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INTERIM SUMMARY REPORT ON THE EVALUATION OF NIOSH'S IDAHO NATIONAL LABORATORY SEC-00219 PETITION EVALUATION REPORT

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Prepared by

Bob Barton
Ron Buchanan
Joe Fitzgerald
Milton Gorden
John Mauro
Amy Meldrum
Stephen Ostrow

S. Cohen & Associates
1608 Spring Hill Road, Suite 400
Vienna, Virginia 22182

Saliant, Inc.
5579 Catholic Church Road
Jefferson, Maryland 21755

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Task Manager: ______________________________ Date: ___________ John Stiver, MS, CHP	Supersedes: N/A
Project Manager: ______________________________ Date: ___________ John Stiver, MS, CHP	Reviewers: Joe Fitzgerald Stephen Ostrow John Stiver

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ABBREVIATIONS AND ACRONYMS

Advisory Board or ABRWH	Advisory Board on Radiation and Worker Health
AEC	Atomic Energy Commission
ANC	Aeroject Nuclear Company
ANL-W	Argonne National Laboratory West
ANP	Advanced Nuclear Propulsion
ARMF	Advanced Reactivity Measurement Facility
ATR	Advanced Test Reactor
ATRCF	Advanced Test Reactor Critical Facility
CAM	constant air monitor
CATI	Computer-Assisted Telephone Interview
CFA	Central Facilities Area
c/m	counts per minute
CPP	Chemical Processing Plant
CFRMF	Coupled Fast Reactivity Measurement Facility
D&D	decontamination and decommissioning
DCAS	Division of Compensation Analysis and Support
DOE	(U.S.) Department of Energy
DR	dose reconstruction
DU	depleted uranium
ER	Evaluation Report
ERDA	Energy Research Development Administration
ETR	Engineering Test Reactor
ETRCF	Engineering Test Reactor Critical Facility
FAP	Fission and Activation Product
FFTF	Fast Flux Test Reactor
IET	Initial Engine Test
IMBA	Integrated Modules for Bioassay Analysis
INC	Idaho Nuclear Corporation
INL	Idaho National Laboratory
HP	Health Physics
LPTF	Low Power Test Facility
MAP	mixed activation product
MDA	minimum detectable activity
MFP	mixed fission product

mrem	millirem
MTR	Materials Test Reactor
MTHM	metric ton of heavy metal
MTU	metric ton unit
MW	megawatt
MWd	megawatt-day
NIOSH	National Institute for Occupational Safety and Health
NOCTS	NIOSH/OCAS Claims Tracking System
NRC	Nuclear Regulatory Commission
NRTS	National Reactor Testing Station
OCAS	Office of Compensation Analysis and Support (now DCAS)
ORAUT	Oak Ridge Associated Universities Team
ORIGEN2	Oak Ridge Isotope Generation and Depletion Code
ORNL	Oak Ridge National Laboratory
R	Roentgen
RFP	Rocky Flats Plant
RMF	Reactivity Measurement Facility
RML	Radiation Measurement Laboratory
RWMC	Radioactive Waste Management Complex
SC&A	S. Cohen and Associates
SEC	Special Exposure Cohort
SMC	Specific Manufacturing Capability
SRDB	Site Research Database
STPF	Shield Test Pool Facility
TAN	Test Area North
TBD	Technical Basis Document
TIB	Technical Information Bulletin
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TRU	transuranic
TSF	Technical Support Facility
WBC	whole-body count
WG	Work Group
WRRTF	Water Reactor Research Test Facility

1.0 INTRODUCTION AND BACKGROUND

Following the National Institute for Occupational Safety and Health's (NIOSH's) release of the Idaho National Laboratory (INL) Special Exposure Cohort (SEC) Petition SEC-00219 Evaluation Report (ER) on March 12, 2015 (NIOSH 2015), and its presentation at the March 26, 2015, Advisory Board on Radiation and Worker Health (ABRWH, or the Advisory Board) meeting in Richland, Washington, SC&A was asked to begin reviewing the ER. The April 22, 2015, planning teleconference held with SC&A, NIOSH (including its contractor), and some INL Work Group (WG) members, as well as several e-mail communications, established the scope of SC&A's preliminary investigations, which are intended initially to support a July 8, 2015, INL WG meeting and a July 23, 2015, session at the ABRWH meeting in Idaho Falls, Idaho. The results of the discussions at these meetings will lead to further investigations to support the Board's consideration of the SEC petition and NIOSH's evaluation report.

In light of the complex nature of the site, its operations, and the SEC ER itself, the Advisory Board determined that its review of the ER should be performed in a graded manner, whereby SC&A would first conduct a preliminary review of certain issues of immediate concern to the Board. It is recognized that this review, in addition to being preliminary due to time constraints, also is very much a work in progress, as NIOSH has acknowledged that the evaluation itself is still a work in progress as new information continues to be obtained, more worker interviews conducted, and coworker and other dose reconstruction (DR) models are developed.

SC&A's initial charge by the Board was to provide information and assessments, primarily on two topics of concern to the WG:

(1) *Class Definition*: NIOSH proposed the following class definition in its ER:

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Idaho National Laboratory in Scoville, Idaho, and were monitored for external radiation at the Idaho Chemical Processing Plant (CPP) (e.g., at least one film badge or TLD dosimeter from CPP) between January 1, 1963 and December 31, 1974 for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.*

The above definition implicitly assumes that all employees who might have been present at CPP during the class period and potentially exposed to internal sources of radiation (which exposure NIOSH claims it cannot reconstruct for CPP during the class period) can be identified by the fact that they were all monitored for external radiation, and that their exposures can be attributed to CPP. As stated in the ER (p. 6): "By policy, INL monitored by dosimeter all personnel who were expected to receive any radiation dose or whose work was centered at the site." The specific concerns for this ER are whether all workers exposed to radiation at CPP were, in fact, monitored; whether their dosimetry records are available; and whether the workers affected by this SEC can be reliably placed at the CPP during the specified time period.

SC&A considered the class definition assumption through two lines of investigation: by reviewing about 60 workers' statements made through three series of interviews conducted during June, September, and November 2014; and by performing a detailed examination of the available dosimetry and location records supporting identification of which workers were exposed at CPP during the given time period (this examination may be referred to as a "gap analysis," since it is attempting to identify any "gaps" in the available data). That analysis (SC&A 2015) was delivered to the INL WG as a stand-alone document on June 29, 2015.

(2) *Dose Reconstructability and Gap Analysis*: The Board recognized that NIOSH's investigations regarding reserved areas (see Table 1) are under way, and it would be premature for the Board and its contractor to perform SEC assessments into such areas before NIOSH had an opportunity to complete those investigations, or at least portions of those investigations. Hence, this report does not address issues that are currently held in reserve by NIOSH. However, the Board did conclude that it would be appropriate for SC&A to begin its investigations into activities, facilities, and time periods at the site that NIOSH recommended denial of an SEC, because it felt that worker doses at such facilities could be reconstructed with sufficient accuracy. The Board also recognized that the scope of such an investigation was vast and should be approached in a highly organized and deliberative manner. As such, the Board directed SC&A to prepare a plan for identifying data gaps for each of the areas, facilities, and campaigns at the site where NIOSH felt that doses could be reconstructed with sufficient accuracy. This status update report is provided in response to that Board directive. The report represents and plans for a data gap analysis, and also begins implementing the plan for specific sites as a means to help determine that such a plan will be useful in assisting the Board in its SEC recommendations.

While the SEC class definition includes a specific cohort of workers over a specific time period, at the same time it also excludes other INL workers (i.e., those working at locations other than at CPP) at other time periods. NIOSH claims that it can adequately reconstruct doses for those workers and, so, did not include them in the SEC class. SC&A approached this issue through several different methods that can be characterized as either horizontal or vertical. In the former, SC&A examined the DR methodology applied by NIOSH for all INL personnel, and for the latter SC&A looked at some specific characteristics of the individual areas at the INL site where personnel might have been exposed to radiation. The working areas considered are:

- Chemical Processing Plant (CPP)
- Test Reactor Area (TRA)
- Test Area North (TAN)
- Misc. Reactor Areas
- Central Facilities Area (CFA)
- Burial Grounds

Information on the characteristics of these areas appears in many sources, including, in addition to the ER, the INL site profile [comprised of six separate technical basis documents (TBDs)], reports produced by NIOSH (and its contractor) and SC&A, documents collected in the Site

Reference Data Base (SRDB), as well as documents appearing elsewhere. SC&A has examined some pertinent documents in this preliminary assessment.

