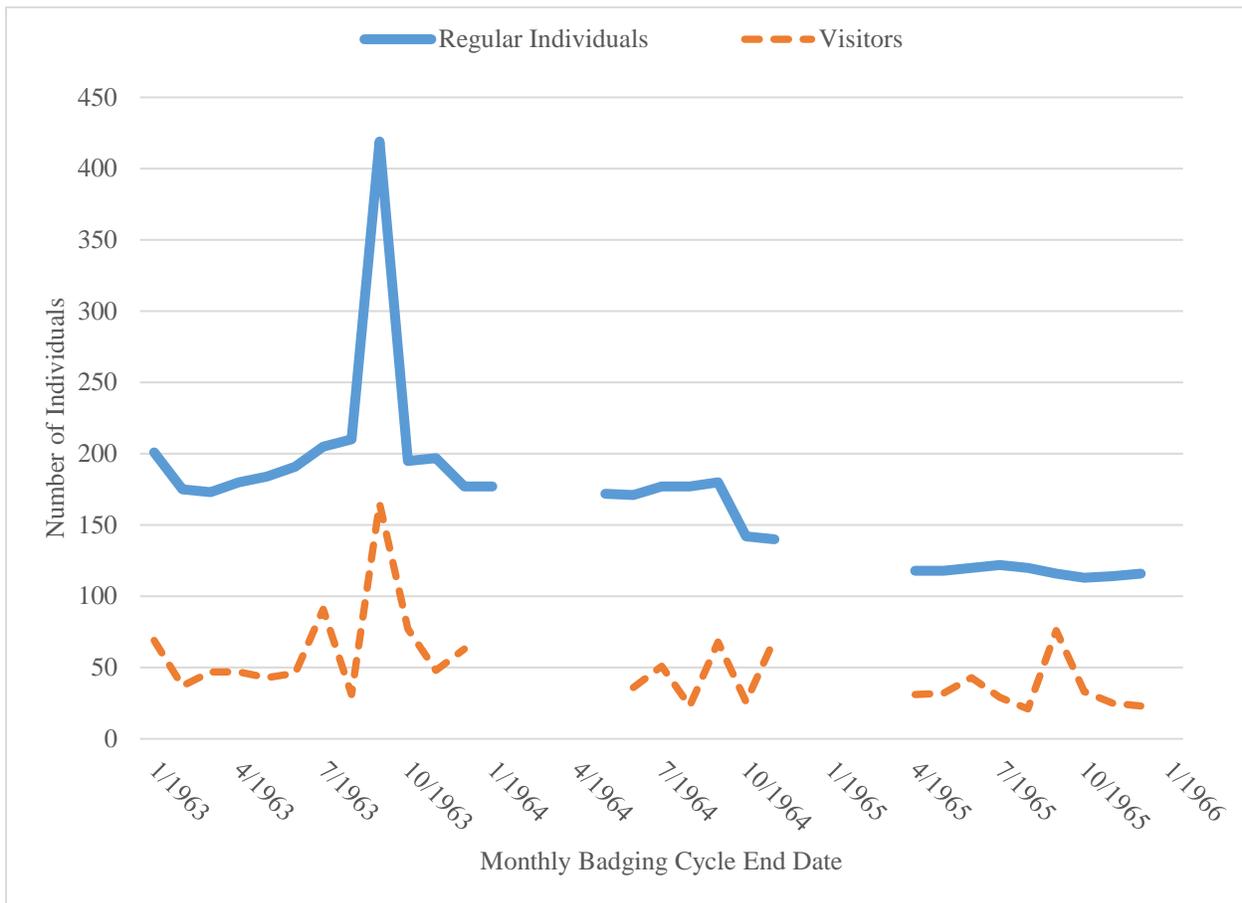
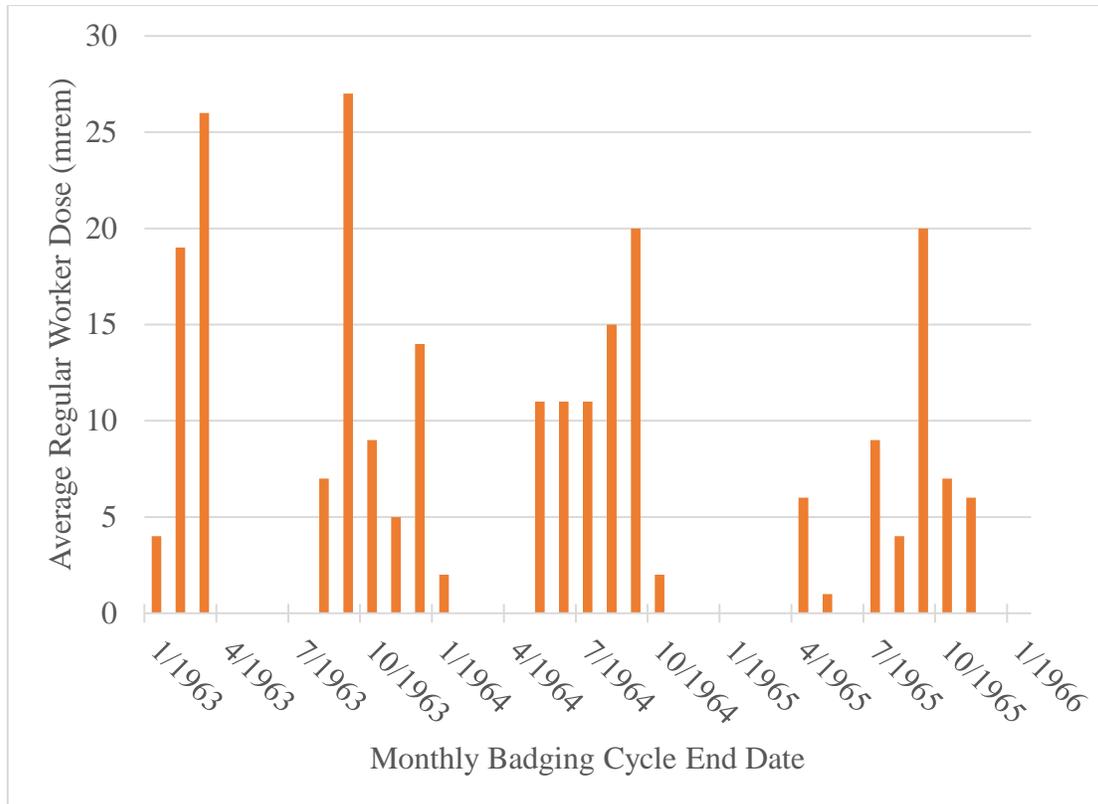


Figure A.1-2 plots the average penetrating dose to regular workers at EBR-I from 1963 to 1965. For the majority of badging periods in which the average penetrating dose was reported (17 of 22 periods), the average worker dose at EBR-I was 15 millirem (mrem) or less. The highest average worker doses were observed in March and September 1963 (26 and 27 mrem, respectively). It is worth noting that a spike in the total number of regular workers and visitors was observed in September 1963. No reports of the maximum external exposure to specific individuals per badging period was available in EBR-I documentation.

Figure A.1-2. Average Penetrating Dose to Regular Workers at EBR-I: 1963–1965



A.2. EXPERIMENTAL BREEDER REACTOR II (1961–1994)

Figure A.2-1 shows the number of routine workers at EBR-II from 1963 through 1975.¹⁰ From 1963 until 1966, the regular work force at EBR-II was in the range of about 400 workers. For the remaining badging periods with records after 1966, documentation generally reported regular workers in the 200-to-300 range for two out of every three months. Interestingly, the number of regular workers reported for EBR-II approximately doubled every third month beginning in 1967. The apparent gap from mid-1973 to mid-1974 is not the result of a lack of records. During this gap, the available records had “cut off” the listing of actual work locations in the scanned electronic versions of the hardcopy file.

Figure A.2-2 plots the number of reported visitors to EBR-II from 1963 to 1972. Unlike the regular worker totals presented in Figure A.2-1, the available badging reports did not indicate the

¹⁰ SC&A was unable to locate additional information for EBR-II after July 1975.

number of visitors after June 1972. It is apparent that visitor totals could vary significantly depending on the timeframe. Totals ranged from a low of 32 visitors in May 1965 to a high of 691 visitors in June 1968. In general, the number of visitors increased somewhat during 1968 and 1969.

Figure A.2-1. Number of Regular Individuals at the EBR-II Area: 1963–1975

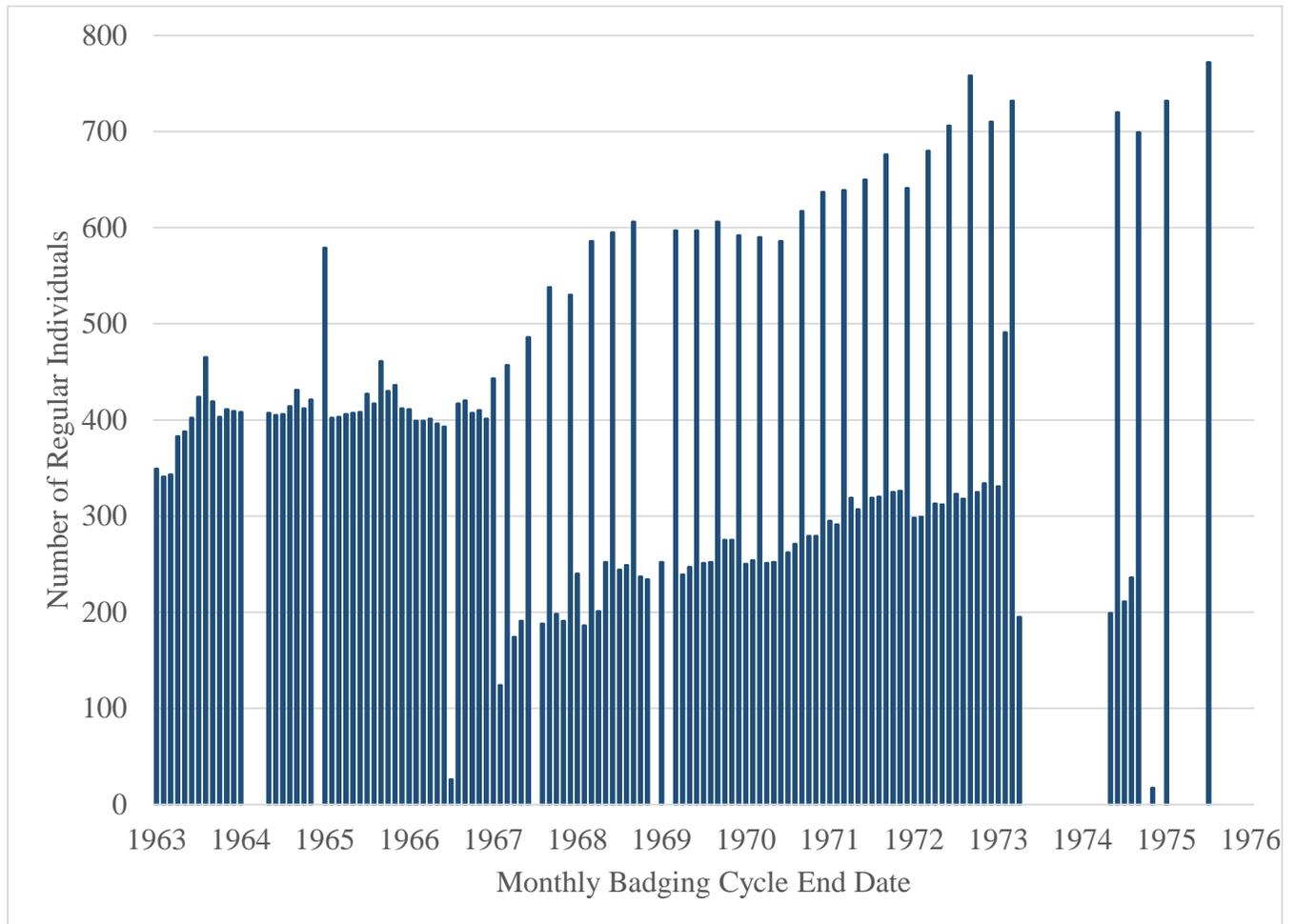


Figure A.2-2. Number of Visitors to the EBR-II Area: 1963–1972

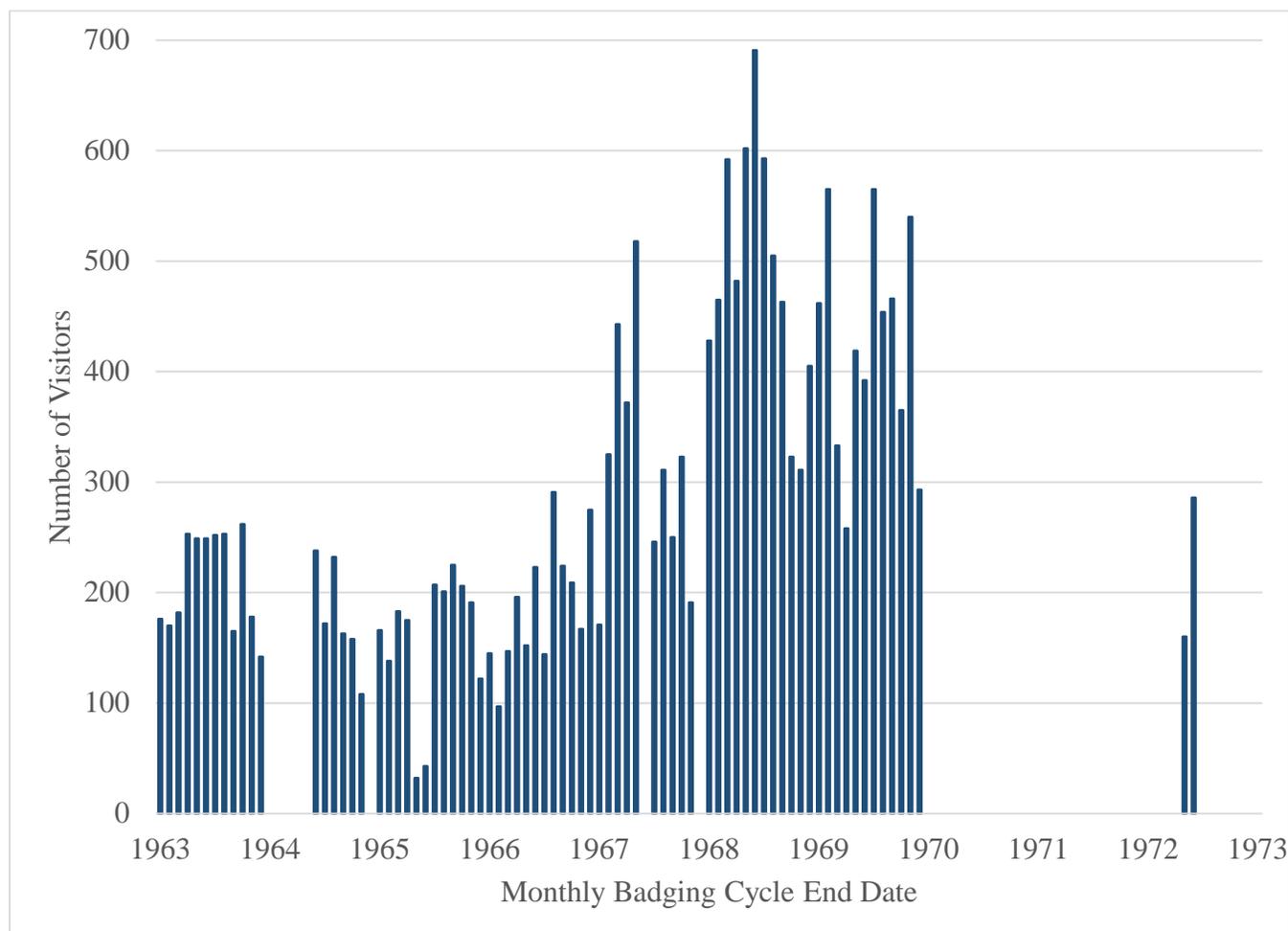


Figure A.2-3 displays the average worker exposure per monthly badging cycle from 1963 to 1971. As a general trend, it appears that average worker doses increased somewhat during this timeframe. The highest observed average penetrating dose was approximately 122 mrem in February 1967, with a similarly high value observed in November 1969. The average over the entire period was approximately 25 mrem per worker per monthly badging period.