Table 1 summarizes the ER's conclusions about which combinations of operating areas and time periods can be reconstructed and which cannot. Note that certain combinations have been "reserved" for further study, which might result in a determination that doses can be reconstructed, or a recommendation that affected personnel be added to the SEC class. In addition, the feasibility of DR for certain area and time period combinations has been assumed based on the assumption of future development of the bioassay database and acceptable coworker models.

For administrative reasons, Argonne National Laboratory West (ANL-W), although physically located within the INL site, will be the subject of a separate SEC investigation, which NIOSH is currently conducting. Hence, ANL-W has been excluded from these discussions.

Table 1. INL SEC 00219 Evaluation Report: Feasibility of Reconstructing Internal Doses for 1/1/63–12/31/74 [a],[b]

Operating Area [c]	Reconstruct?
Chemical Processing Plant (CPP)	**Yes:** 1/1/53–12/31/62: Bioassay data available. **No:** 1/1/63–12/31/74: Insufficient bioassay data to support reconstruction of internal exposures to U, Np, Pu, and other related transuranics. Beginning in 1963, increased α-contamination levels were detected with no accompanying increase in Pu bioassay. **Reserved:** 1/1/75 – ?: Evaluate CPP HP Program Improvements 1971–1980.
Test Reactor Area (TRA)	**Yes:** • 1/1/52–12/31/66 • 1/1/67–12/31/70: **coworker model**.[d]
Test Area North (TAN)	**Yes:** • 1/1/55–12/31/66 • 1/1/67–12/31/70: **coworker model.** **Reserved:** 1/1/61–12/31/70: TAN-607 (Fuel Storage Vaults) and TAN-615 (Actuator Building) due to potential U exposures in the absence of MFPs.
Misc Reactor Areas	**Yes:** • 1/1/55–12/31/66 • 1/1/67–12/31/67 and 1/1/69–12/31/70: **coworker model**. **Reserved:** 1/1/68–12/31/68: Insufficient data currently available for ARA-I (Auxiliary Reactor Area-I) due to potential unmonitored exposures to Pa-233 from separation of that isotope from irradiated Th slugs.
Central Facilities Area (CFA)	**Yes:** • 1/1/49–12/31/66 • 1/1/67–12/31/70: **coworker model**.
Burial Grounds	**Yes:** • 1/1/52–12/31/66 • 1/1/67–12/31/68: **coworker model**. **Reserved:** 1/1/69–12/31/70: Evaluate a newly-implemented procedure of waste exhumation and retrieval.

(a) SEC 00219 time period evaluated by NIOSH is 1/1/63–12/31/74 for CPP based on insufficient data to reconstruct internal doses.
(b) Operations at INL began 1/1/49, but some of the areas began operations later.
(c) ANL-W is not included in this SEC, but will be the subject of SEC 00224, which is under development.
(d) NIOSH is developing an internal, mixed fission product (MFP) coworker model stratified by area for time periods beginning 1/1/67.

2.0 SC&A ASSESSMENTS

SC&A addressed the issue of class definition as requested by the Board in a recently issued paper (SC&A 2015), and has addressed our traditional review of the INL ER by an early focus on significant issues and gaps for those operations and time periods for which dose reconstructability is being claimed. This latter review is being performed through a number of relatively quick studies, which are a combination of scoping analyses and selected "vertical" investigations. Since the studies are still in progress, and recognizing that the Board would like to preview some of the results, though preliminary, for discussion at the July 8, 2015, INL WG meeting and the July 23, 2015, INL SEC session at the ABRWH meeting, SC&A has combined summaries of these individual draft studies into this interim summary report. Some of the reports and studies are concerned with specific INL areas, while others are concerned with common methodologies and concerns that crosscut the different operating areas. Upon completion of the individual studies, SC&A will provide a final report that combines all aspects of these studies, and our overall preliminary findings, or a series of individual reports. The progress to date of all investigations is summarized in the following sections.

2.1 CLASS DEFINITION

SC&A addressed the issue of class definition with two very different approaches; the first combed through worker interview summaries and the second examined in depth the availability of worker records pertaining to external dosimetry at the CPP. As previously noted, the detailed review of external dosimetry was reported in the stand-alone document SC&A 2015. The description of the interview summaries is discussed in this status update report.

2.1.1 Interview Summaries

SC&A reviewed about 50 sets of worker interview summaries that were conducted by SC&A, NIOSH, and Board members in Idaho in June, September, and November 2014. At the time of the review, not all of the summaries examined were finalized. SC&A believes that none of the summaries contradict NIOSH's SEC class assumption that all workers at CPP had personal monitors for external radiation during the proposed SEC time period of January 1, 1963–December 31, 1974. Of course, it should be realized that absence of a contradictory statement in the interview summaries does not confirm the validity of the SEC class definition, since not all workers at the plant were interviewed and memories can be faulty.

Several statements indicate that the requirement for all workers at CPP to be badged upon entry to the area might have been relaxed at some later time (i.e., later than the end date of the proposed SEC time period) to require badging only for entry into radioactive areas, or area monitoring substituted for personal monitoring in some cases (which might have been only for construction in "new" areas outside the radioactive boundaries). This could be an issue if NIOSH decides to extend the time period of the SEC class beyond the current December 31, 1974 endpoint.

<u>Recommendation</u>: SC&A recommends that this line of inquiry be continued to include both past interviews that were not available when SC&A did its initial review, as well as new interviews

where more specific questions could be asked of the workers regarding external dosimetry for CPP workers.

2.2 DOSE RECONSTRUCTABILITY: DATA COMPLETENESS, DATA QUALITY, AND DATA ADEQUACY

The second major issue examined in this preliminary assessment of the INL SEC-00219 ER consists of initial investigations into whether NIOSH can adequately reconstruct doses as it claims for the areas and time periods outside the proposed SEC class definition. These investigations are best described as data gap analyses, where SC&A reviewed selected Site Research Database (SRDB) folders as a means to identify areas, sub-areas, and time periods at INL, including various research projects and special campaigns, where worker external and/or internal dosimetry data might be incomplete, or might have employed protocols that could have missed potentially important sources of exposures; examples of the latter are internal exposures to exotic radionuclides or neutrons. SC&A performed several preliminary studies to examine these issues. These investigations are still underway, but are summarized in the following sections.

2.2.1 Fission and Activation Product (FAP) Bioassay Indicator Radionuclides

SC&A assessed the appropriateness of the methods that NIOSH proposes to use to determine internal doses for areas and times where it asserts that it can reconstruct doses. NIOSH's basis for assigning internal doses for most years and locations not covered by the proposed SEC class and the reserved areas and dates relies on the following four assumptions:

A. <u>FAP Bioassays</u> – Sufficient workers' records containing fission and activation product (FAP) bioassay (in-vitro and in-vivo) results are available to assign intakes and resulting doses from FAP (some periods/areas may need an FAP coworker model developed).

B. <u>FAP Intakes</u> – Except for special situations, all the dosimetric significant FAP intakes are directly tied to an indicator radionuclide (Sr-90 or Cs-137). The FAP ratios and intake assignment methods provided in ORAUT-OTIB-0054 [ORAUT 2015] bound all FAP exposure potentials at INL.

C. <u>Actinide Intakes</u> – Except for special situations, the actinides (uranium, plutonium, thorium, etc.) intakes are directly tied (in a constant ratio) to the FAP; therefore, actinide intakes and resulting doses can be assigned using Table 5-22 (Sr-90 ratios) and/or Table 5-23 (Cs-137 ratios) of ORAUT-TKBS-0007-5 [ORAUT 2010].

D. <u>Special Situations Actinides</u> – Personnel involved in operations and situations (planned or unplanned) with actinides present, that were not directly tied to an FAP in a constant ratio, were adequately monitored, and the results are available in the workers' records. Therefore, actinide intakes and resulting doses can be reconstructed in these special situations.

SC&A has evaluated Item B [application of FAP indicator ratios in Tables 7-3 of ORAUT-OTIB-0054 (ORAUT 2015) to INL FAP intakes], and Item C [application of actinide indicator ratios in Tables 5-22 and 5-23 of ORAUT-TKBS-0007-5 (ORAUT 2010) to INL actinide intakes]. SC&A performed benchmark assessments for some of the ratio values by comparing them to actual measured results. SC&A attempted to assess this situation by searching the INL claim files in the NIOSH/OCAS Claims Tracking System (NOCTS) and locating claims that have FAP and/or actinide exposures with results greater than the minimum detectable activity (MDA), and indicator radionuclide data for workers potentially exposed to intakes from reactor fuels (not intakes of FAP or actinides that have been separated).

The following two sections outline the background for the investigations into Items B and C.

Item B – FAP Intakes Using Indicator Radionuclide Ratios

Both the ER and the INL internal dosimetry TBD (ORAUT 2010) rely on the methodology of ORAUT-OTIB-0054 (ORAUT 2015) to assign FAP intakes using Sr-90 or Cs-137 as the indicating radionuclide for numerous other FAP radionuclides for most locations and time periods. Specifically, often workers' bioassays, air samples, or smears were only analyzed for Sr-90 by gross beta counting, Cs-127 by gross gamma counting, or Cs-137 by whole-body counts (WBCs); other FAPs were not identified or quantified during the process. The general use of ORAUT-OTIB-0054 (ORAUT 2015) for assigning FAP intakes from reactor fuels has been previously addressed and will not be further evaluated here.[1] However, because a wide variety of different reactor fuels were used at INL over the years, it is prudent to determine if there are benchmark examples where the actual recorded data at INL agree or disagree with the values recommended in ORAUT-OTIB-0054 (ORAUT 2015).

Item C – Actinide Intakes Using Indicator Radionuclide Ratios

The ER notes that NIOSH assigns actinide intakes (except for certain special situations) directly tied in a constant ratio to the FAP through Table 5-22 (Sr-90 ratios) and/or Table 5-23 (Cs-137 ratios) of the internal dosimetry TBD (ORAUT 2010). Because this method involves assigning intakes and doses over a long time span and numerous areas/operations, it results in the majority of the alpha internal dose assignments for INL workers. Therefore, it is necessary to evaluate the origin of these data in some depth to determine their applicability to the DR process.

SC&A evaluated the ratio methodology used by NIOSH to determine internal exposures in many cases by employing three different strategies.

1. <u>Claims</u>: SC&A searched the INL claim files on NOCTS to locate claims involving FAP and/or actinide exposures, with results greater than the MDA, and indicator radionuclide data for workers potentially exposed to intakes from reactor fuels (not intakes of FAP or actinides that have been separated). However, search efforts to date of over 6,000 INL claims have not located many claims with these criteria. Further investigation along these lines is needed.

[1] SC&A extensively evaluated ORAUT-OTIB-0054 (ORAUT 2015) for the Board's Subcommittee on Procedures Review. Of the many documents produced relevant to the TIB, a good, technical presentation of how NIOSH modeled several different representative reactors and fuel combinations appears in ORAUT-RPRT-0067 (ORAUT 2014).

2. MDA: SC&A analyzed several scenarios that involved applying the MDA values for Sr-90 or Cs-137 and plutonium (the largest actinide alpha dose contributor) to several claims and some hypothetical cases. This analysis was performed because it might indicate any obvious inconsistencies between plutonium intakes assigned by the Integrated Modules for Bioassay Analysis (IMBA) program using the MDA method, compared to those using the Pu/Sr or Pu/Cs ratios of Tables 5-22 and 5-23 of ORAUT 2010 respectively. Preliminary results using this approach were mixed and inconclusive to date, but should be pursued further.
3. SRDB: SC&A performed some keyword searches on the SRDB for documents containing positive bioassay, air filter, or smear data. SC&A succeeded in locating four relevant documents consisting of a total of approximately 35,000 pages total. The recorded data were analyzed to determine FAP and actinide radionuclide ratios compared to the indicator radionuclide.

These investigations resulted in a number of observations and areas of concern, some of which require further investigation, including the following. This preliminary investigation indicates that for the small sampling of data points in the 1983–1985 time period:

- FAP intakes derived using OTIB-0054 (ORAUT 2015) would be greater than those that would be derived from measured data.
- Pu-238 intakes derived using ORAUT-TKBS-0007-5 (ORAUT 2010), Tables 5-22 or 5-23, would be greater than those that would be derived from measured data for nasal swabs and a smear.
- Pu-238 intakes derived using ORAUT-TKBS-0007-5 (ORAUT 2010), Tables 5-22 or 5-23 would be less than those that would be derived from measured data for CAM filters.
- Am-241 intakes derived using ORAUT-TKBS-0007-5 (ORAUT 2010), Tables 5-22 or 5-23 would be less than those that would be derived from measured data for nasal swabs.

Recommendations: Given these initial observations and areas of concern, we suggest the following path forward concerning Item B (FAP) and Item C (actinides):

- Determine if the burnup in the fuel elements used by NIOSH is applicable/ bounding to the situations at INL.
- Investigate the use of one model and only three fuel elements to bound the intakes/doses.
- Determine if records of analyses of dissolver contents (chopped/shredded fuel elements) are available; preferably, for a variety of INL reactor fuel elements.
- Evaluate NIOSH's recommended ratio value, especially for actinides. Further investigations along these lines may be aided by the electronic bioassay database; even if this database is presently incomplete, paired FAP and actinide bioassays could provide relevant information.

2.2.2 Reactor Modeling

This section examines the validity of the isotopic ratios of ORAUT-OTIB-0054 (ORAUT 2015), regarding whether the parameters of the representative reactor cases of the TIB envelope the operational parameters of the INL reactors, starting with those in the Test Reactor Area (TRA). Future investigations will need to expand to other reactors on the site, such as in Test Area North (TAN). The ER cites OTIB-0054 (ORAUT 2015) multiple times. For example:

> *At TRA, urine samples were typically analyzed only for gross beta, gross gamma, and/or strontium radioactivity. For such samples, NIOSH will assess missed Sr-90 and/or Cs-137 intakes in accordance with ORAUT-OTIB-0054* [ORAUT 2015] *and ORAUT-OTIB-0060* [ORAUT 2007].

NIOSH uses OTIB-0054 (ORAUT 2015) to determine internal doses using indicator radionuclides in cases where only gross beta or gross gamma measurements are available. The general validity and applicability of the TIB has been evaluated by SC&A as part of the Procedures Review Subcommittee review process and all findings were closed. In brief, after initially considering seven reactor cases and after a complicated process involving sets of ORIGEN runs (Croff 1980), the downselect process reduced the cases to four representative reactors:

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

Multiple ORIGEN-S runs [ORNL 2015 – ORIGEN-S is part of the SCALE code system developed and maintained by Oak Ridge National Laboratory (ORNL) for the U.S. Nuclear Regulatory Commission (NRC)] produced a total of nine representative cases. OTIB-0054 (ORAUT 2015), Table 5-2 (reproduced here as Table 2) lists the parameters and bases selected for each of the cases.

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

Case	Parameters	Basis
ATR 1	Specific power = 2,379.1 MW/MTU Irradiation time = 132.27 days Burnup = 314,684 MWd/MTU	Maximum burnup at nominal power.
ATR 2	Specific power = 8,651.2 MW/MTU Irradiation time = 36.4 days Burnup =314,904 MWd/MTU	Maximum burnup at maximum assembly power.
ATR 3	Specific power = 2,379.1 MW/MTU Irradiation time = 56 days. Burnup = 133,230 MWd/MTU	Nominal burnup at nominal power.
FFTF 1	Specific power = 163.8 MW/MTHM Irradiation time = 929.4 days Burnup = 152,230 MWd/MTHM	Maximum burnup at nominal power.

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

Case	Parameters	Basis
FFTF 2	Specific power = 163.8 MW/MTHM Irradiation time = 488.3 days Burnup = 79,984 MWd/MTHM	Nominal burnup at nominal power.
N Reactor 1	Specific power = 10.4 MW/MTU Irradiation time = 114.2 days Burnup = 1,188 MWd/MTU	Production of weapons-grade plutonium (nominal 6% Pu-240 content) at nominal power.
N Reactor 2	Specific power = 10.4 MW/MTU Irradiation time = 285.6 days Burnup = 2,970 MWd/MTU	Production of fuel-grade plutonium (nominal 12% Pu-240 content) at nominal power.
TRIGA 1	Specific power = 15.57 MWd/MTU Irradiation time = 730.1 days Burnup = 11,368 MWd/MTU	Maximum burnup at nominal power.
TRIGA 2	Specific power = 15.57 MW/MTU Irradiation time = 115.2 days Burnup = 1994 MWd/MTU	Nominal burnup at nominal power.

Source: ORAUT 2015, Table 5-2

The TIB indicates that NIOSH will customarily consider all nine reactor cases when doing a DR and, in the absence of individual worker data, might apply data from all four decay times (10 days, 40 days, 180 days, and 1 year), as well, if required by a specific DR case.

Given the parameters of the representative reactor cases shown in Table 2, SC&A is investigating, starting with TRA, whether all the normal and off-normal operating scenarios of the INL reactors are adequately enveloped, so that the ratios given in OTIB-0054 (ORAUT 2015) will be valid. Table 3 lists the TRA reactors, of which the first three, intended to test materials in high neutron flux environments, are by far the largest and most significant.