Monthly badging reports often provided information on how many regular workers fell into certain dose ranges (0–299, 300–1,199, and 1,200+ mrem). SC&A did not observe any reported workers falling into the highest dose category (1,200+ mrem); however, some workers did fall into the intermediate category for some months. Figure A.2-4 displays the percentage of regular workers who had monthly penetrating doses that were between 300 and 1,199 mrem. The highest observed percentage of workers falling into this dose range was approximately 13% in November 1969. Not surprisingly, many of the “spikes” observed in in Figure A.2-4 correspond to the higher monthly averages shown in Figure A.2-3. Figure A.2-5 overlays the average worker doses from Figure A.2-3 with the percentage of workers receiving doses greater than 300 mrem in Figure A.2-4. As expected, the two figures show significant temporal agreement.

Figure A.2-3. Average Worker Penetrating Dose in the EBR-II Area per Monthly Badging Cycle: 1963–1971

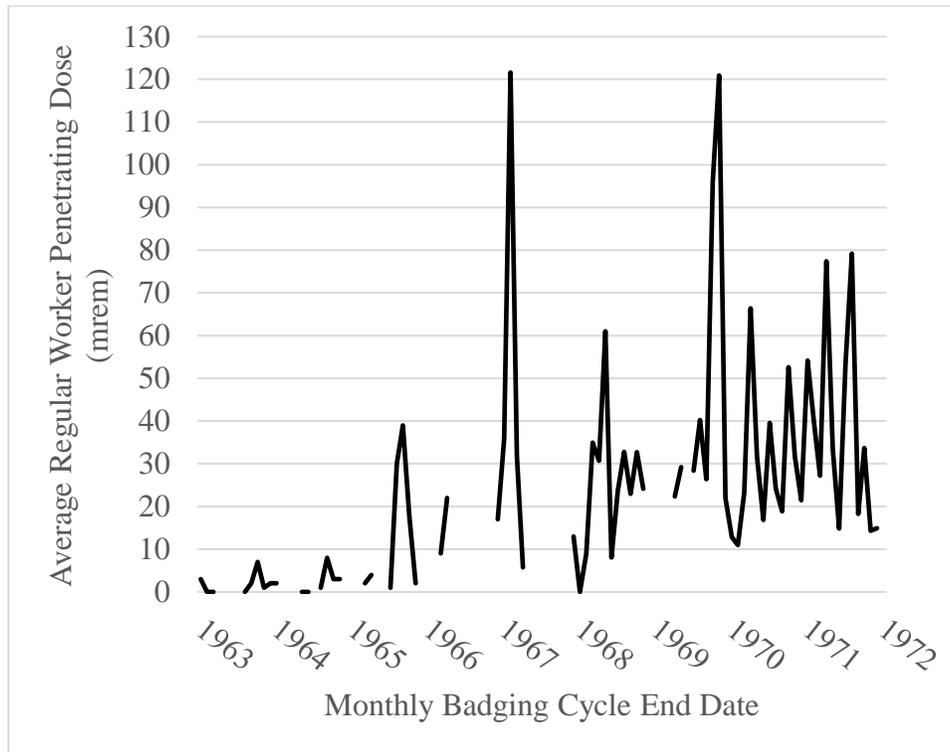


Figure A.2-4. Percentage of Workers at EBR-II Exceeding 300 mrem per Monthly Badging Cycle: 1963–1971

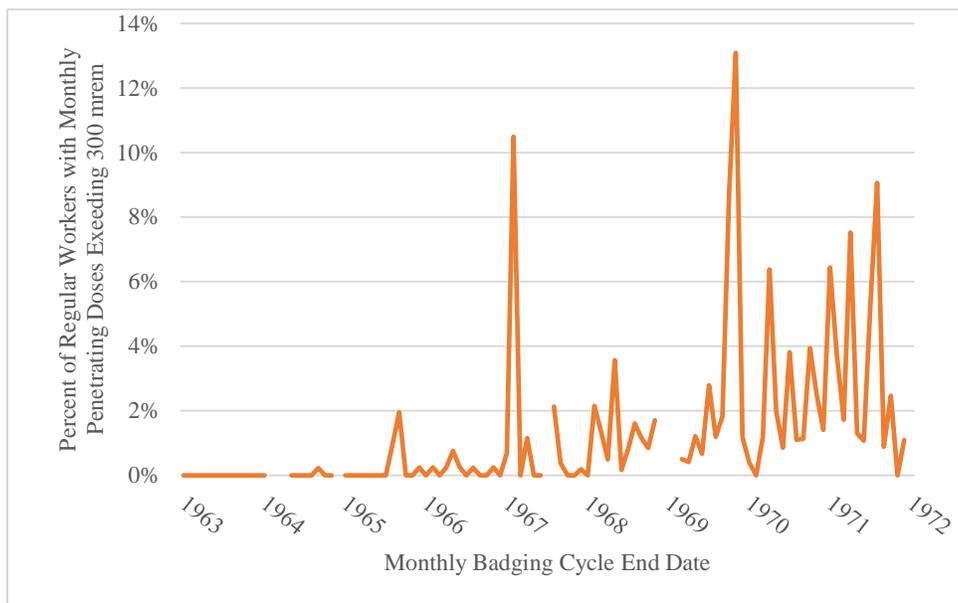
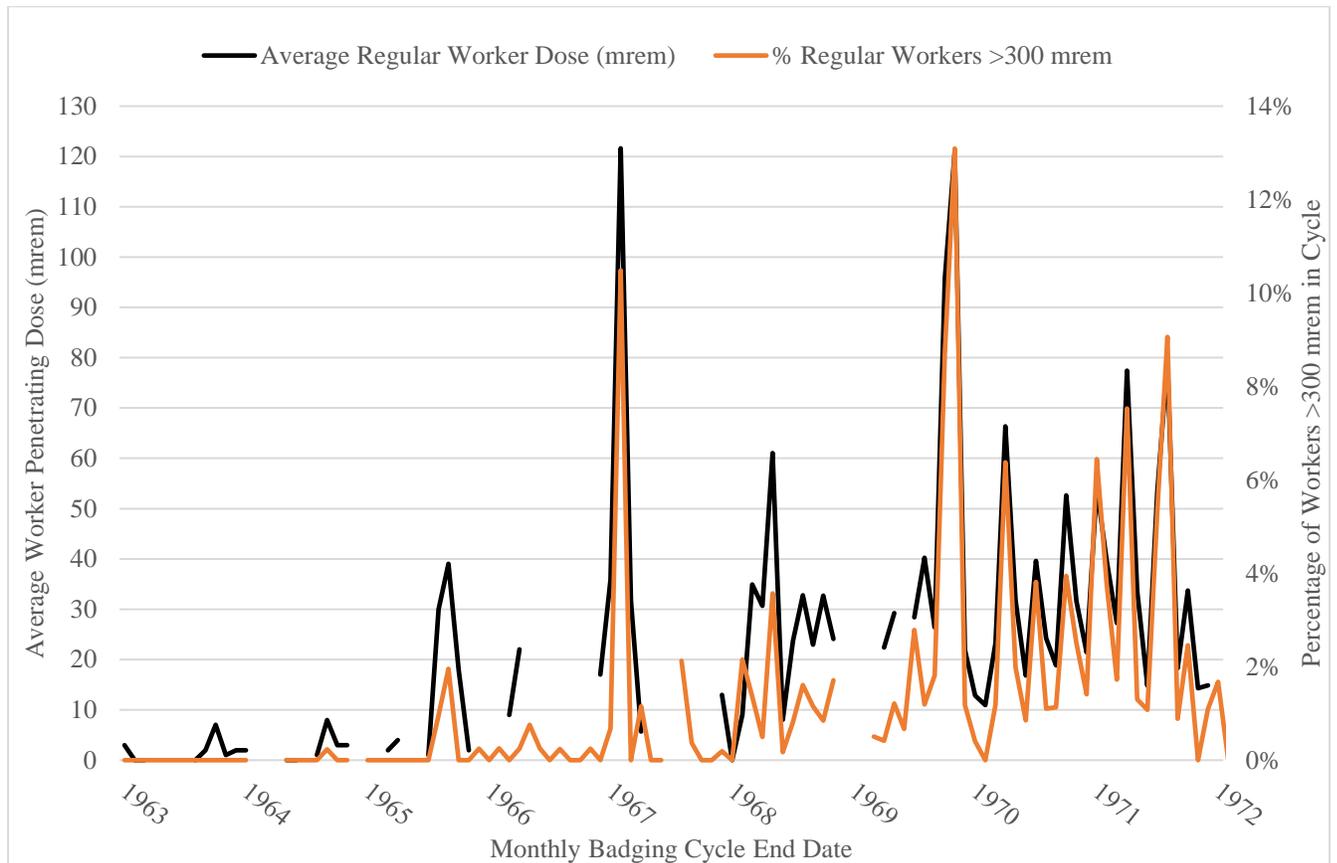


Figure A.2-5. Overlay of Figures A.2-3 and A.2-4 Demonstrating the Correlation between Average Observed Penetrating Doses and the Number of Workers with Doses Greater than 300 mrem



A.3. POWER BURST FACILITY (1972–1985)

SC&A could not locate any badging reports after 1975, so the analysis of regular workforce and exposure potential was limited to the first few years of operation. In contrast to EBR-I and EBR-II, the PBF had significantly fewer regular workers during the early years of its operation. As seen in Figure A.3-1, the number of regular workers assigned to PBF during these years was between 25 and 30 for most periods for which records were available. The maximum observed number of workers during this period was 77 in January 1975. It is worth noting that the large observed gap from May 1973 to April 1974 was also observed for EBR-II (see Figure A.1-1) and was the result of the specific area designation being cut off in scanned version of the badging reports for those months (i.e., an artifact of the record-keeping system, rather than an actual occurrence).

Interestingly, the number of visitors often outnumbered the number of regular workers, as shown in Figure A.3-2. The greatest number of visitors at PBF was 140 in February 1973. Note that the absence of a visitor total in Figure A.3-2 is not indicative of there being no visitors to PBF for a particular badging cycle, but rather the lack of available records indicating the actual visitor totals.

Figure A.3-1. Number of Regular Individuals at PBF per Monthly Badging Cycle

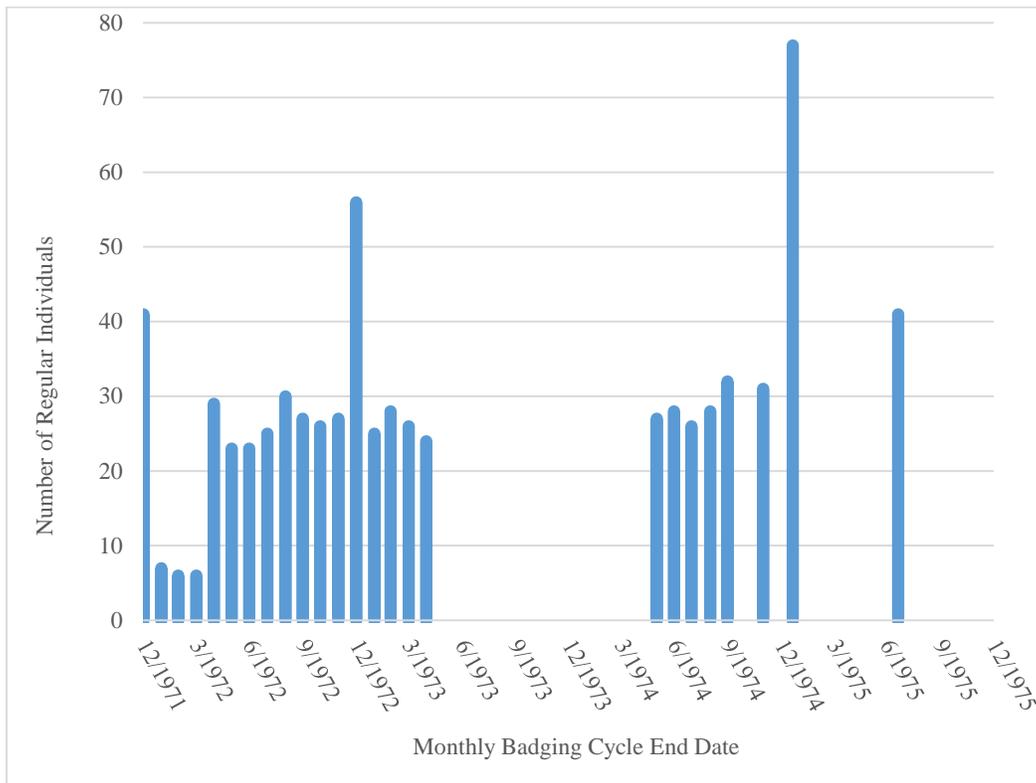
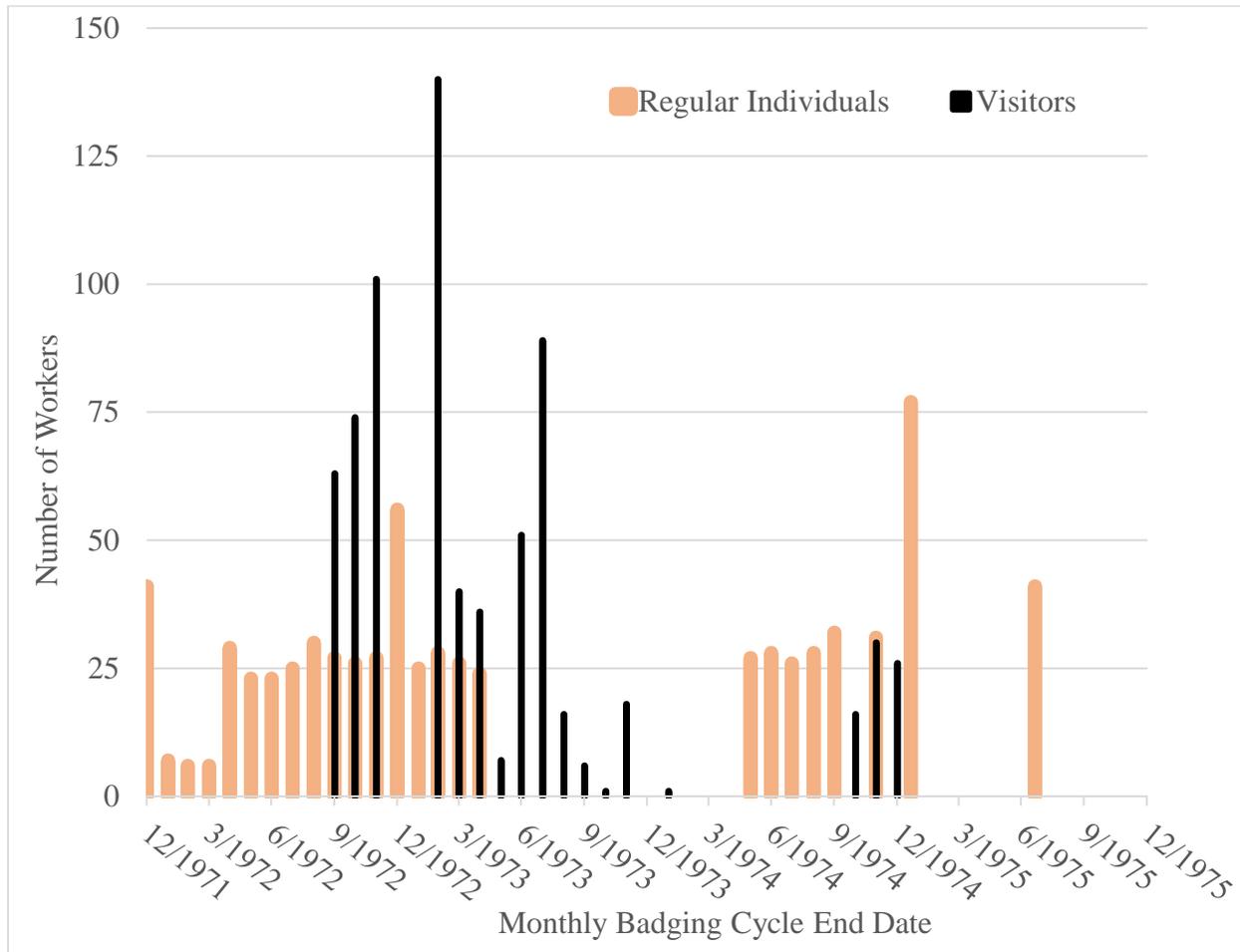
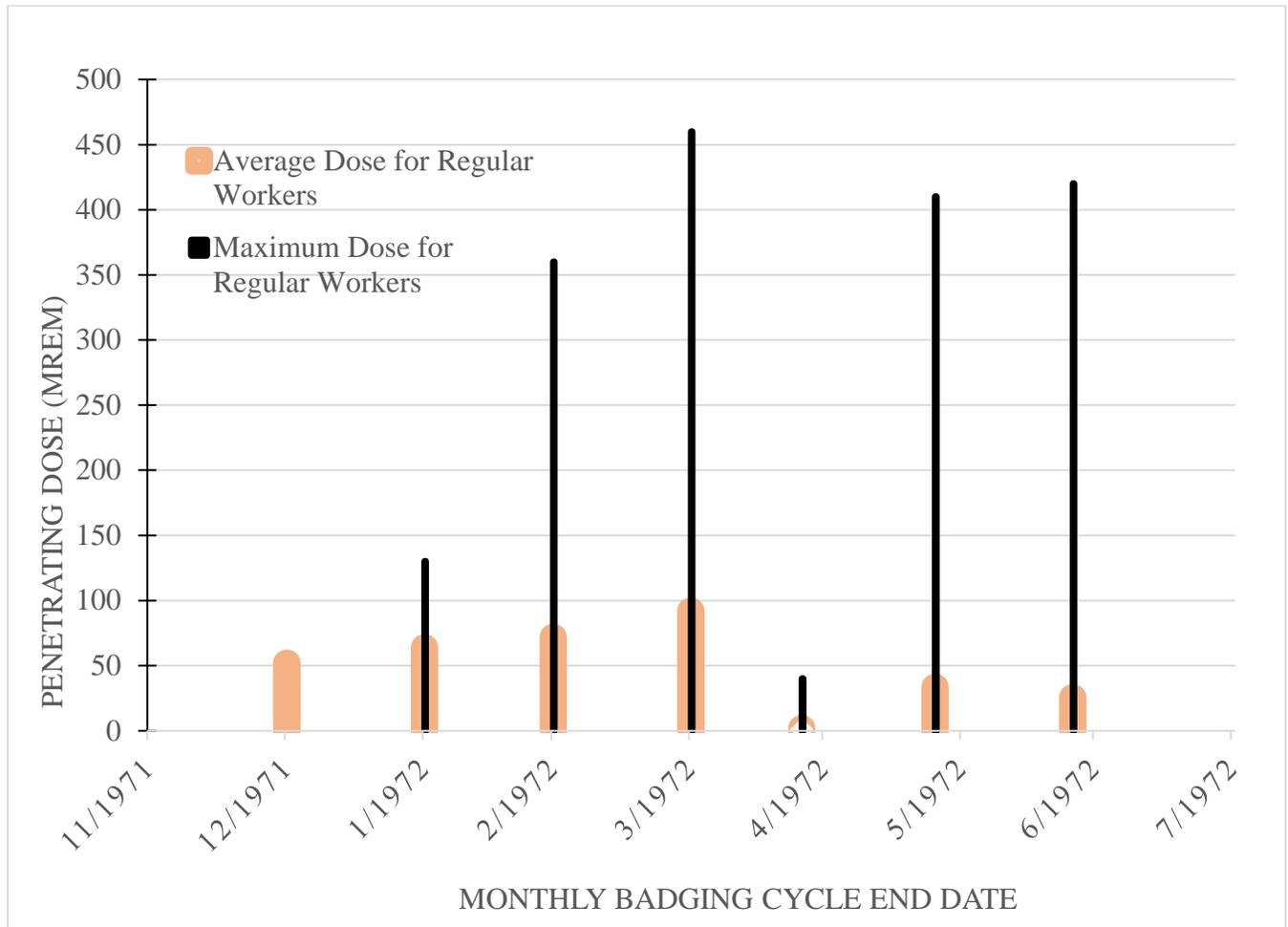