Table 3. INL Test Reactor Area (TRA) Reactors and Support Facilities

Facility Name	Building No.	First Used	Last Used	Facility Type
Materials Test Reactor (MTR)	TRA-603	1952	1970	Reactor
Engineering Test Reactor (ETR)	TRA-642	1957	1981	Reactor
Advanced Test Reactor (ATR)	TRA-670	1967	In use	Reactor
Reactivity Measurement Facility (RMF)	TRA-603	1954	1962	Reactor
Advanced Reactivity Measurement Facility No. I (ARMF-I)	TRA-660	1960	1974	Reactor
Advanced Reactivity Measurement Facility No. 2 (ARMF-II); renamed Coupled Fast Reactivity Measurement Facility (CFRMF)	TRA-660	1962	1991	Reactor
Engineering Test Reactor Critical Facility (ETRCF)	TRA-654	1957	1982	Reactor
Advanced Test Reactor Critical Facility (ATRCF)	TRA-670	1964	In use	Reactor
Hot Cell Facility	TRA-632	1954	In use	Research
Gamma Facility	TRA-641	1955	ND	Research
Radiation Measurement Laboratory (RML)	TRA-604	1952	In use	Sampling/ Research
Radiochemistry Laboratories	TRA-604	1950	In use	Sampling/ Research
Alpha Laboratories	TRA-652	1961	1970	Sampling/ Research
HB-4 Crystal Spectrometer	TRA-603	1952	1970	Research
Fast (Neutron) Chopper	TRA-665	1952	1970	Research

SC&A's preliminary examination of the operating scenarios of the three main, high-power reactors and their fuel, burnup, and power level combinations indicates that the use of the OTIB-0054 (ORAUT 2015) methodology and values appears appropriate.

Recommendation: SC&A believes that further study should be done for off-normal operations, including "special" materials irradiation runs, and for any incidents that might have occurred and exposed personnel to radiation. In addition, this line of investigation should be extended to other reactor areas, especially TAN, which hosted very unusual reactor experiments, such as the Advanced Nuclear Propulsion (ANP) reactors, which were decidedly different in fuel composition and arrangement and operation than other types of reactors.

2.2.3 Burial Grounds

While mixed fission products (MFPs) and Rocky Flats Plant (RFP) plutonium undoubtedly did dominate the radioactive waste being managed at the Burial Grounds, the actual radionuclide content of specific onsite and offsite solid waste being handled at any given time was not normally known. For example, the onsite NRTS waste was nominally described as "mixed fission products," but could consist of a variety of radioactive constituents (SRDB Ref ID: #138957). Likewise, for RFP wastes, shipments did not include paperwork describing physical form and radionuclide content until 1964 (RWMC 1985). For nationwide commercial and military radioactive waste received at the Burial Grounds in 1960–1963, it was found that these offsite shipments arriving at the Burial Grounds did not have documentation that identified the waste container contents, and proposed shipments "seldom, if ever, [were] the same as the shipment itself" (SRDB Ref ID: #138704).

Health Physics (HP) personnel were assigned to the Burial Grounds to perform monitoring, but also fulfilled a prime operational role for managing the radioactive waste disposal at the Burial Grounds. This operational role encompassed initiating and receiving work requests for Burial Grounds operation, reporting on costs for operations to the Atomic Energy Commission (AEC), developing and recording plot plans for pits and trenches, witnessing and maintaining a log of all burials, in addition to routine radiological surveillance functions (Burial Grounds 1962). This "dual" responsibility of Idaho Nuclear Corporation's (INC's) Health and Safety Branch for both operations and safety of the Burial Grounds was found to be an organizational conflict of interest and led to a lowered performance rating for INC in an audit conducted by the AEC in 1971.[2]

Safe work permits were prepared for dumping operations, but a cursory review of those for RFP waste disposal from the 1960s shows a wide range of work controls prescribed—from no controls or precautions identified (except for steel-toed guards) to a wide range of provisions calling for tool checks, hand and foot counting, intermittent surveying, and final monitoring with an alpha survey meter (SWP 1962–1964; SRDB Ref ID: #141608). Some of the interview accounts by Burial Grounds operators (backed up by contemporary photographs) suggest little in

[2] Due to leaking barrels received at the Burial Grounds from RFP in 1970, the AEC requested that Burial Grounds operations be suspended pending an independent internal review of HP procedures, which found this organizational conflict to be a contributing problem (INC 1971).

the way of protective clothing except for a hardhat and coveralls. The requirements cited in the permits seem to be more related to the HP writing them than to the job, itself.

In terms of contamination control during operations, former Burial Grounds workers noted that when RFP TRU waste drums were routinely dumped from trucks into waste pits, the drum lids would sometimes pop off, leading to scattering of contents throughout the pit (SRDB Ref ID: #141861). An NRTS-wide road contamination survey in 1969 found the following:

> *The number and intensity of the contaminated areas along the route to the Burial Ground indicates inadequate handling procedures and is a matter of concern. It appears that either waste shipped to the Burial Ground has been improperly packaged or waste unloaded at the Burial Ground was carelessly handled causing contamination of the returning vehicle and then the roadway. Surveys of vehicles and equipment leaving the burial area should indicate if the latter is occurring.* (SRDB Ref ID: #125523)

This 1969 survey found contamination by various fission product nuclides, including Cs-137 and Co-60, with measured dose rates (at contact) as high as 20 and 30 R/hour.

By their own admission, the capability of the contractor to even administer a valid contamination control program is questionable for the Burial Grounds. In an April 5, 1972, internal memorandum within Aeroject Nuclear Company (ANC) (ANC 1972), the HP organization identified poor and outdated instrumentation due to lack of funding as undercutting its program. It is noted that a longstanding problem existed with outmoded HP smear counters at CFA, with no smear counting equipment at the Burial Grounds before February 1972. This was considered to be "a completely intolerable situation which must be corrected as soon as possible if we [ANC HP] are to avoid a serious contamination incident."[3] The HP organization further stressed that smear counting was the only capability available to them to detect "loose" contamination and to differentiate between beta-gamma and alpha contamination. In its final weighing of potential consequences related to the lack of this equipment, ANC HP makes the following assessment:

> *The Burial Ground has been operating for years without any smear counting equipment and has avoided serious contamination incidents by luck and/or by the experience of well-trained health physicists. The Burial Ground operation cannot comply with present radioactive on-site shipping regulations because of a lack of proper detection equipment. With increased emphasis on contamination control, it is absolutely imperative that state of the art simultaneous smear counting equipment be purchased for the Burial Ground as a minimum.* (ANC 1972)

In conclusion, the ANC HP manager concludes that, "without smear counting equipment, or outdated and inadequate smear counting equipment, there is, at best, only a token effort of contamination control."

[3] It was noted that as an interim measure, at that time in 1972, the ANC HP office would be deploying an Eberline RM-14 with HP 210 probe to count smears at the Burial Grounds, and would seek a surplus beta-gamma counter as a stop-gap measure.

A number of concerns relevant to the SEC petition emerge from the material quoted above, as well as from the SRDB references cited below:

(1) It is questionable whether a "strict" contamination control program existed at the Burial Grounds, given evidence suggesting a haphazard approach to limiting contamination when dumping transuranic (TRU) waste drums, inadequate HP counting instrumentation, and apparent lax surveying of personnel and vehicles leaving the burial site (ANC 1972; SRDB Ref IDs: #141861, #125523).

(2) The apparent lack of a suitable smear counting capability at the Burial Grounds would have removed or severely impaired a key "workplace indicator" for indicating a potential intake and the need for a "special" bioassay. It is also not clear whether the HP program was sufficiently robust and independent at the Burial Grounds to support the necessary onsite surveillance to detect potential intakes. Pending the completion of an electronic database for INL bioassays, it is not possible at this time to ascertain the adequacy and completeness of that database and to what extent "special" bioassays figure in it (ANC 1972; Burial Grounds 1962).

(3) Radioactive waste was not specifically identified for most drums, boxes, and other containers received at the Burial Grounds for disposal. A substantial amount of offsite waste was received from commercial, university, Energy Research and Development Administration (ERDA) and military sources when the NRTS was a national radioactive waste site; some of this waste was identified as to radionuclide content and activity levels, but much of it was not. It is not clear whether a suitable source term can be derived for what workers may have been exposed to from the standpoint of internal intakes and uptakes, and whether such exposures can be bounded by proposed NIOSH methods (INEL 1977; RWMC 1985; SRDB Ref IDs: #138957, #138704, #141859).