Figure A.3-2. Comparison of Regular Worker Totals to Visitors at PBF per Monthly Badging Cycle



Only 7 of the 46 badging cycle entries identified among available documentation also provided the maximum and/or average penetrating dose to regular workers at PBF. Nonetheless, Figure A.3-3 plots these average and maximum penetrating doses by monthly badging period. Average worker doses ranged from 2 to 92 mrem, while the maximum observed monthly dose was 460 mrem.

Figure A.3-3. Comparison of Average Doses to Maximum Doses for Regular Workers at PBF



A.4. SPECIAL POWER EXCURSION REACTOR TESTS (1958–1970)

Available records for SPERT do not indicate a total number of regular workers or visitors at the area until 1963. However, the total number of regular (or routine) badges versus non-routine badges issued is a reliable indicator of the approximate number of workers potentially exposed at SPERT during a given badging cycle. Figure A.4-1 plots the number of regular and non-routine badges reported for SPERT from 1958 to 1962.

Beginning in 1963, the H&S reports provided the actual number of regular workers versus visitors. These totals are presented in Figure A.4-2. As seen in the figure, the number of regular workers generally decreased from 1963 through 1966, at which point the number fluctuates between 20–30 workers for some periods and over 150 workers for others. The number of visitors also shows similar fluctuations during the period after 1966, ranging from around 100 visitors to over 300.

Figure A.4-1. Number of Routine and Non-Routine Dosimeters Issued for SPERT: 1958–1962

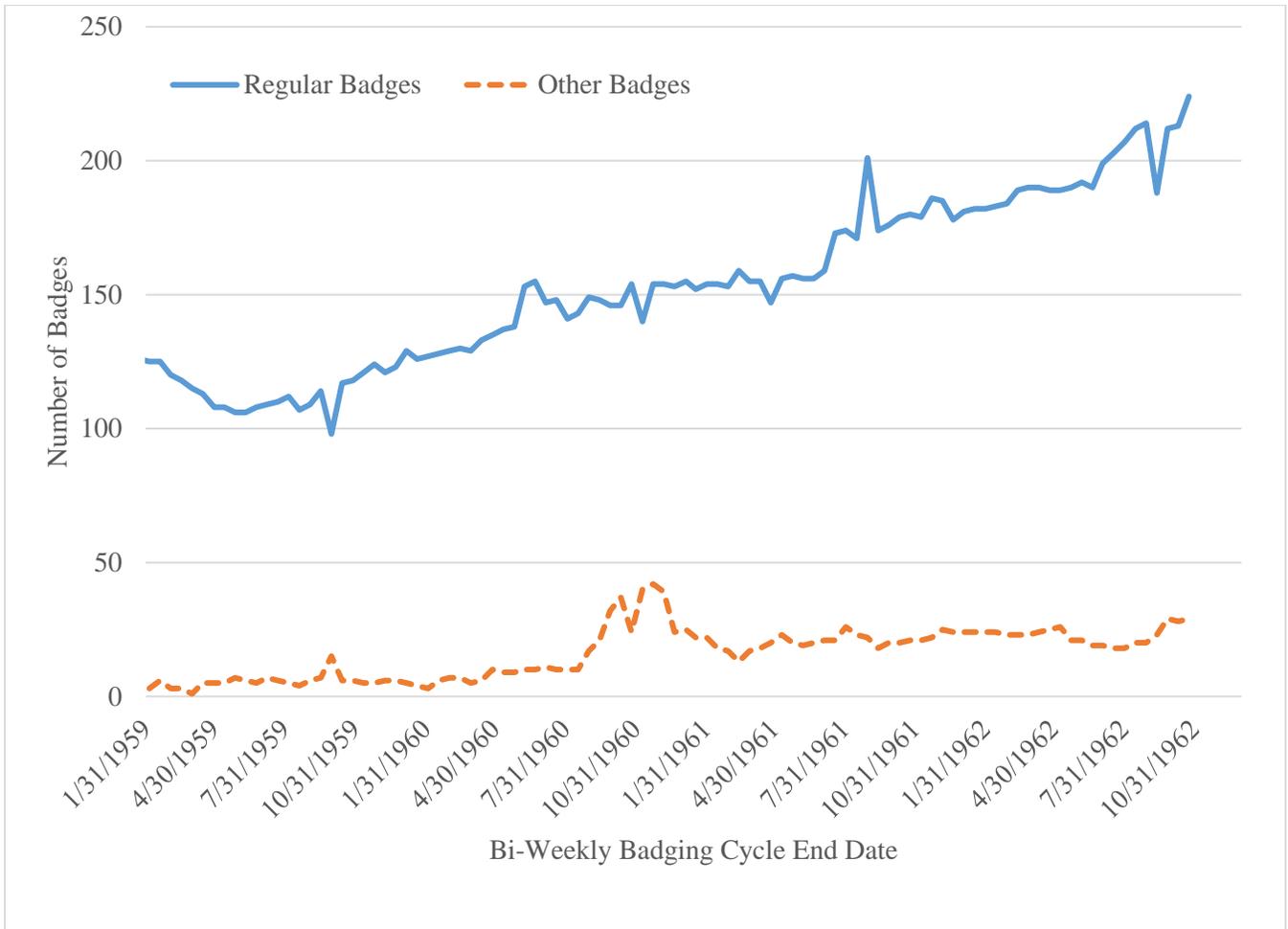


Figure A.4-2. Number of Regular Workers and Visitors at SPERT by Monthly Badging Period

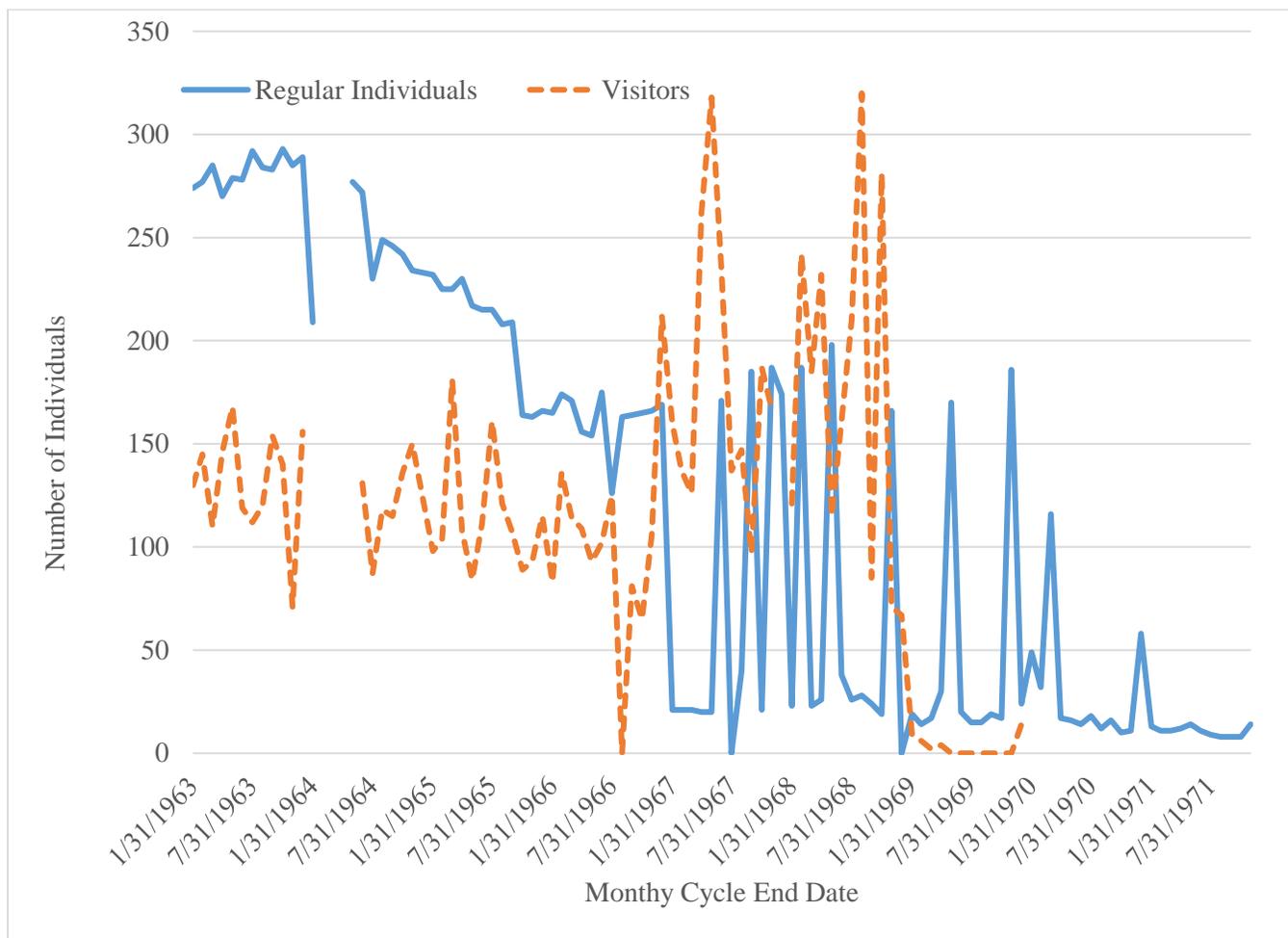


Figure A.4-3 shows the maximum reported dose by monitoring period for workers at SPERT from 1958 through 1971. It should be noted that until 1962, SPERT workers were on a biweekly badging exchange schedule, after which they switched to a monthly schedule. As shown in the chart, one uncharacteristically high value was observed in the beginning of 1959 (2,120 mrem). Inspection of H&S reports subsequent to this badging period indicates that this exposure was likely related to exposure incurred by the worker while at the Chemical Processing Plant (CPP).

Figure A.4-4 plots the maximum reported penetrating dose without the outlier from CPP included. As shown, the maximum exposures to SPERT workers varied significantly by badging period. Many of the biweekly badging results ranged from 100 to 200 mrem, with spikes exceeding 400 mrem. After 1963, the dose totals generally increased per badging period, which is to be expected when switching from biweekly to monthly badging periods. The greatest observed penetrating dose during the biweekly period was 940 mrem in 1960, while during the monthly badging periods, the highest observed dose was 820 mrem in 1971.

Figure A.4-3. Maximum Penetrating Dose by Badging Cycle for Regular Workers at SPERT (All Data)

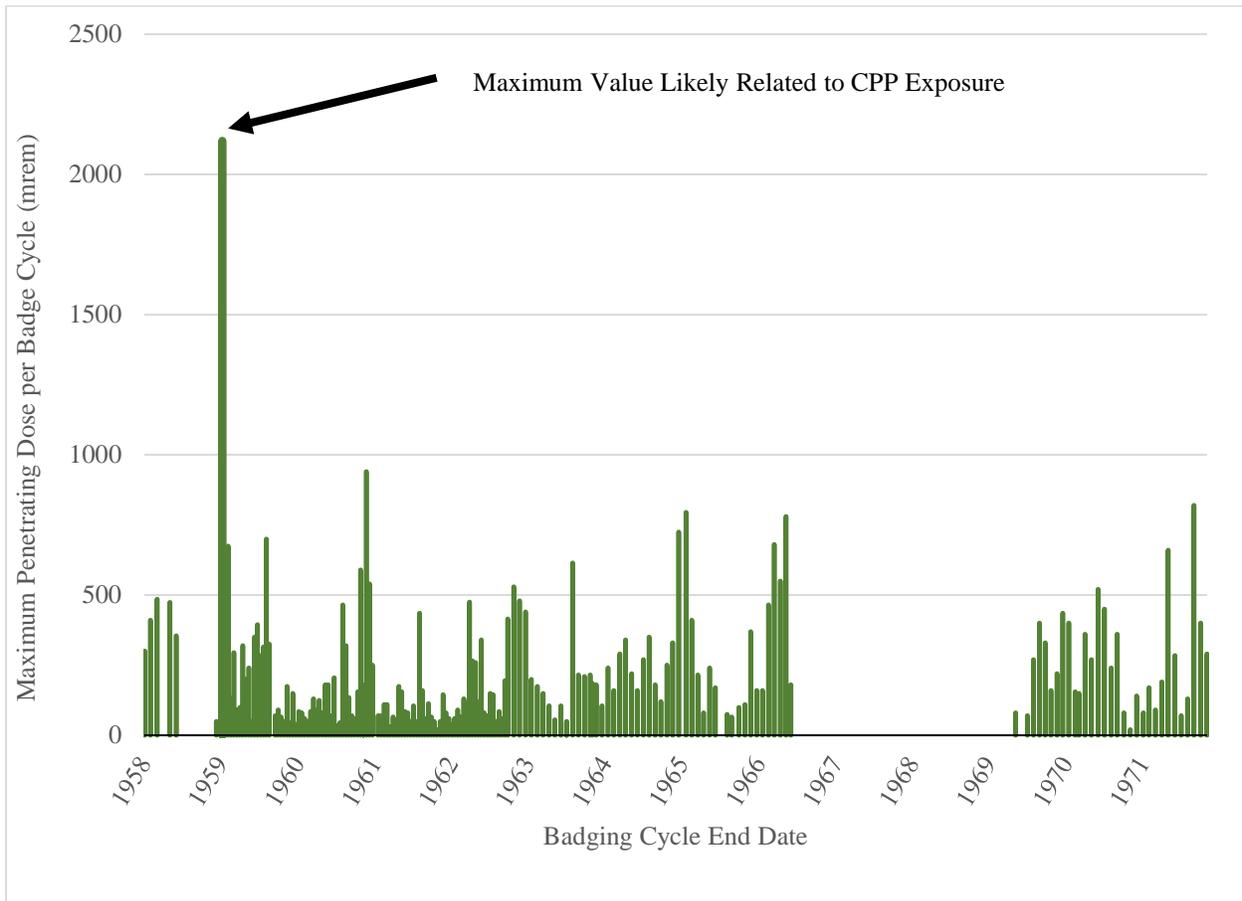
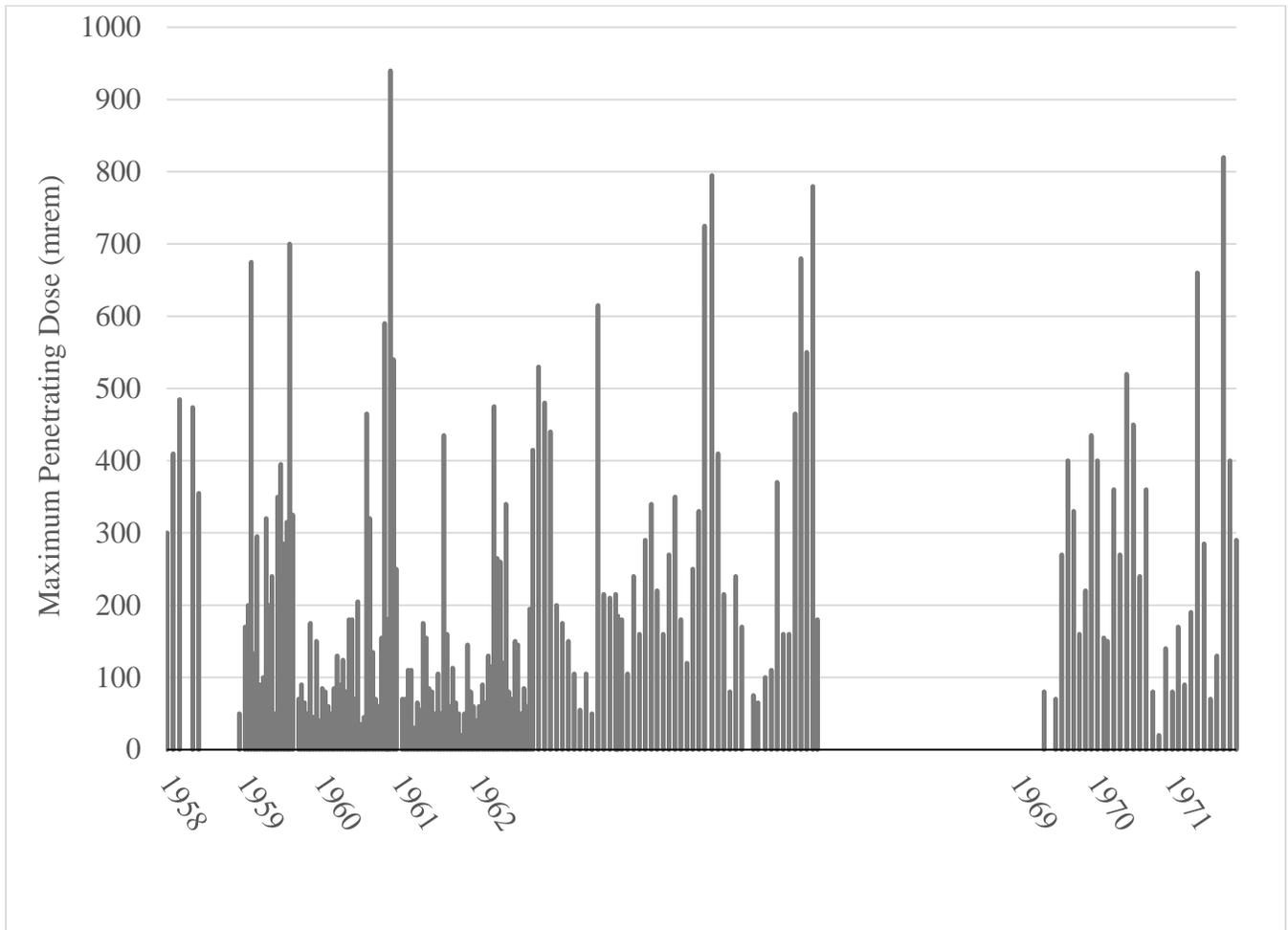
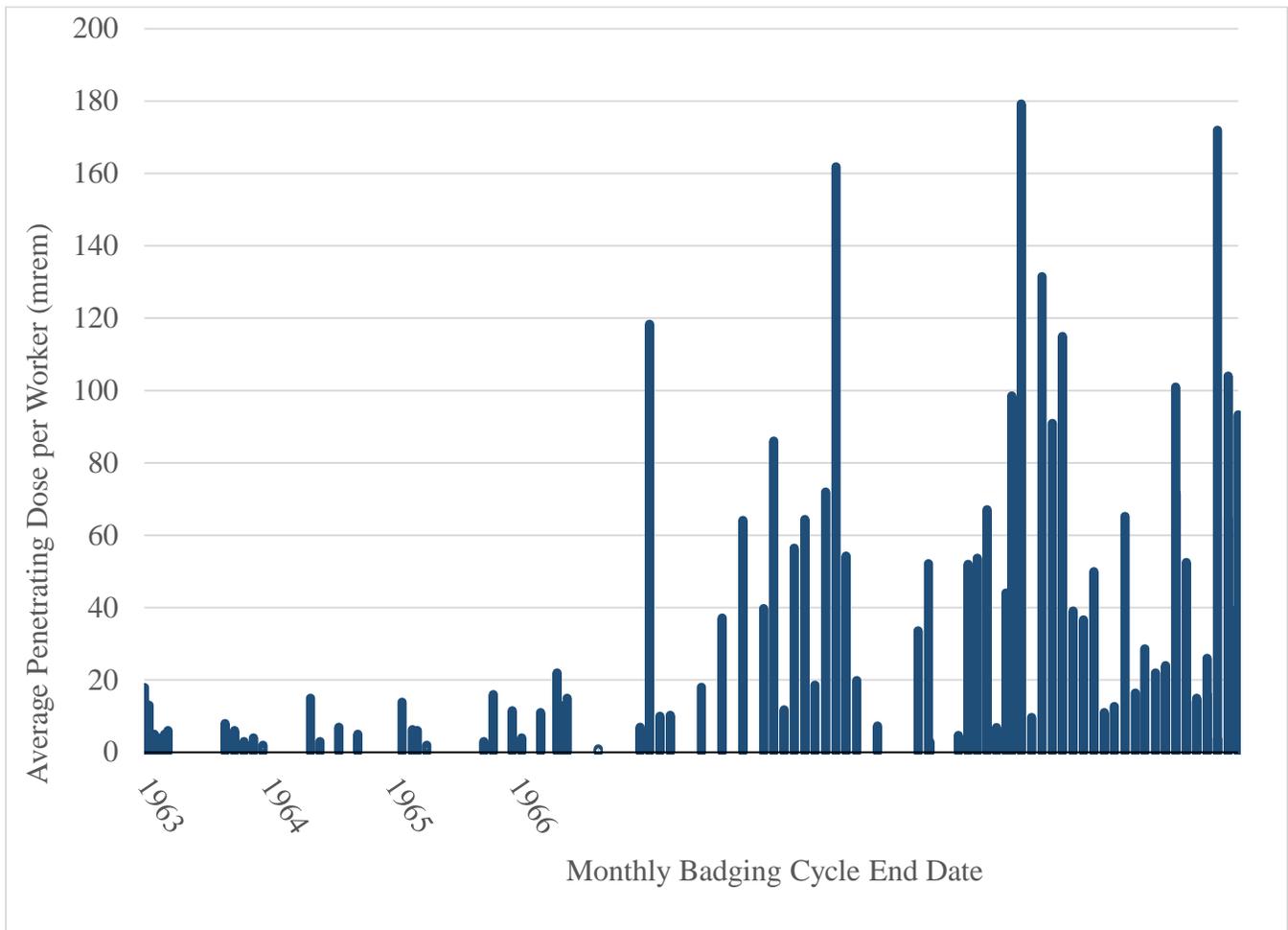


Figure A.4-4. Maximum Penetrating Dose by Badging Cycle for Regular Workers at SPERT (Outlier Removed)



Like the reporting of the number of individuals and visitors in available H&S reports, average worker doses were not reported until 1963, when SPERT appears to have switched to a monthly dosimeter exchange cycle. Figure A.4-5 plots the average penetrating dose to workers from 1963 to 1971. Until 1967, the average dose to regular workers was generally less than 20 mrem. After this time, average doses generally increased, with marked spikes in August 1968 (162 mrem), February 1970 (180 mrem), and September 1971 (172 mrem).

Figure A.4-5. Average Regular Worker Dose per Monthly Badging Cycle: 1963–1971



A.5. ORGANIC MODERATED REACTOR EXPERIMENT (1958–1963)

Documentation concerning the workforce and exposure potential specific to the OMRE project was only located for 1963 (the last year of operation). The total number of regular workers, as well as the number of visitors to the OMRE area, are shown in Figure A.5-1. Because data were only available for the last year of experimental operation, it is not surprising that the number of regular workers decreased consistently from the start of 1963 through October 1963. OMRE appeared on H&S reports after October 1963; however, these reports simply listed the number of regular workers and visitors as zero.

Figure A.5-1. Number of Regular Workers and Visitors to the OMRE Area by Monthly Badging Cycle in 1963

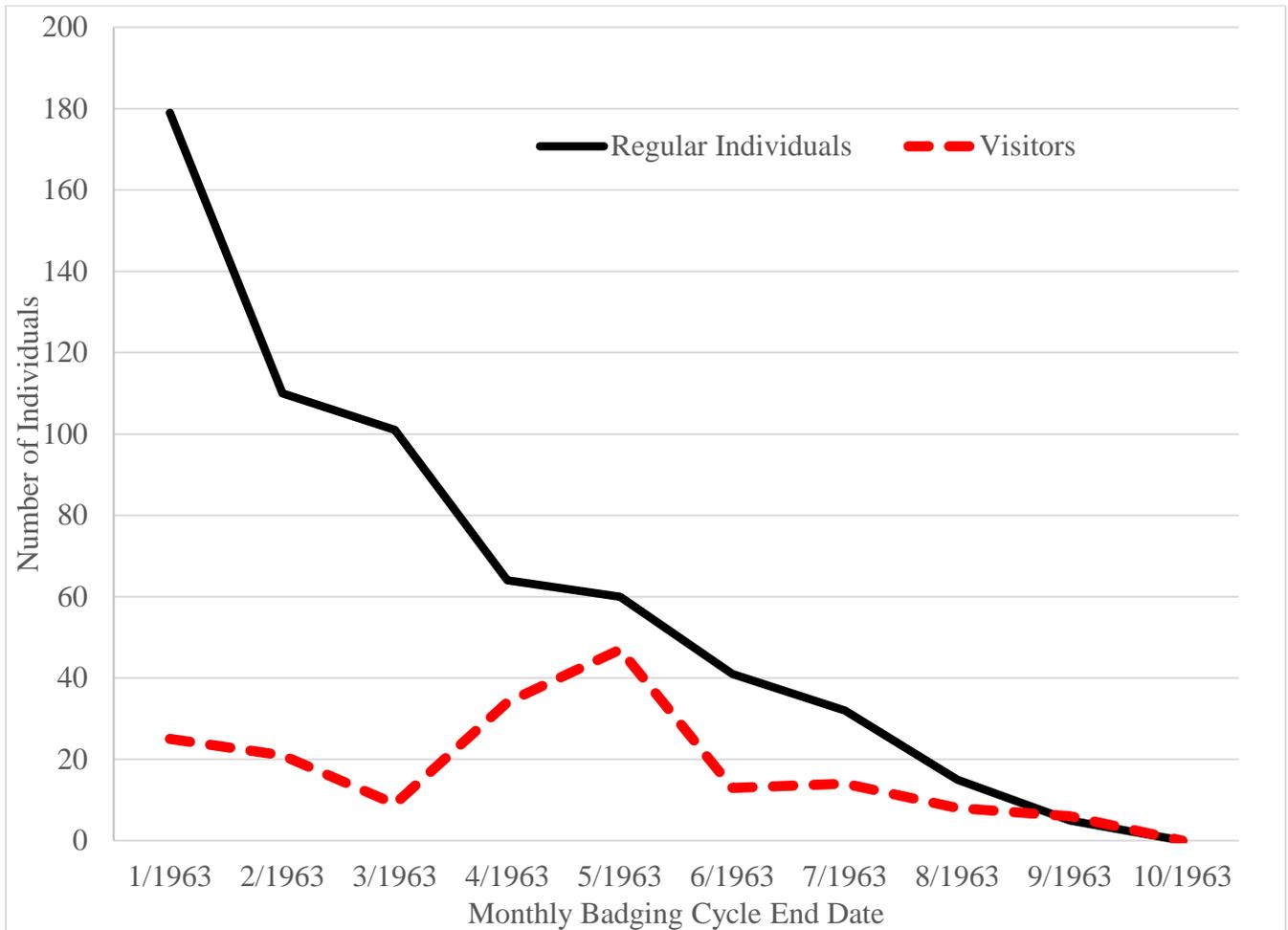
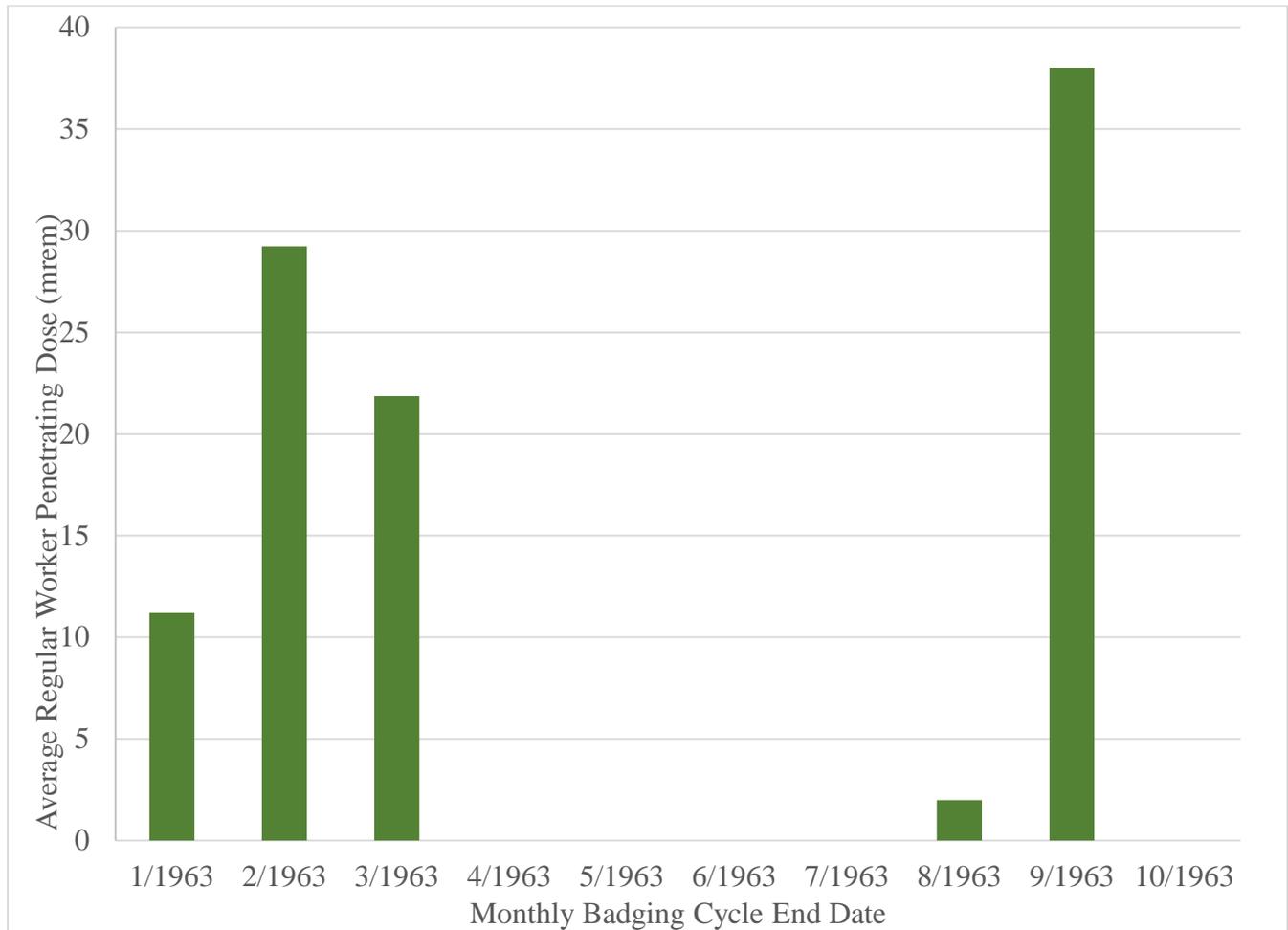


Figure A.2-2 plots the average doses to regular workers by monthly badging cycle. The data are limited to just five monthly badging cycles in 1963 and ranged from an average penetrating dose of 2 mrem to 38 mrem. Interestingly, the highest observed average value occurred in the last month for which the H&S department reported workers for OMRE (five regular workers and six visitors).