(4) The ER relies on a programmatic basis, i.e., strength of the HP program, to support its contention that no plutonium intake would have gone unmonitored. However, a number of investigations, program appraisals, and internal communications related to the Burial Grounds over the years have found fundamental shortcomings in that very program, particularly as it pertains to the detection, monitoring, and control of contamination. This record shows the AEC's concern over the conflicted role of health physicists at the Burial Grounds, who were essentially responsible for much of its operation, as well as radiation protection. It also finds lack of management support for the Burial Grounds, generally, and to funding needed for contamination detection equipment, specifically. While some of these findings come from documents just after the 1970 cutoff in the current ER version, they nonetheless represent significant program deficiencies that transcend time periods at the Burial Grounds and raise serious doubt about the so-called "defense-in-depth" approach taken by RWMC in the 1950s–1970s. In summary, given this checkered radiological program history, a programmatic basis alone is not sufficient to claim RWMC historic practices would have precluded any unmonitored plutonium uptake in the early years up to 1970 (ANC 1972; Burial Grounds 1962; SRDB Ref IDs: #141861, #138957, #141876).

The NIOSH ER concludes that worker exposures at the Burial Grounds can be dose reconstructed for 1952–1970 on the basis of stringent contamination controls, a radiation control program exemplifying a "defense-in-depth" approach, and available internal dose data for known radioactive waste source terms that lend themselves to standard DR methods [e.g., OTIB-0054 (ORAUT 2015) and OTIB-0060 (ORAUT 2007)]. SC&A believes all of these basic tenets are suspect, given a preliminary review of available SRDB documentation.

Instead, SC&A believes available evidence indicates that the Burial Grounds was considered a low priority by INL management; was so underfunded that needed HP instrumentation was lacking; apparently lacked a management culture that supported disciplined operations and a formality of radiological controls to minimize unnecessary contamination; and dealt with high exposure potential MFPs and TRUs that were unidentified as to specific isotopic content, activity levels, and container content. From worker interviews, radiological incidents, and photographs of dumping operations, it is clear that an exposure potential existed for waste handlers. It is also clear that any resuspended contamination may not have been detected and necessary bioassay follow-up would not have occurred, given the management deficiencies at the site and the apparent lack of smear counting equipment available for Burial Grounds activities.

Recommended Areas for Further Inquiry:

(1) Investigate whether additional evidence (beyond worker interviews, incidents, and photographs) exists to support potential exposure of Burial Grounds waste handlers to radionuclide intakes. Were HP program practices, instrumentation, and dosimetry adequate to detect and identify intakes and uptakes?

(2) Investigate how contamination control was administered at the Burial Grounds during the period 1952–1970. Were surface smears taken of personnel, equipment, and containers to identify loose contamination, and if so, how was that program affected by the apparent lack of smear counting equipment at the Burial Grounds (as identified in 1972)? Was alpha contamination given attention?

(3) Investigate whether routine and special air sampling data are available for the Burial Grounds that can characterize the airborne particulate concentrations of MFPs, TRUs, and other resuspended contamination during disposal operations. Was there any air sampling performed after barrels or other containers broke open during disposal?

(4) Investigate the effectiveness of the management of waste disposal operations at the Burial Grounds during this period, particularly the radiological controls program (i.e., did it constitute a "defense-in-depth" approach)? Was there any evidence of operational priorities taking precedence over those of radiation protection (HP)?

(5) Investigate the bioassay monitoring history at the Burial Grounds. When were routine in-vivo and in-vitro bioassays conducted for Burial Grounds workers and what "special" bioassays were conducted and for what reason? Is there a complete and adequate bioassay database to support dose reconstruction? Is it feasible to identify all INL workers who handled radioactive waste at the Burial Grounds (RWMC); what proportion of these workers were outside contractors or subcontractors?

(6) Investigate whether information exists that enable source term characterization of all radioactive waste disposed in the Burial Grounds during 1952–1970. What radionuclides or mixed radioactive waste would not be covered under OTIB-0054 (ORAUT 2015) or

OTIB-0060 (ORAUT 2007)? How will unknown radioactive wastes be addressed in terms of dose assessment? Would MFPs, as a source term, dominate any other potential internal radiation exposure at the Burial Grounds?

2.2.4 Central Facilities Area (CFA)

SC&A conducted a review of the SEC Petition Evaluation SEC-00219 concerning radiation exposures and monitoring data available for the CFA at INL. NIOSH proposes bounding the internal doses from fission products by relying on Sr-90 and/or Cs-137 intakes in accordance with ORAUT-OTIB-0054 (ORAUT 2015) and ORAUT-OTIB-0060 (ORAUT 2007). Potential intakes of other radionuclides (i.e., uranium) in the presence of MFPs would be estimated on a case-by-case basis using the approach described in ORAUT-TKBS-0007-5 (ORAUT 2010). NIOSH believes this approach is sufficiently accurate to estimate the internal doses for CFA workers during the period from January 1, 1953, through December 31, 1970.

According to the SEC, routine in-vitro monitoring was initiated in 1953, consisting of single-void urine samples. Gross beta analyses began in 1953, followed with gross gamma analyses beginning in 1957 (ORAUT 2010). According to Table 7-14 of the ER, in-vivo (WBC) monitoring did not begin until 1961. Routine in-vivo and in-vitro analyses were not performed for alpha-emitting radionuclides.

SC&A has identified four CFA facilities that may not be suitable for the internal dose bounding approach proposed by NIOSH. They are listed in Table 4.

Table 4. CFA Facilities of Concern

Facility Number	Facility Name	Activity Description
CF-640	**Machine Shop**	Radioactive material that the plant shops could not handle was worked on in the more fully equipped CFA Machine Shop. Usually this material was of a low radiation and contamination level.
CF-665	**Maintenance Shop**	The Maintenance Shop personnel would sometimes work on vehicles and equipment that were used to haul radioactive material. Such vehicles were surveyed prior to shop maintenance work and, when necessary, sent to CPP for decontamination.
CF-669	**Central Facilities Laundry (constructed in 1950)**	The laundry washed coveralls and other protective clothing items that were used in radiological work. The laundry drain went to a septic tank and drain field with other sanitary waste. The laundry facility and drain field(s) are sources of low-level radioactive contamination, which covers the spectrum inherent to work in radiological contamination areas. The old laundry facility (CF-699) was used from 1950 and was demolished in 1994. The new laundry facility (CF-617) was used from 1981 to 2001 and demolished in 2002.
CF-674	**Sewage Treatment Plant**	In the late 1950s and into the 1960s, small amounts of radioactivity were processed through a Sewage Treatment Plant, CF-674, to a drying pond. Most of the radioactivity was from the hot laundry, although small amounts could enter from CF-656 and CF-690.

Each of the facilities listed in Table 4 would handle/process radioactive materials from the entire INL site, consisting of MFPs, mixed activation products (MAPs), actinides, or a mixture of any or all. For this reason alone, it would be difficult to use Sr-90 and/or Cs-137 ratios to determine MFP and MAP intakes, and not practical for actinide intakes, due to the uncertainty of the source term.

SC&A conducted a limited search of the SRDB for operations and survey information regarding the buildings identified in Table 1. Very little information was found concerning the operation and conditions in Buildings CF-640, CF-665, and CF-674. Several interoffice memos, reports, and radiological surveys were found for the Hot Laundry, CF-669. The concerns listed below, while specific to the CF-669 Hot Laundry, also apply to each of the other facilities in Table 4.

Bioassay Monitoring

There appears to be a lack of consistent and regular bioassay monitoring for MFP/MAP and no monitoring for uranium or other actinides. The routine urine bioassay sampling frequency was at most annual, and it is not clear if each employee was sampled annually or if a random group was sampled and considered to be representative of the workforce. The laundry was contaminated with both beta/gamma and alpha contamination, as indicated in the 1953–1956 contamination surveys (SRDB Ref ID: #139224).