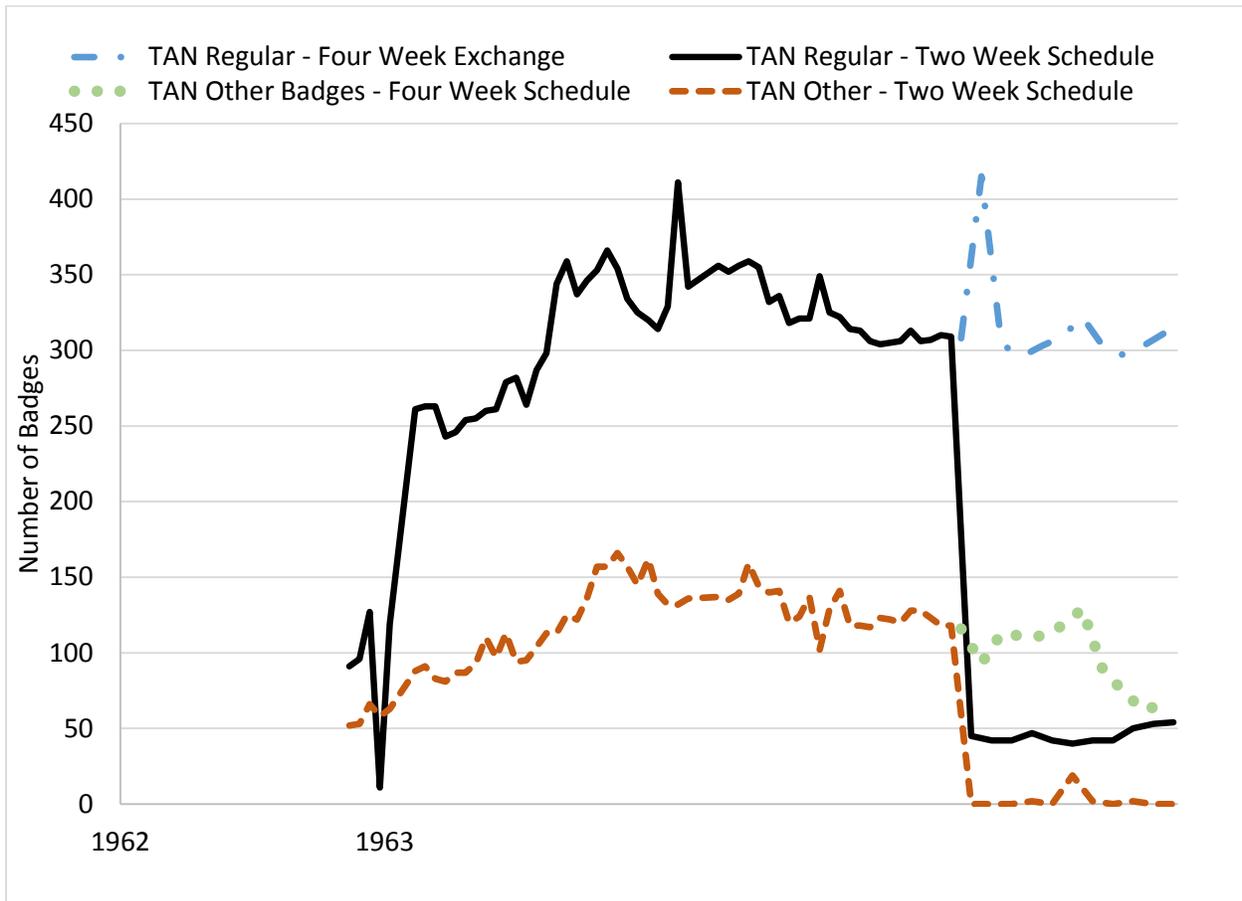
Figure A.5-2. Average Penetrating Dose to Regular Workers at the OMRE Site per Monthly Badging Cycle in 1963



A.6. TEST AREA NORTH REACTOR AREA

As noted in Section 4.1, SC&A was unable to locate specific information about the HTRE during its operational period (approximately 1955–1961). Nonetheless, SC&A compiled available data for the TAN area beginning in subsequent years when records became available. Figure A.6-1 plots the number of regular and “other” badges for the TAN area beginning in November 1962. As seen in the figure, the number of badges was relatively low for the first few badging cycles available before increasing to approximately 300–400 regular badges per cycle. It should be noted that all badges were issued on a 2-week exchange schedule until late February 1965.

Figure A.6-1. Number of Regular and “Other” Badges Issued at TAN Beginning in November 1962 for Workers on 2- and 4-Week Exchange Schedules



In addition to statistics concerning the TAN area as a whole, H&S reports also contained badging totals for the Initial Engine Test (IET) area of TAN where the HTREs were located. These totals are shown in Figure A.6-2. It should be noted that while the HTREs were located at the IET area, data were only available starting in the fourth quarter of 1963, when one of the main activities at IET was the Systems for Nuclear Auxiliary Power (SNAP) program. It is not clear to what extent exposure potential would still have existed for material from the HTRE program.

Figure A.6-2. Number of Regular and “Other” Badges Issued at IET Beginning in October 1963 for Workers on 2- and 4-Week Exchange Schedules

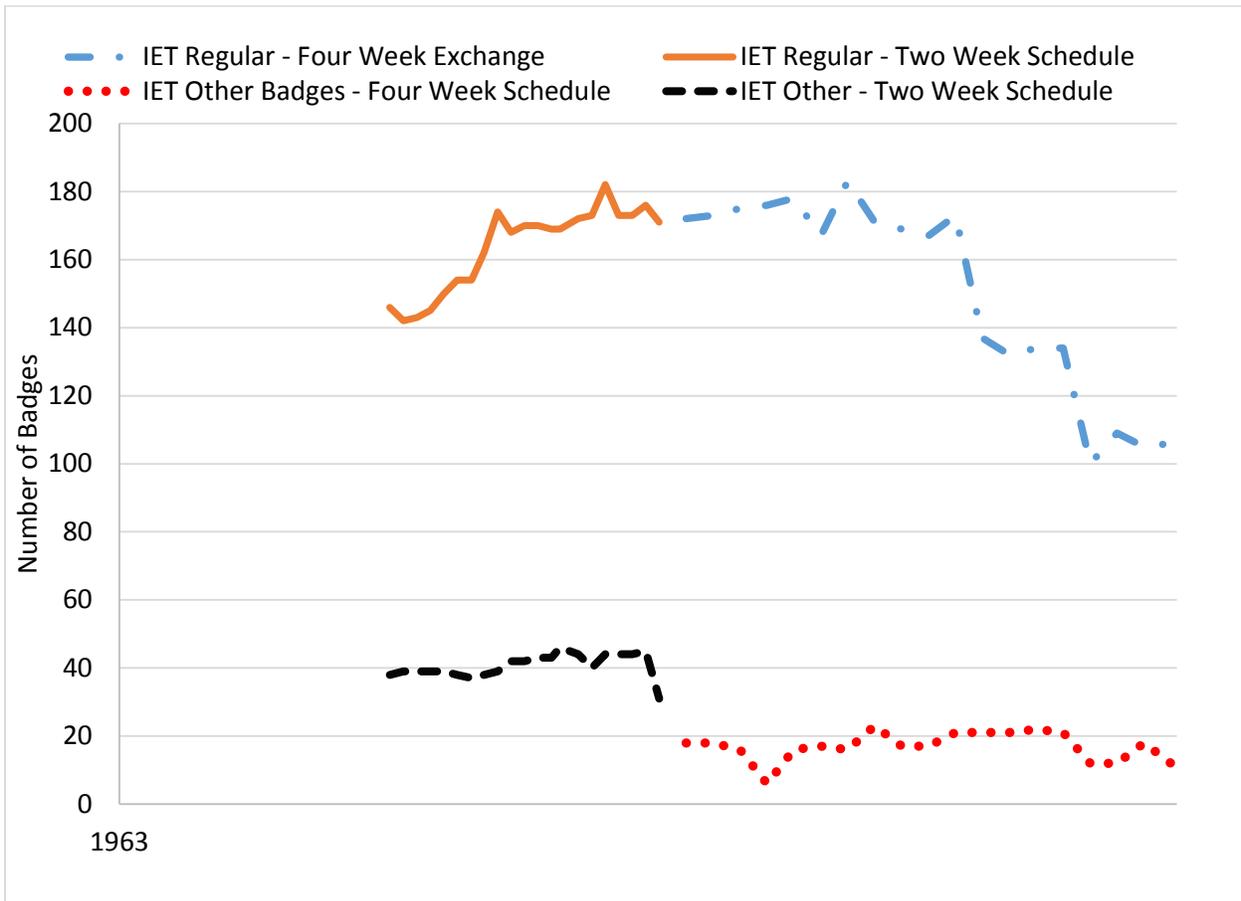
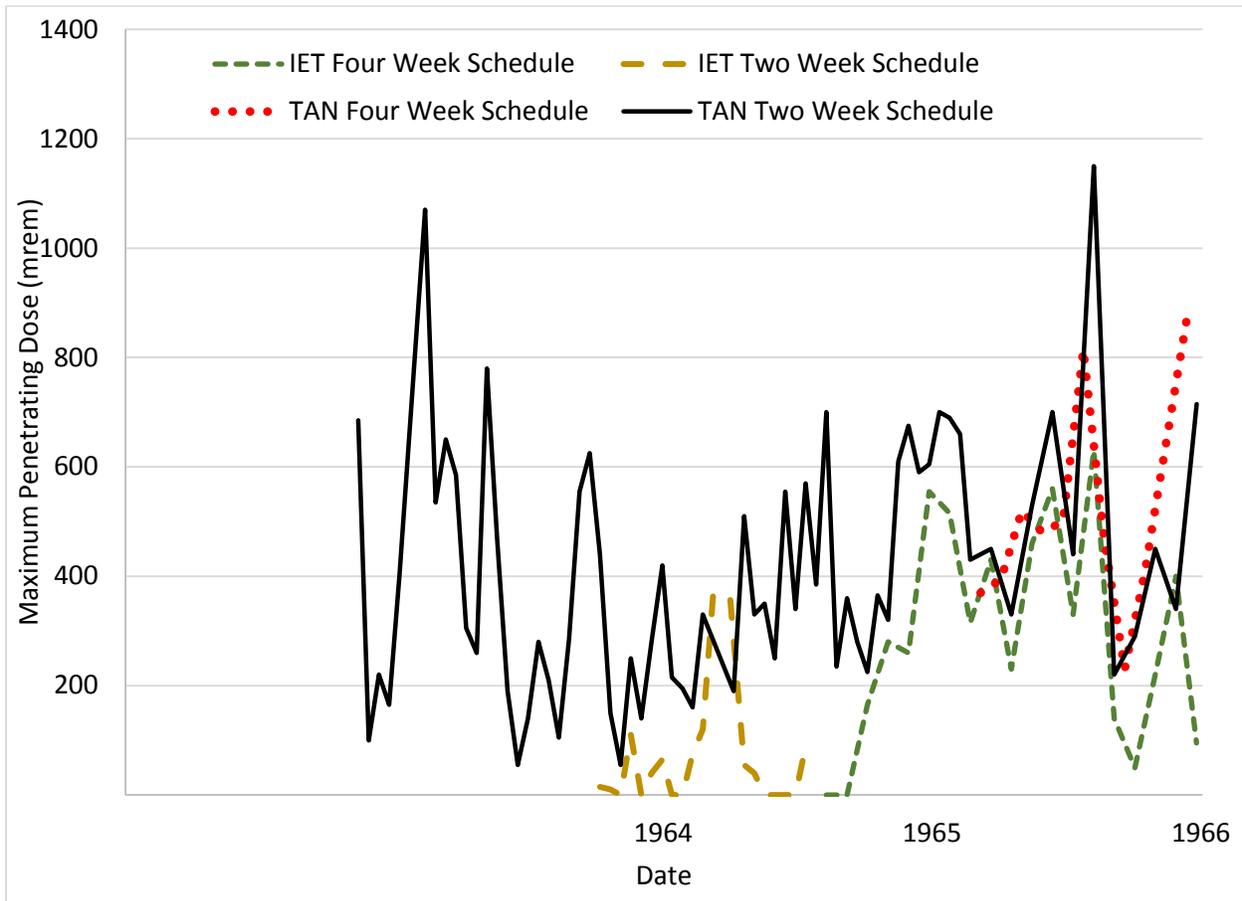


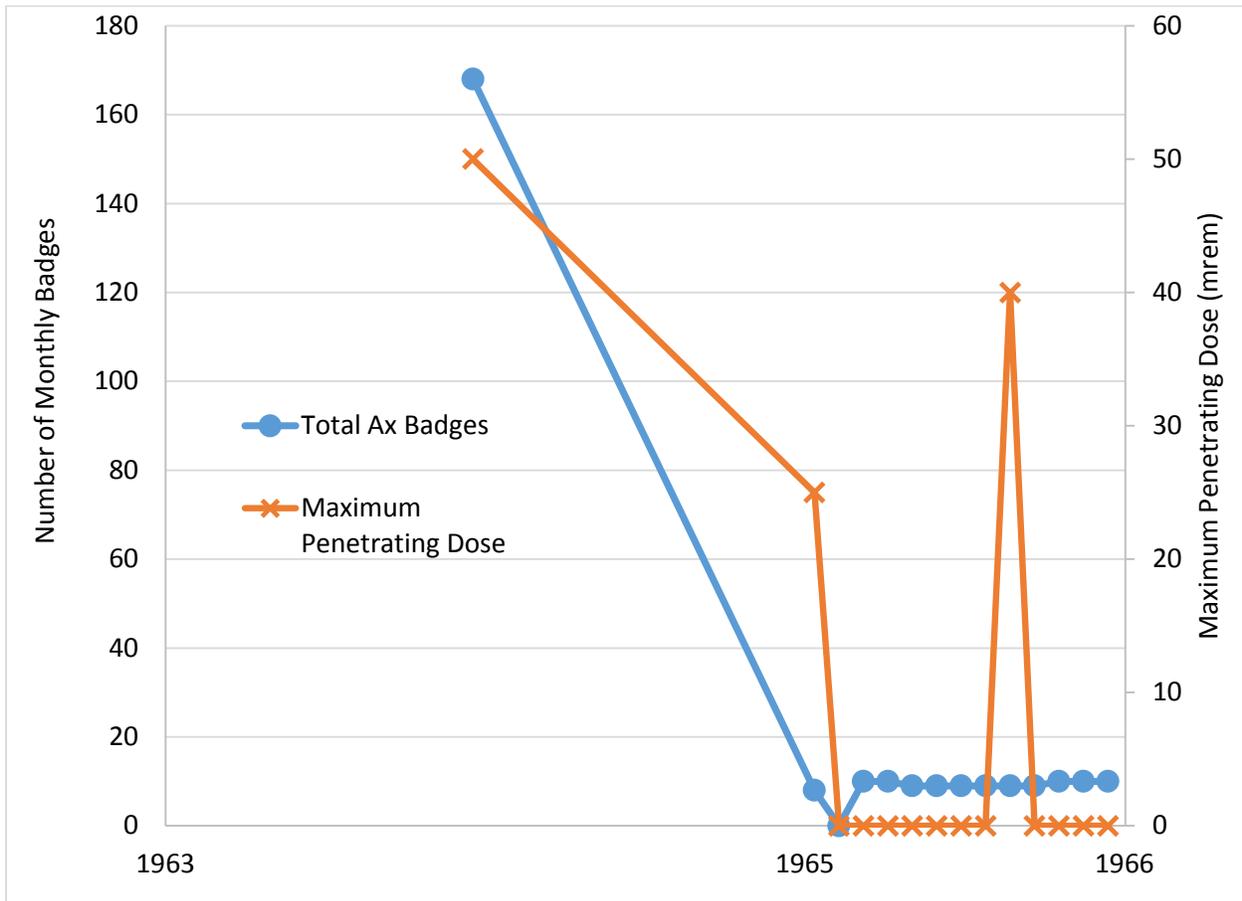
Figure A.6-3 plots the maximum penetrating dose observed for any given badging period at both IET and TAN as a whole. As the figure shows, the maximum penetrating dose at TAN generally bounded the dose at IET. The highest observed penetrating dose was 1.15 rem during a 2-week badging period at TAN in August 1965. In general, the maximum 2-week penetrating exposure was greater than 200 mrem at TAN.

Figure A.6-3. Maximum Penetrating Dose (mrem) per Badging Cycle at TAN and IET for Workers on 2- and 4-Week Exchange Schedules



In addition to regular and “other” badges at TAN and IET, a separate category for construction work at TAN was used (dosimetry code Ax). The total number of Ax badges issued and the maximum penetrating dose are shown in Figure A.6-4. As seen in the figure, data concerning construction badging at TAN were sparse. For most badging cycles, only 10 or fewer badges were issued for TAN construction. The sole badging cycle higher than these totals was 168 badges in December 1963. In addition, the exposure potential for TAN construction activities appears consistently low (often zero mrem), with only two data points that were non-zero. The highest penetrating dose was 50 mrem, which coincided with the highest number of badges observed.

Figure A.6-4. Total Number of Ax Badges and Maximum Penetrating Dose (mrem) per Badging Cycle at TAN for Construction



A.7. TEST REACTOR AREA (1956–1981)

Figure A.7-1 displays available records tabulating the number of badges issued in TRA beginning in October 1956. In general, 1,000 regular badges were issued per badging period from 1957 to 1962. After this time, there were spikes in the number of regular badges issued, which ranged as high as 4,000 badges per period. The number of “other badges” generally mirrored the trend seen in the regular badges up through 1966. No data were available on regular badging for 1968 and 1969. The number of regular and “other” badges were comparable after 1969.

Figure A.7-1. Total Number of Regular and “Other” Badges Issued per Badging Period at TRA from Late 1956 through 1974

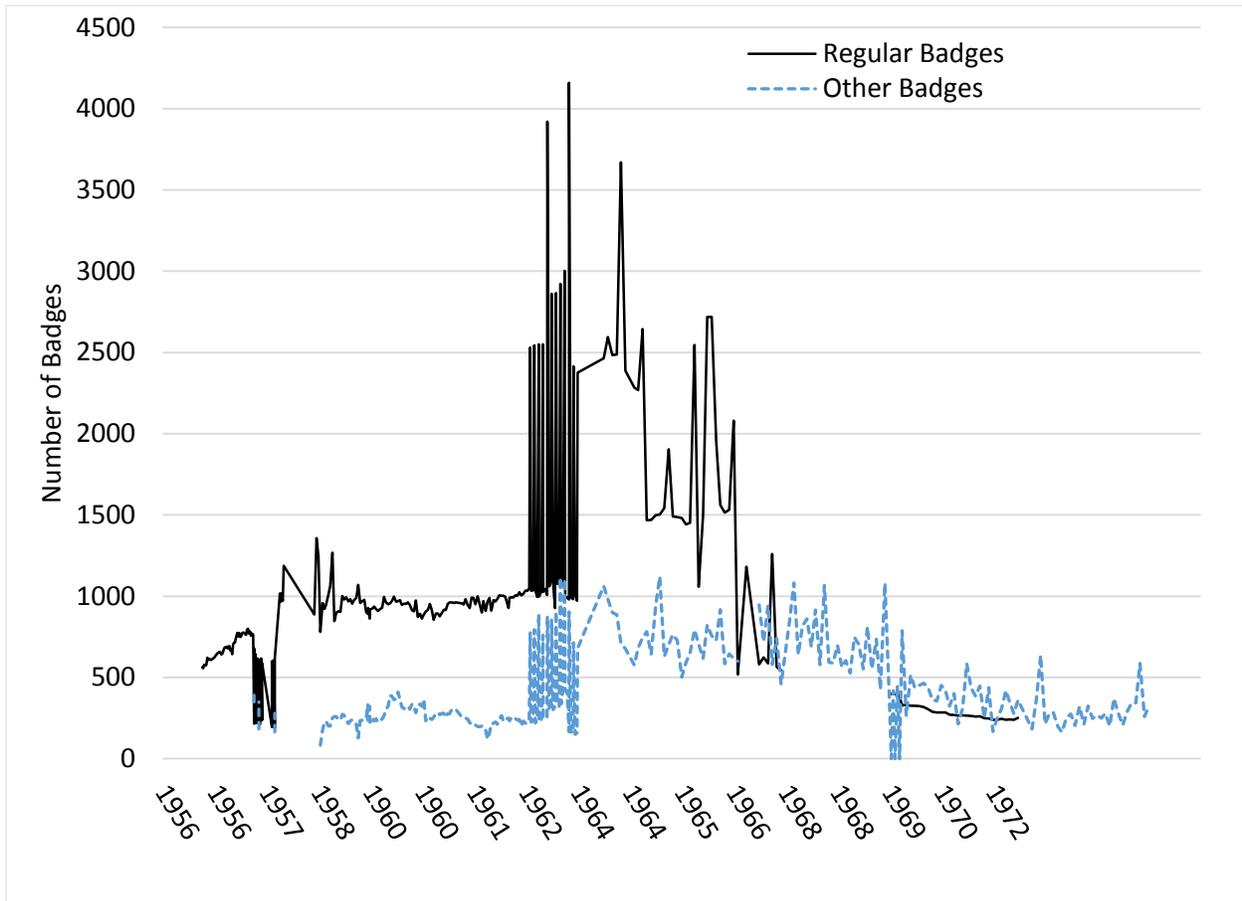


Figure A.7-2 plots the maximum observed penetrating dose (in units of mrem) for TRA in available documentation. As shown in the figure, three different badging exchange schedules were noted based on the time period. For the earlier time periods, badges were on a weekly exchange schedule; this changed to a bi-weekly schedule from approximately 1959 into early 1970. After this time, badges were exchanged on a monthly schedule.

The highest observed penetrating dose was 11 rem during a 2-week period in January 1961. However, information provided in the source documentation indicates that this exposure was likely related to the Stationary Low-Power Reactor (SL-1) incident. Figure A.7-3 plots the maximum penetrating dose with this data point removed. As seen in both Figure A.7-2 and Figure A.7-3, the maximum penetrating dose was approximately 250 mrem during the weekly badging cycles, 500–1,000 mrem for bi-weekly exchanges, and 500–1,500 mrem for monthly exchanges.

Figure A.7-2. Maximum Penetrating Dose (mrem) for TRA

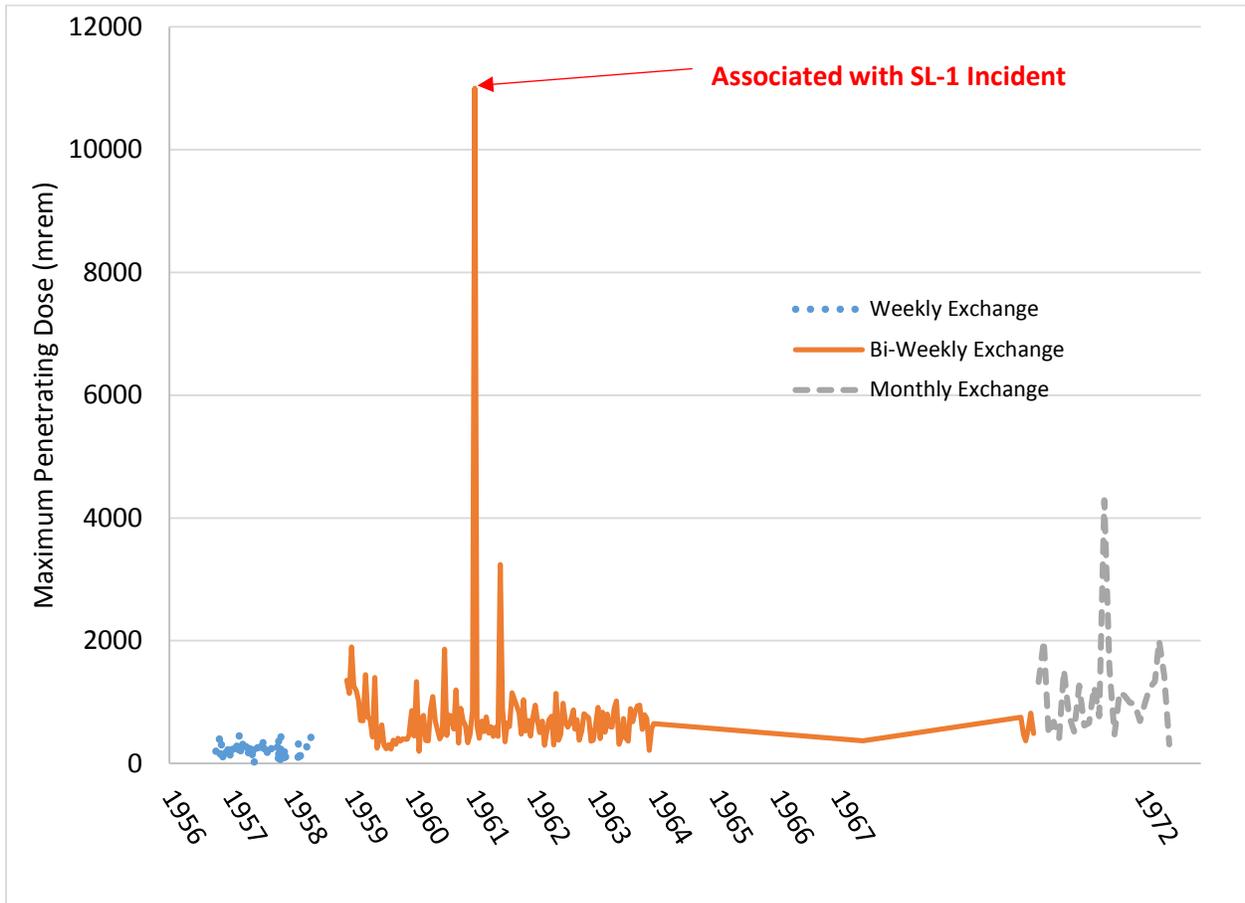


Figure A.7-3. Maximum Penetrating Dose (mrem) for TRA with Suspected SL-1 Result Removed

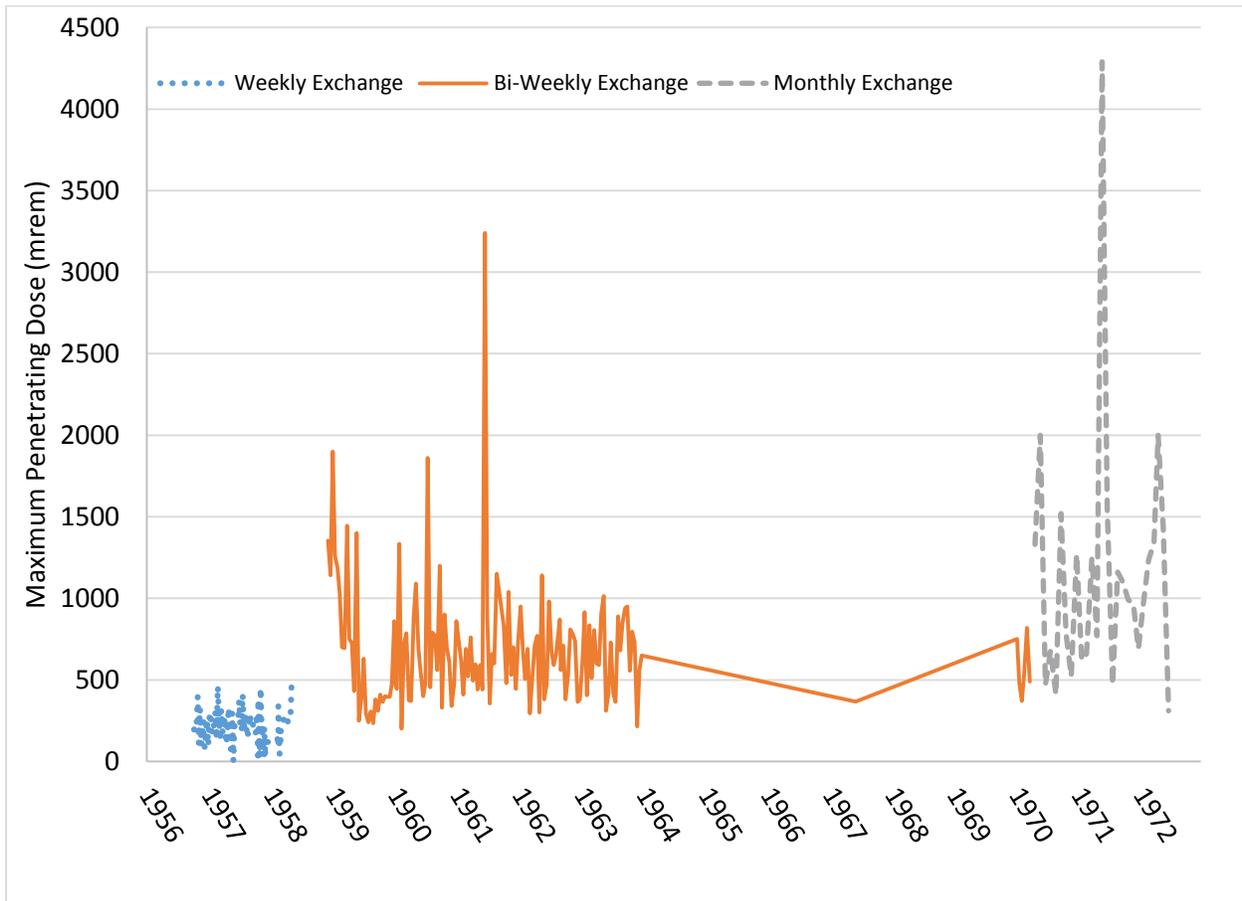
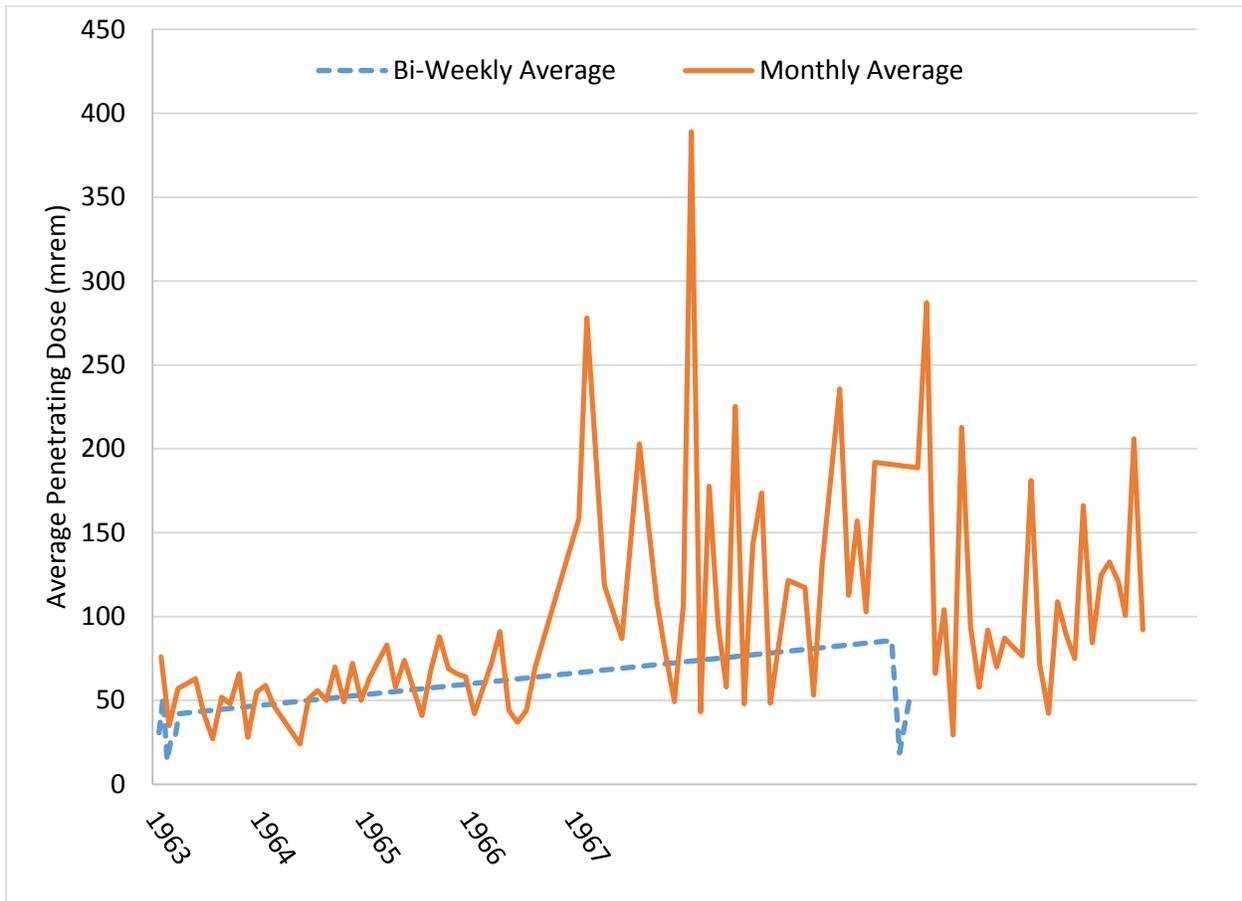


Figure A.7-4 plots the average penetrating dose (mrem) for TRA. It should be noted that while TRA workers were mainly on a bi-weekly schedule during this period, averages were most often reported on a monthly basis. Monthly averages from 1963 through mid-1966 gradually increased but were generally on the order of 50 mrem. After this time, monthly averages ranged from 50 mrem to approximately 250 mrem. The highest observed monthly average was 389 mrem in February 1968.