Air Monitoring/Ventilation

The quality and adequacy of the air monitoring and ventilation in CF-699 is also questionable. According to an interoffice memo, "Laundry Improvements," from November 19, 1971 (SRDB Ref ID: #143310):

> *The contaminated air that is being released to the atmosphere from the five clothes dryers is of immediate concern. These dryers have two sets of lint removers, both of which are grossly inadequate for preventing contaminated lint or particles from escaping. Using an RM-14 Eberline instrument, we surveyed the first stage lint screens and the second stage lint screens and found contamination to 50,000 and 30,000 c/m, respectively. Using the same instrument we found the roof contaminated to a maximum of 25,000 c/m with an average of about 2000 c/m. The roof contamination appears to be on escaped lint.*

The same document also states:

> *The constant air monitor (CAM) is worn out.*
>
> *There is no designed ventilation control system. Natural ventilation is wholly inadequate for this type of facility.*
>
> *The areas handling "hot" laundry should be kept at negative pressure relative to the "clean" area.*
>
> *Hood ventilation should be provided for sorting incoming laundry.*

Contamination Control/Monitoring

Concerns also exist regarding the quality and adequacy of contamination control and monitoring methods. According to a "CFA Health Physics Monthly Report for July 1961" (SRDB Ref ID: #139258), bagged laundry arrives without the proper contamination identification:

> *Irregularities still occur at the radioactive laundry. Personnel become apprehensive when clothing comes in without tags or without Form ID-110 indicating the contamination levels.*

There is also a concern with the contamination detection equipment. It was noted in the 1971 memo (SRDB Ref ID: #143310): "Table monitors and clothing monitors are obsolete and inadequate. Proximity to "hot" laundry causes excessive background."

On a more general note regarding the state of the HP equipment, an interoffice memo dated April 4, 1972 (ANC 1972) states in part: "The instruments now being used at the Central Facilities Health Physics office are outdated because of the reasons enumerated above, and because of their unreliability."

According to the April 4 memo, the HP smear counting equipment in use at the time (1972) consisted of a beta-gamma smear counter acquired in 1957, a beta-gamma smear counter acquired in 1958, and an alpha smear counter from 1952. The quality and reliability of the equipment in use impacts the quality and reliability of the contamination smear data from buildings, vehicles, shipments, and personnel monitoring in the CFA.

SC&A believes that because of the concerns listed above and that CFA facilities handled radioactive materials from the entire INL site, it would be difficult to use Sr-90 and/or Cs-137 ratios to determine MFP and MAP intakes, and not practical for actinide intakes due to the uncertainty of the source term.

SC&A recommends that an evaluation should be conducted of the CFA radiological survey and air sampling results both during operations and just prior to decontamination and decommissioning (D&D) to determine actinide to Sr-90 and actinide to Cs-137 ratios. These ratios should then be compared to the values in Tables 5-22 and 5-23 of ORAUT 2010.

2.2.5 Chemical Processing Plant (CPP) Pre-1963

As noted in Section 1, the current SEC class definition as proposed by NIOSH includes all externally monitored workers at the CPP from January 1, 1963, to December 31, 1974. The period subsequent to the proposed SEC is still being evaluated by NIOSH, with a focus on the practical implementation of improved radiological safety policies beginning in 1975. For the period prior to the proposed SEC (pre-1963), NIOSH believes that sufficient information and monitoring records exist to reconstruct both internal and external doses to all relevant workers at the CPP. This section presents SC&A's preliminary concerns and progress on evaluating potential SEC issues at the CPP prior to 1963.

Table 5 (reproduced from Table 7-5 of NIOSH 2015) shows NIOSH's assessment of the feasibility for internal DR in different sources and time periods for CPP. Prior to 1963, NIOSH has determined internal doses can be reconstructed for MFPs, plutonium, neptunium, uranium, thorium, and any other relevant TRUs. The table also indicates that no coworker models are necessary prior to 1962.

Table 5. Feasibility Summary for CPP (1953–1974)

Exposure Source	Years																					
	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74
MFP	F	F	F	F	F	F	F	F	F	F	F	F	F	F	C	C	C	C	C	C	C	C
Pu/Np	F	F	F	F	F	F	F	F	F	F	I	I	I	I	I	I	I	I	I	I	I	I
Uranium	F	F	F	F	F	F	F	F	F	F	I	I	I	I	I	I	I	I	I	I	I	I
Th/Transuranics	F	F	F	F	F	F	F	F	F	F	I	I	I	I	I	I	I	I	I	I	I	I
Other	F	F	F	F	F	F	F	F	F	F	F	F	F	F	C	C	C	C	C	C	C	C
Photon	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
Neutron	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F

F = Dose reconstructions are feasible.
C = Dose reconstructions are feasible, but a mixed-fission-product coworker model is needed.
I = Dose reconstructions are infeasible.
Source: NIOSH 2015, Table 7-5

Page 189 of NIOSH 2015 summarizes the rationale for the current SEC class definition as follows:

> *Increased potential for intake due to poor contamination control and inadequate personnel monitoring for exposures to transuranics separated from mixed fission products makes it unlikely that exposures to alpha emitters can adequately be reconstructed from January 1963 through December 1974.*

Therefore, SC&A's preliminary investigation into periods at CPP prior to 1963 focuses on two main facets:

1. Characterization of contamination events and contamination control prior to 1963 and the associated implications concerning chronic exposure potential to CPP workers.
2. Assessment of the available internal dosimetry program and its ability to capture and bound the potential internal exposure to CPP workers. This is particularly important in the assessment of internal exposure potential to alpha-emitting TRU isotopes.

With regard to Item 1, SC&A began by inspecting relevant HP reports, which often contain summary characterizations of contamination identified within the CPP. In particular, SC&A has tabulated a preliminary data compilation concerning the number of contaminated surfaces identified by month at the CPP for periods prior to the proposed SEC class definition. This is shown in Figure 1. As seen in the figure, there are periods prior to 1963 in which there were a significant number of identified contamination events. In some cases, the number of contamination events appears to be higher than the first couple years of the SEC period. SC&A has not yet established information as to the magnitude and type of contamination events prior to 1963.

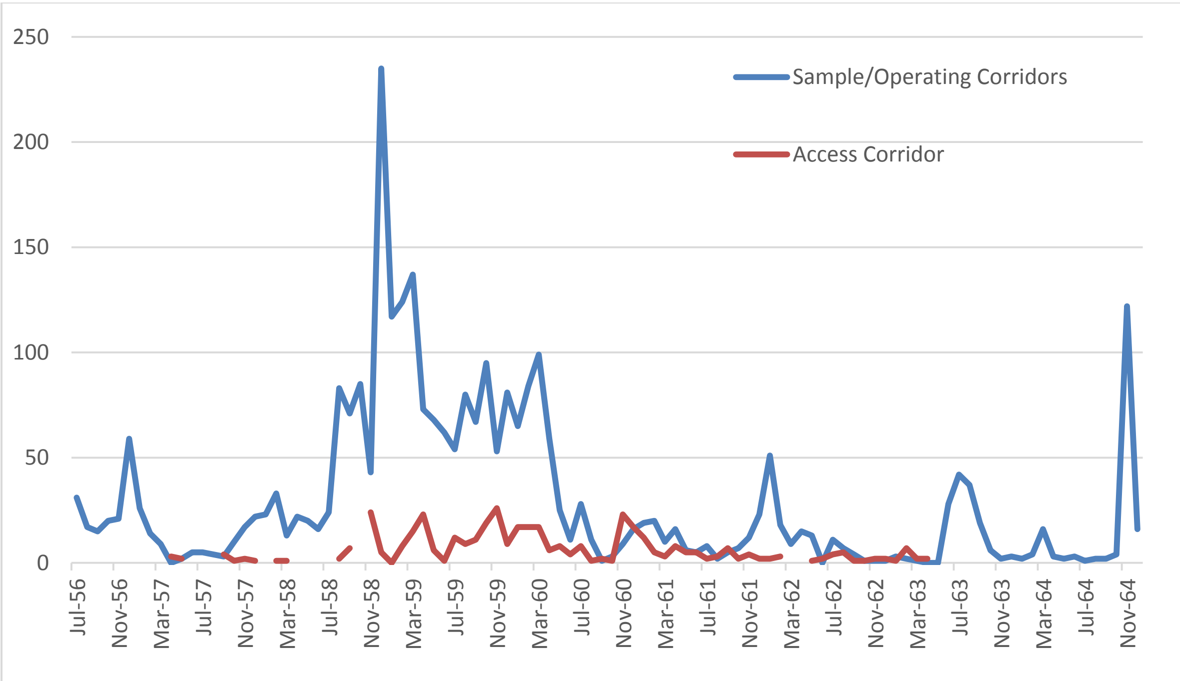

Figure 1. Number of Contaminated Surfaces in CPP-601 Sample/Operating Corridors and Access Corridor

In addition to the standard periodic HP reports, SC&A has begun investigating specific incident reports in which contamination was discovered. In particular, information is being reviewed and compiled that can be used to characterize the magnitude of contamination, nature of event (off-normal or "unusual" events versus more common events), decontamination procedures undertaken, special and/or routine sampling, and steps taken to mitigate the occurrence of future events.

With regard to Item 2, SC&A has begun investigating the files of claimants who have direct evidence of work at CPP prior to 1963 in order to characterize information concerning:

- Internal personnel monitoring frequency and type (in vitro, in vivo, nasal swipes, etc.)
- Evidence of contamination events or incident descriptions [includes both relevant Computer-Assisted Telephone Interview (CATI) information, as well as Department of Energy (DOE)-supplied monitoring records]
- Any other relevant information pertinent towards characterizing the internal exposure potential and monitoring program at CPP prior to 1963

In addition to the review of claimant-specific records, hardcopy bioassay records exist on the SRDB that will help characterize the internal monitoring program as it relates to exposures at CPP. Currently, these bioassay data only exist in a complete form as hardcopy records; however, a partial electronic compilation of such records is available. Such a compilation is somewhat useful in making preliminary judgments as to the direction of future investigative lines of inquiry, with the stipulation that such records are not currently complete, but will likely become complete at a future date.

Recommendations: The combination of information gathered for the items described above will allow SC&A to evaluate the extent to which internal exposure potential existed for the relevant worker population at CPP, as well as the sufficiency of the internal monitoring program at CPP to capture all relevant exposures. SC&A recommends that the following review actions continue to fully assess the ability to reconstruct internal doses at CPP prior to 1963:

- Continue research and data collection to characterize the number, location, and magnitude of contamination events/incidents at CPP prior to 1963.
- Evaluate actions taken both prior to, during, and after contamination events/incidents to assess the radiological control program in place at CPP during the earlier period.
- Assess the extent to which chronic lower-level contamination may have existed at CPP (beyond noted incidents and contamination events) that may have presented a long-term chronic internal exposure potential to CPP workers.
- Review individual claimant dosimetry and other records available for CPP workers prior to 1963 to characterize the extent of the routine and incident-based dosimetry program as it relates to internal doses from all relevant contaminants present at CPP.
- Evaluate current hardcopy bioassay data, to the extent feasible, for completeness and adequacy bounding internal doses at CPP. Additionally, evaluate any completed electronic database records as they become available.
- Research changes in production processes and amount of materials processed that could provide an indication of the type of contamination that was occurring.

2.2.6 Test Area North (TAN)

Because of the complexity of the operations at TAN, it is informative to establish a work breakdown structure based on the different operations and sub-operations that occurred at that site. The following is an attempt to create such a structure, which can be used to help organize an approach to evaluate data completeness, data quality, and data adequacy for the broad range of operations, research activities, and campaigns that took place at TAN.

- Aircraft Nuclear Propulsion Program (ANP) (1952–1961)
- Initial Engine Test (IET) – This may be a subset of ANP and also a stand-alone program
    - Heat Transfer Reactor Experiments
- Technical Support Facility (TSF)
    - TAN 607 Hot Shop
    - LOFT (TAN 650)
    - Storage Pool
    - Storage Pads (TAN 790 and 791)
    - Radwaste Liquid Disposal System
    - Storage Building
    - Radiography Facility (TAN 607)

- Water Reactor Research Test Facility (WRRTF)
  - Low Power Test Facility (LPTF)
  - Shield Test Pool Facility (STPF)
- Specific Manufacturing Capability (SMC) – DU machining for armor

Table 5-4 in the SEC ER provides a breakdown and description of each of these facilities and associated radiological issues. Our initial overarching observations regarding the external dosimetry programs at these facilities are favorable, with the possible exception of neutron dosimetry. Possible concerns regarding the internal dosimetry program as described above also apply to the TAN facilities.

Given (1) the complexity of the activities at TAN, (2) the attention that has been given to the internal dosimetry programs at INL in general, and (3) the apparent quality of the external dosimetry data collected at TAN (with the possible exception of neutron dosimetry), this section of the report focuses on an initial review of the external dosimetry records for TAN, as part of a larger effort to review the SEC ER.

The following analysis was limited to records available on the SRDB. This investigation focuses on determining whether the external dosimetry data for workers at TAN are of sufficient quality and completeness. The level of detail given regarding the various types of dosimeters used, as well as calibration practices in ORAUT-TKBS-0007-6 (ORAUT 2011), indicates that the external dosimetry data appear to be high quality. Overall, it was seen that the external dosimetry records for TAN as a whole (including all sub-areas) appear to be fairly complete, spanning from mid-1955 through part of 1970 with a gap from June to December of 1961. However, when attempting to separate the data into the various subdivisions of TAN, the data do not appear to be very complete, or the labeling of the records does not always provide information that can place a worker at a given facility within TAN at the time of the readout. The available dosimetry records often indicate a sub-area of TAN, but very limited dosimetry records were found for some areas (e.g., STPF). Also, the dosimetry reports do not indicate specific buildings within TAN for workers, only a general area.

In order to begin investigations into external data completeness at the various TAN facilities, SC&A performed numerous searches within the SRDB in an effort to capture as many documents as possible that contained dosimetry information. Examples of search terms used include dosimetry, dosimeter, external, personnel, badge, and film. For each of these documents, the date of each dosimeter cycle, the areas or area codes mentioned, and the number of neutron badges were recorded. In some instances, the document would primarily contain dosimeters for one area (e.g., ANP), but some workers would have multiple badges that may include other areas or area codes (e.g., STPF). Any records of this type were noted and recorded in this analysis.

<u>Characterization and Commentary on Neutron Dosimetry Data</u>

The manner in which neutron badges were reported in these dosimetry documents differs depending on the year and format of the report. In the older records, neutron badge readouts were handwritten in the "remarks" section of the report. Within these types of records, this notation would sometimes have the word "neutron" or just an "n" before the dosimeter reading of “xx mrem.” Therefore, when the remarks section would only say “xx mrem,” it was also taken to mean a neutron badge reading. Other neutron badge data were formatted with separate columns for neutron results and were thus much easier to read. These types of records were primarily seen for the early to mid-1960s. The final way in which neutron badges were presented had summary pages for each badging period, which helped to infer which columns in these reports corresponded to beta, gamma, and neutron readings. Even though there are 00s reported in the neutron column in some of the records, these cannot be taken to mean a zero neutron reading. It was found that if a worker had a neutron badge reading in a given period, then a second set of numbers would be present in the neutron column in the rows associated with that worker.

Commentary on Completeness of External Dosimetry Data at TAN

Table 6 gives information about each document containing detailed dosimetry records that were analyzed for this initial completeness analysis. The number of badges per document was taken to be an estimate of the average number of badges per page, multiplied by the number of pages in the document. These are likely to be over-estimates, as it was found that some pages within the documents were summaries, had zero entries, visitor badges, or contained badges that were not read out, often accompanied by an irregularity code of 14 (Not in area). The number of neutron badges in each document only takes into account those reported for badges associated with an area of TAN, and were not from visitor badges.

Table 6. Documents Found for TAN in SRDB with Detailed Dosimetry Records

SRDB Reference ID #	Area(s) Mentioned	# Pages	# Estimated Badges	# Neutron Badges
116020	IET/STEP	535	8025	226
116021	TAN GE (ANP)	376	3760	231
116022	TAN GE (ANP)	387	3096	185
116023	TAN GE (ANP), LPTF[4], STPF[1]	125	2625	0
116030	TAN GE (ANP), STPF[1]	264	4488	362[5]
116031	TAN GE (ANP)	561	5610	396
116032	TAN GE (ANP)	373	3357	201
116033	TAN GE (ANP)	612	5508	312
116034	TAN GE (ANP)	337	3033	125
116035	TAN PPCo, TSF	525	9450	3
116036	TAN GE (ANP)	464	4176	294
116037	TAN GE (ANP)	393	3930	250
116038	TAN GE (ANP)	483	4830	106
116039	TAN GE (ANP)	476	3332	117
141636[6]	TAN GE (ANP), LPTF, IET/STEP	24	264	0
119002	TSF	671	6710	2
119005	GE (ANP)	213	5325	309

[4] These areas are present in some workers’ multiple badges within a report for a different area, not as a separate report for the area.

[5] All neutron badges are associated with workers’ badges for STPF.

[6] Document is titled as an ‘excerpt;’ some pages are contained within other documents listed.