Figure A.7-4. Average Penetrating Dose (mrem) Evaluated on Monthly and Bi-Weekly Intervals



In addition to regular and “other” badges issued to TRA personnel, construction personnel were tabulated under a separate dosimetry code (Mx or MTX). The number of construction badges issued in TRA and the maximum observed penetrating dose are shown in Figure A.7-5. As seen in the figure, the number of construction badges issued varied significantly based on the specific time period. In general, the number of construction badges could be generally characterized as 100–150 badges per exchange cycle, with a range from a low of 16 to a high of 225.

Likewise, the maximum penetrating dose among construction workers also varied widely depending on time period. Available documentation suggests that the two highest observed values (2.88 and 1.60 rem) were associated with the SL-1 incident. Figure A.7-6 displays the same data with these two data points removed. As seen in Figure A.7-6, maximum penetrating exposures were generally about 100–200 mrem, with only a few occasions exceeding this range and several badge cycles in which no external penetrating dose was accrued by construction personnel.

Figure A.7-5. Total Number of Construction Badges Issued and Maximum Observed Penetrating Dose (mrem)

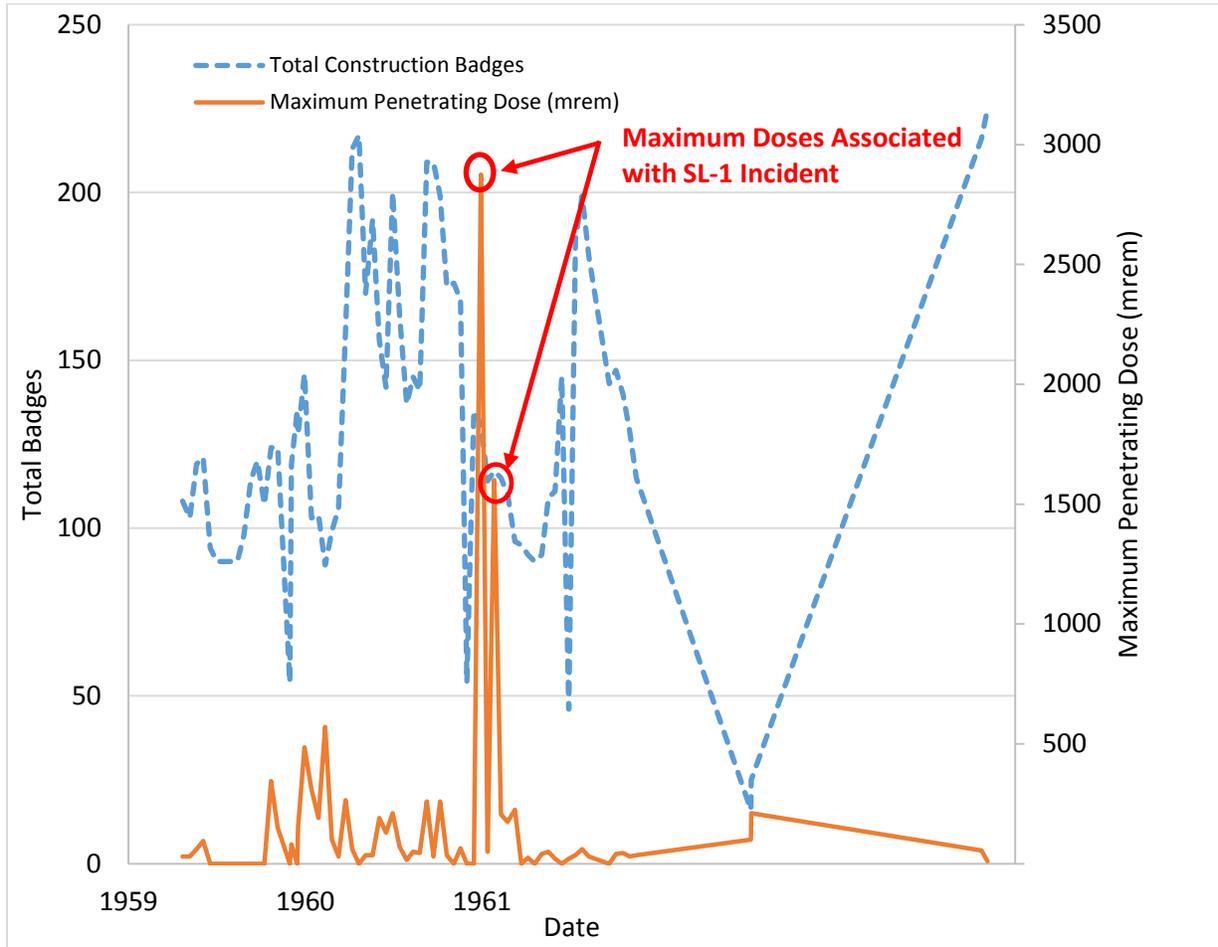
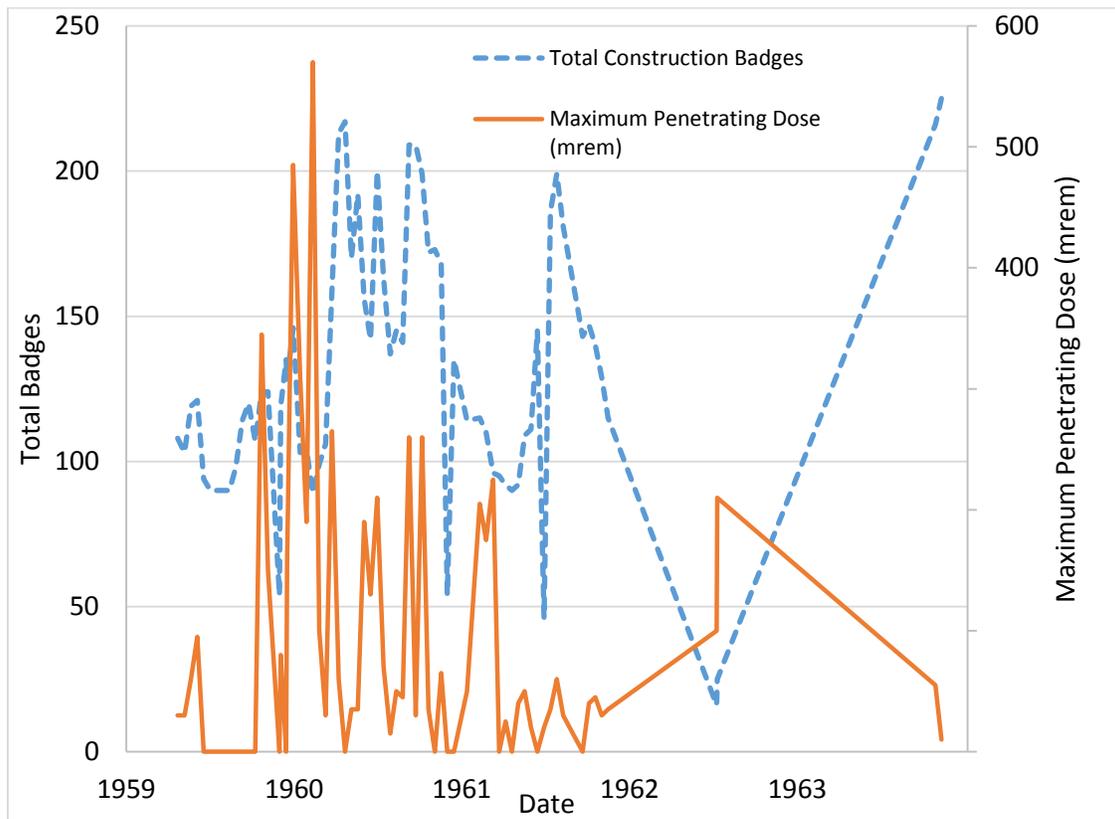


Figure A.7-6. Total Number of Construction Badges Issued and Maximum Observed Penetrating Dose (mrem) with Suspected SL-1 Exposures Removed



A.8. SITE RESEARCH DATABASE REPORTS USED IN DATA COMPILATION

SRDB #	Document Title
125789	Health and Safety Records Section Monthly Reports 1962
125790	Health and Safety Records Section Monthly Reports 1963
125920	Dosimetry Branch Reports 1972–1975
125924	Dosimetry Branch Activity Monthly Reports 1963
125926	Dosimetry Branch Activity Monthly Reports 1964
125927	Dosimetry Branch Activity Monthly Reports 1965
125928	Dosimetry Branch Activity Monthly Reports 1966
125930	Dosimetry Branch Activity Monthly Reports 1967–1970
125934	Dosimetry Branch Activity Monthly Reports 1970–1972
125935	Health and Safety Records Section Monthly Reports 1959
125938	Health and Safety Records Section Monthly Reports 1960

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Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 52 of 59
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SRDB #	Document Title
125948	Health and Safety Records Section Monthly Reports 1961
125950	Health and Safety Records Section Monthly Reports 1962
125959	Health and Safety Records Section Monthly Reports 1964
125961	Health and Safety Records Section Monthly Reports 1965
125967	Health and Safety Records Section Monthly Reports 1966
125974	Health and Safety Records Section Monthly Reports 1969
125975	Health and Safety Records Section Monthly Reports 1970
125976	Health and Safety Records Section Monthly Reports January and February 1971
125992	Health and Safety Records Section Monthly Reports January and June 1972
125993	Health and Safety Records Section Monthly Reports March through December 1971
126015	Health and Safety Records Section Yearly Summary for 1963
126023	Health and Safety Records Section Yearly Summary for 1965
138710	SPERT Health Physics Monthly Report January – November 1963
139205	SPERT HP Reports 1958
153691	Monthly Dosimetry Badge Reports 1973
153692	Monthly Dosimetry Badge Reports 1974

NOTICE: This report has been reviewed to identify and redact any information that is protected by the Privacy Act 5 U.S.C. § 552a and has been cleared for distribution.

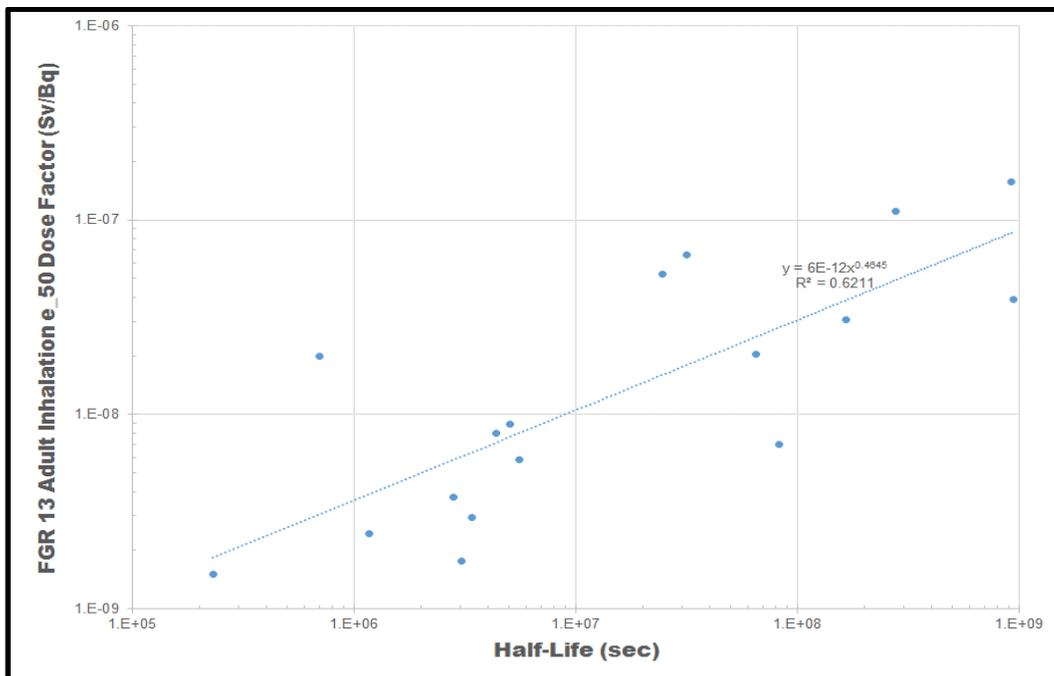
Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 53 of 59
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APPENDIX B: EVALUATION OF APPLICABILITY OF OTIB-0054 TO RECONSTRUCTING INTERNAL DOSES AT TEST AREA NORTH

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Stephen F. Marschke, SC&A
John Mauro, SC&A

In the various reports and discussions leading up to this report, it was SC&A’s impression that the methodology in ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses*, Revision 04 (ORAUT 2015; hereafter referred to as “OTIB-0054”), is generally claimant favorable as applied to the Test Area North (TAN) reactors, especially the Heat Transfer Reactor Experiment (HTRE) reactors. This judgment assumed that the burnup times for the HTRE reactors were much shorter than those of the reactors that were used to develop the radionuclide mixes in the irradiated fuel used in OTIB-0054. Thus, the mix of radionuclides used in OTIB-0054 would be composed of a greater fraction of the long-lived radionuclides (i.e., radionuclides that should have higher inhalation dose conversion factors [DCFs] simply because of their generally longer residence times in the body). Figure B-1 plots the limiting effective $e(50)^1$ doses in Federal Guidance Report (FGR) No. 13 (EPA 1999; hereafter “FGR 13”) as a function of the radioactive half-lives of the radionuclides and demonstrates the general validity of this premise.

Figure B-1. FGR 13 Adult Inhalation $e(50)$ Dose Factor versus Half-Life



However, given the complexity of this subject, SC&A took additional steps to evaluate the degree to which OTIB-0054 is, in fact, applicable or claimant favorable to workers at TAN

¹ $e(50)$ refers to the committed effective dose per unit acute intake over 50 years following the exposure.

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 54 of 59
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facilities, keeping in mind that OTIB-0054 is used to reconstruct the intake rates of individual radionuclides by TAN workers who were routinely monitored on a gross beta/gamma urine bioassay program.

B.1. RADIONUCLIDE INVENTORY IN THE IRRADIATED FUEL

The first step was to use the ORIGEN code to calculate the number of radionuclide atoms in the irradiated fuel per metric ton uranium (MTU) at 10 days after shutdown as a function of power level and duration of burnup. These were then converted to activities by multiplying the number of atoms by the radioactive decay coefficient of each radionuclide (i.e., $A = \lambda N$). These calculations were performed for burnup times ranging from 20 hours to 200 days for both low- and high-power cases (i.e., 20 and 200 MW, respectively). The results of the low-power ORIGEN run are shown in Figure B-2 for several key radionuclides. As shown, the activities for many of the radionuclides (e.g., strontium-90 [Sr-90], cesium-137 [Cs-137], and ruthenium-106 (Ru-106) continue to build up even after 200 days of burnup time, while other radionuclides (most notably iodine-131 [I-131]) have reached equilibrium, and their activities begin to level off.

It is noteworthy that at 20 hours burnup at low power levels, the activity of Cs-137 and Sr-90 are about the same (i.e., 2×10^{12} becquerel [Bq]/MTU),² which is expected, because they both have about the same cumulative fission product yield (about 6%),³ and the rate of increase of the activity of these radionuclides is about the same because their half-lives are similar (i.e., about 30 years). Thus, their absolute activities continue to increase at the same rate over time and do not flatten out over the period examined in this ORIGEN run. However, other radionuclides start out at a much higher level because they have relatively shorter half-lives than Cs-137 and Sr-90, and, therefore, even if a relatively small number of atoms of these radionuclides are produced relative to Cs-137 and Sr-90 during the fission process, their activity is often initially high because of their higher decay rates (shorter half-lives) as compared to Sr-90 and Cs-137.

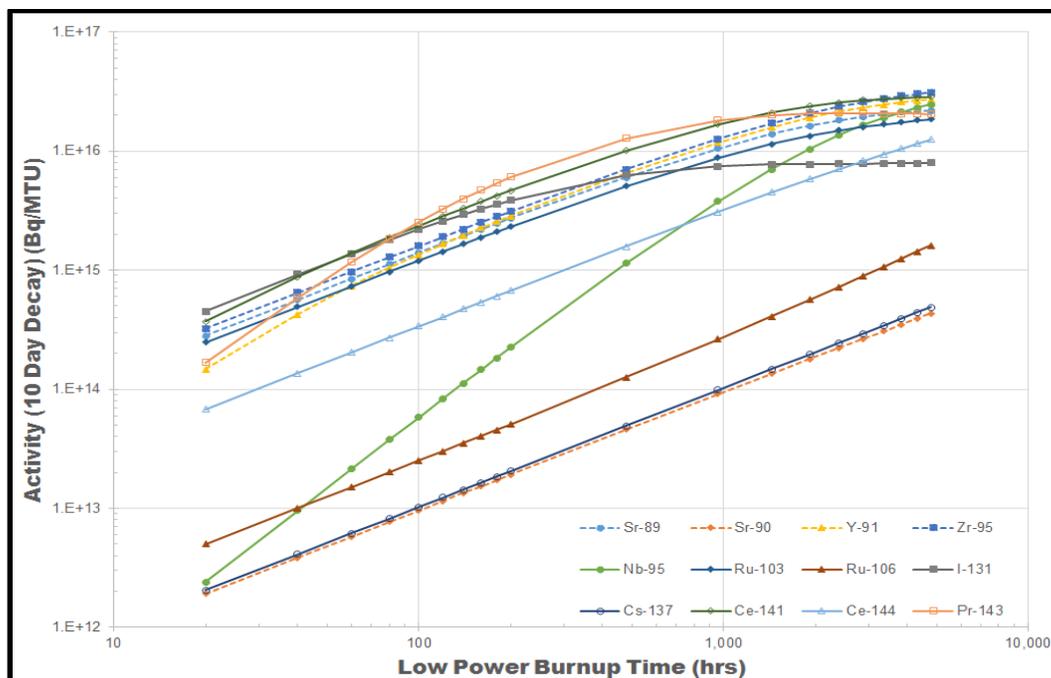
Niobium-95 (Nb-95) is interesting because the combination of its small independent fission yield and the relatively long half-life of its parent, zirconium-95 (Zr-95) (64 days) results in very little Nb-95 being produced at 20 hours of burnup time, yielding approximately the same activity as Sr-90 and Cs-137 in the irradiated fuel. However, the Nb-95 activity climbs rapidly once a sufficient Zr-95 activity has been established and approaches an equilibrium level at 200 days of buildup (as shown in Figure B-2). This occurs because its rate of production due to Zr-95 decay is relatively high (atoms produced per unit time). However, its rate of decay is also high, thereby resulting in a rapid rise in activity as a function of time relative to the activity of Cs-137 and Sr-90 in the irradiated fuel. Figure B-2 is the foundation upon which subsequent calculations related to the low-power case index of harm are based. A similar figure could be made for the high-power case. The results of the index of harm calculations are not always intuitively obvious because the inventory of a given radionuclide associated with a given burnup time is the result of

² Note: 1 Bq = 1 disintegration per second.

³ The term “cumulative fission product yield” is used because many of the radionuclides in the irradiated fuel result from the direct fission product yield in combination with the decay of parent fission products, each of which has its own fission product yield and decay rate. Of course, sufficient burnup time must have occurred to allow for the buildup of the parent fission products; otherwise, use of the “cumulative fission product yield” will overestimate the radionuclide’s inventory.

a combination of factors, including the fission yield (atoms produced per fission), the decay rate of the radionuclide and its parents, and the assumed cooldown time.

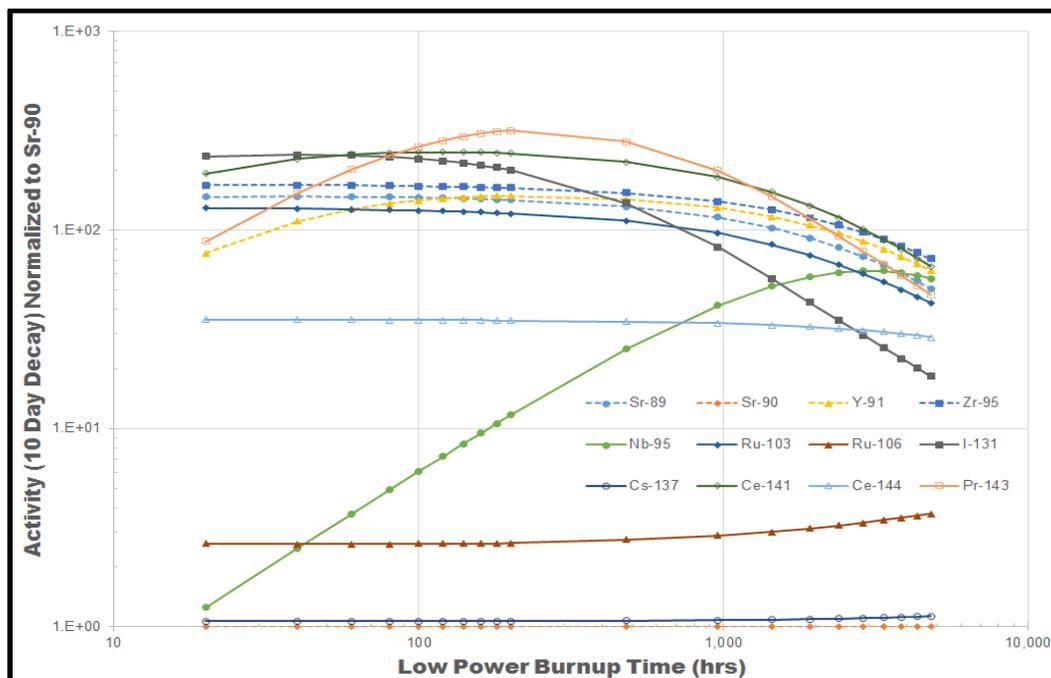
Figure B-2. ORIGEN-Calculated Low-Power Radionuclide Activity



B.2. NORMALIZATION OF RADIONUCLIDE INVENTORY TO THE INVENTORY OF SR-90

Because the bioassay monitoring results are often provided as gross beta, SC&A normalized the ORIGEN results to Sr-90, a beta emitter and a prominent fission product. Figure B-3 shows the results of the Figure B-2 low-power activities normalized to the Sr-90 activity. The I-131 curve bends downward, not because the I-131 activity is decreasing, but because the Sr-90 activity is increasing at a faster rate, making for a decreasing ratio between the two radionuclides. Also, for praseodymium-143 (Pr-143), and to a lesser extent yttrium-91 (Y-91), the curves in Figure B-3 initially increase and then decrease, forming a kind of arch. This behavior indicates that the activities for these radionuclides are initially increasing more rapidly than for Sr-90, but then the Pr-143 and Y-91 activities reach equilibrium, while the Sr-90 activity continues to increase.

Figure B-3. Low-Power Radionuclide Activity Normalized to Sr-90

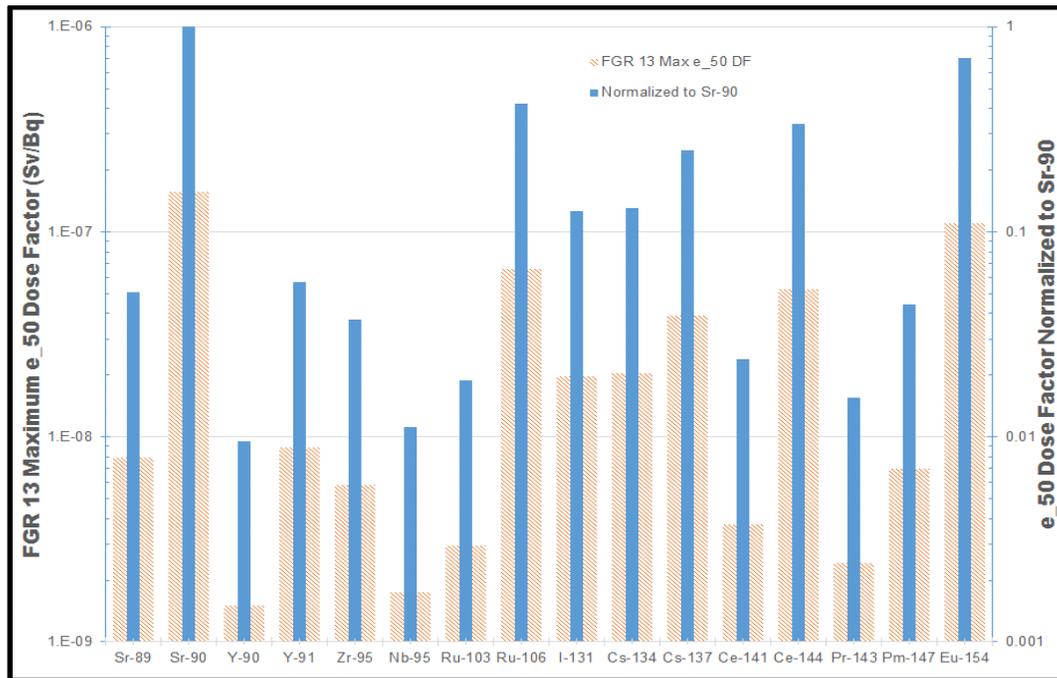


B.3. DOSE CONVERSION FACTORS NORMALIZED TO SR-90 DOSE CONVERSION FACTOR

The next step in the calculation was to obtain dose conversion factors for each of the organs of interest from FGR 13. For this exercise, adult inhalation DCFs were used. For each radionuclide and age group, FGR 13 provides multiple inhalation DCFs, depending on the chemical nature of the material inhaled. For most radionuclides, FGR 13 provides three inhalation DCFs to account for slow, medium, and fast clearance of the material from the lungs. For a few radionuclides (e.g., I-131), FGR 13 provides additional inhalation DCFs. For this study, all the FGR 13 adult inhalation DCFs for each radionuclide were examined, and the largest factors were used in the calculation.

Figure B-4 shows the FGR 13 e(50) DCFs that were selected for use in this study. Because each radionuclide's activity has been normalized to the Sr-90 activity, it is also necessary to normalize the individual radionuclide DCFs to the Sr-90 DCF. These normalized e(50) DCFs are also shown in Figure B-4. This procedure was employed to obtain Sr-90 normalized DCFs for each organ of interest for this study (i.e., bone surface, breast, lower large intestine (LLI) wall, red bone marrow, skin, lung, liver, kidneys, and thyroid); figures like Figure B-4 could be developed, if desired, for each organ.

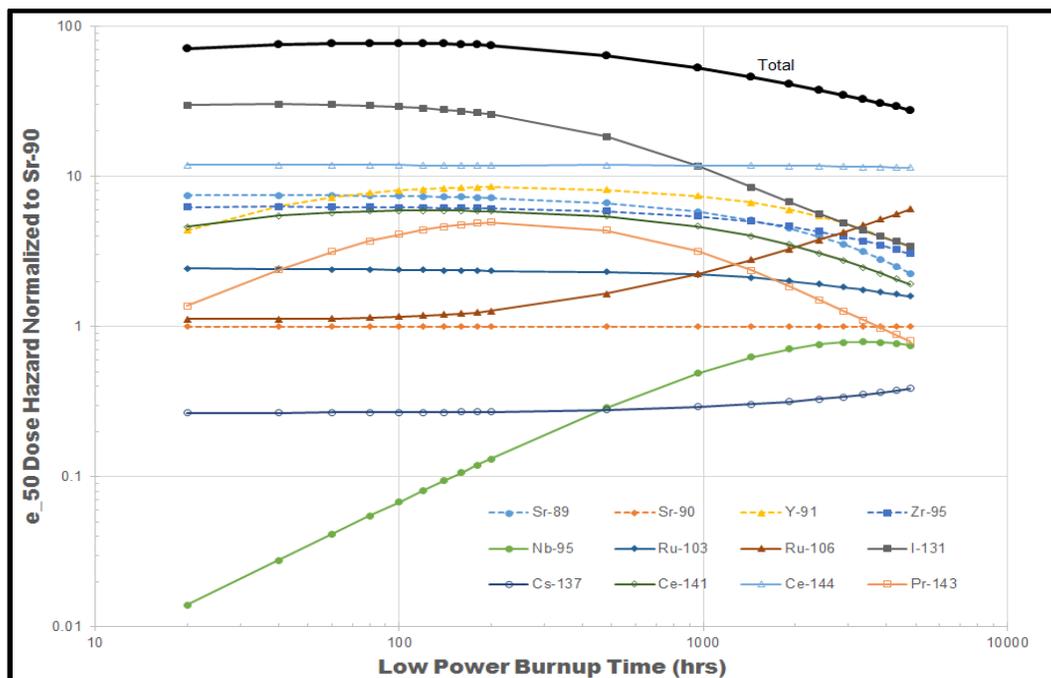
Figure B-4. FGR 13 Inhalation e(50) Dose Conversion Factors and Sr-90-Normalized Dose Conversion Factors



B.4. RELATIVE INDEX OF HARM

The Sr-90 normalized activities (e.g., Figure B-3 for the low-power case) were multiplied by the Sr-90 normalized inhalation DCFs (e.g., Figure B-4 for e(50) dose) to obtain an Sr-90 relative hazard measure for each radionuclide. For the low-power case, Figure B-5 presents e(50) hazard for individual radionuclides relative to that of Sr-90. Notice that the e(50) hazard index for I-131 relative to that for Sr-90 decreases over time. This occurs because the amount of I-131 remains constant in the irradiated fuel over time, while the amount of Sr-90 continues to increase. The e(50) hazard index for other radionuclides, such as Nb-95, starts low and rises as a function of time relative to that of Sr-90.

Figure B-5. Low-Power e(50) Dose Relative Hazard Measures



The final step in the SC&A analysis was to compare the hazard measures at different burnup times (e.g., 20 hours versus 200 days). Table B-1 presents the calculated relative hazard measures for the low-power case at burnup times of 20 hours and 200 days, as well as the ratio of the two. Notice in Table B-1 that the e(50) organ 20-hour and 200-day relative hazard measures are 70.6 and 27.6, respectively, and that these are the same values shown by the Total curve in Figure B-5.

Table B-1. Relative Hazard Measures for Low Power Level (20 MW)

Organ	Relative Hazard		Ratio 20hr/200d
	200 d	20 hr	
e(50)	2.76E+01	7.06E+01	2.55
Bone surface	3.25E+01	5.50E+01	1.69
Breast	7.57E+02	1.19E+03	1.57
LLI wall	1.94E+02	3.78E+02	1.95
Red bone marrow	2.94E+01	4.29E+01	1.46
Skin	5.79E+02	8.66E+02	1.50
Lung	2.41E+01	4.05E+01	1.68
Liver	2.62E+04	3.36E+04	1.28
Kidneys	9.22E+02	1.37E+03	1.49
Thyroid	1.27E+04	1.56E+05	12.29

Effective Date: 12/08/2016	Revision No. 0 (Draft)	Document No./Description: SCA-TR-2016-SEC012	Page No. 59 of 59
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Likewise, Table B-2 presents the calculated relative hazard measures for the high-power case at burnup times of 20 hours, 20 days, and 200 days, as well as the ratio of the two shorter times to 200 days.

Table B-2. Relative Hazard Measures for High Power Level (200 MW)

Organ	Relative Hazard			Ratios	
	200 d	20 d	20 hr	20d/200d	20hr/200d
e(50)	3.65E+01	6.63E+01	7.07E+01	1.82	1.93
Bone surface	3.66E+01	5.88E+01	5.51E+01	1.61	1.50
Breast	1.21E+03	1.23E+03	1.19E+03	1.02	0.98
LLI wall	2.71E+02	4.58E+02	3.79E+02	1.69	1.40
Red bone marrow	3.38E+01	4.94E+01	4.29E+01	1.46	1.27
Skin	9.70E+02	9.16E+02	8.68E+02	0.94	0.89
Lung	3.17E+01	4.64E+01	4.05E+01	1.47	1.28
Liver	3.07E+04	3.80E+04	3.37E+04	1.24	1.10
Kidneys	1.40E+03	1.91E+03	1.38E+03	1.37	0.98
Thyroid	1.89E+04	9.68E+04	1.57E+05	5.12	8.29

Those relative indices of harm (shown in the Ratios columns) that are substantially greater than 1.0 indicate those organs for which OTIB-0054, with the assumption of longer burnup times than the TAN reactors, might underestimate internal doses. However, for all organs other than the thyroid, the ratios that exceed 1.0 do so only modestly. These or similar tables could be used as adjustment factors to the doses derived using OTIB-0054 to apply to those workers for whom it might be important to increase the calculated assigned dose.