Table 6. Documents Found for TAN in SRDB with Detailed Dosimetry Records

SRDB Reference ID #	Area(s) Mentioned	# Pages	# Estimated Badges	# Neutron Badges
119006	GE (ANP)	182	5460	173
119017	GE (ANP)	221	5525	50
119018	GE (ANP)	309	4635	477
119019	GE (ANP)	141	2115	280
119020	GE (ANP)	233	6990	319
119030	GE (ANP)	286	7150	138
119031	TSF	168	2688	1
119032	TSF	493	12325	0
119033	TAN PPCo	474	11850	2
119034	GE (ANP)	236	5900	424
119035	GE (ANP)	117	2925	236
119036	GE (ANP)	388	9700	645
119037	GE (ANP)	166	4150	239
119038	GE (ANP)	341	8525	101
119039	GE (ANP)	241	4820	184
119040	GE (ANP)	474	11850	55
119041	GE (ANP)	293	7325	409
119042	GE (ANP)	405	10125	175
119043	TSF	73	1460	0
119044	TSF	117	1872	0

Overarching Observations Regarding the Completeness of the External Dosimetry Data at TAN

In order to assess the completeness of data for TAN as a whole, the dosimeter dates for all of the documents listed in Table 6 were combined, regardless of the TAN sub-area the dosimeters might have been associated with. Figure 2 is a graphical representation of the dates for which dosimeters exist for an area of TAN that are contained within the analyzed SRDB documents. The y-axis also represents the year of the dosimeter, such that the data could be broken up by year to allow for easier identification of temporal gaps in the available records. The points that make up the section of Figure 2 for 1956 are shown in more detail by themselves in Figure 3. What looks like an irregularity in the dosimeter dates for 1956 could be attributed to non-routine readouts; for example, if a badge was designated "late-pull" or "HP request," then a date usually different than those of the typical dosimeter cycles was recorded.

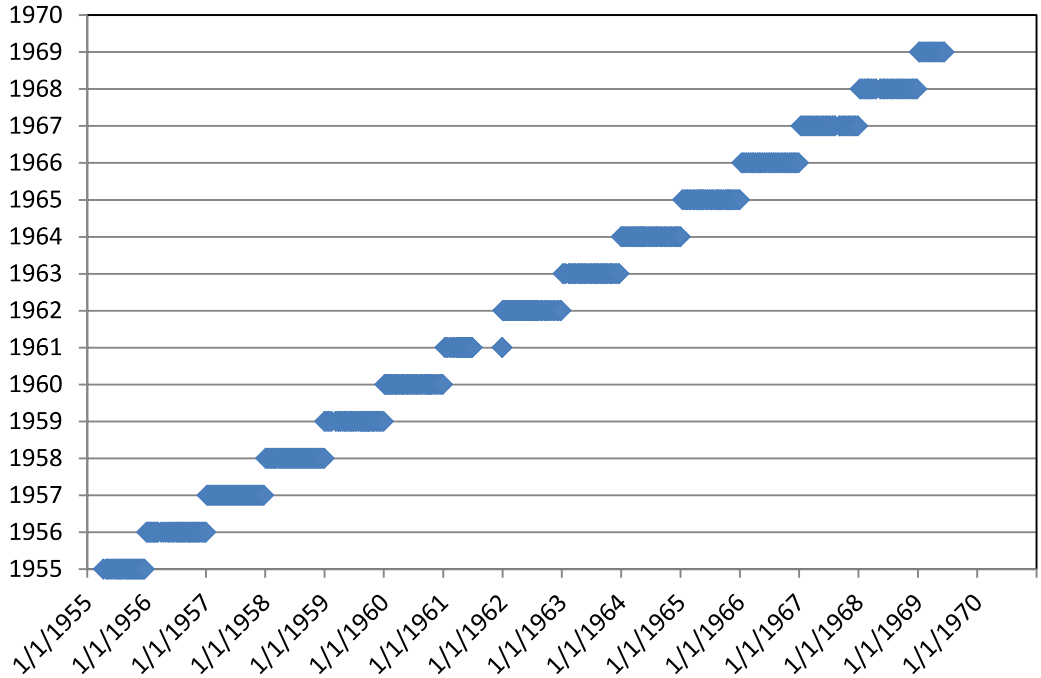

Figure 2. Dates of Dosimeters Available in SRDB Documents for All of TAN

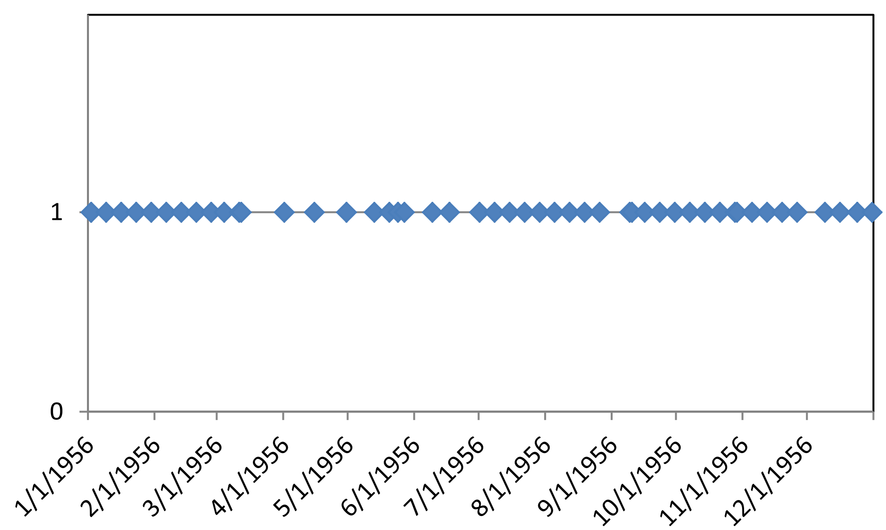

Figure 3. Dates of Dosimeters for 1956 taken from Figure 2

Overall, most years appear to have dosimeter cycles that reasonably cover that entire year, with the possible exceptions of 1955, 1961, and 1970. The earliest dosimeter cycle within the documents was dated April 10, 1955. The available dosimeter records in 1961 do not cover June 18 through December 27. Dosimeters for the month of August in 1970 were also not found. In SRDB Ref ID: #119002, the dosimeters appear to be reported on a monthly basis, but skip from July 31 to September 1, 1970. Investigation into whether the records for these gaps exist may be necessary. It is also possible that the records exist in the SRDB, but were not captured during the searches.

Completeness of External Dosimetry data for Sub-Areas of TAN

Due to the fact that most of the dosimeter records either had an area written into its header or had area codes associated with the dosimeters, an effort was made to see if the data are complete enough to break the dosimetry records into sub-areas of TAN. Figure 4 illustrates the dates of dosimeters available for various sub-areas of TAN within the analyzed SRDB documents.

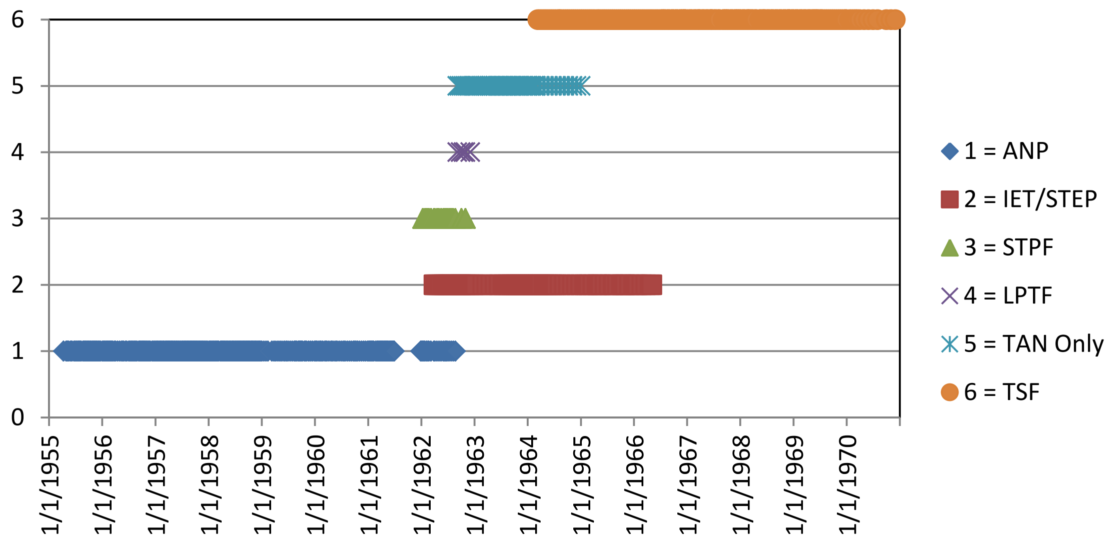

Figure 4. Dates of Dosimeters for Various Sub-Areas of TAN

Other Evidence of Badging

While searching the SRDB for dosimetry records, many documents were found that contained monthly or annual summary reports, as well as records containing only visitor badges. Considering TAN as a whole, Figure 5 shows the number of each type of document found in the SRDB that covers at least part of a given year. If a document covered multiple years, it was counted for each year. It can be seen that summary and/or visitor badge reports are available outside of the time period for which detailed records are available (1955–1970). Table 7 presents a list of the summary and visitor reports found.

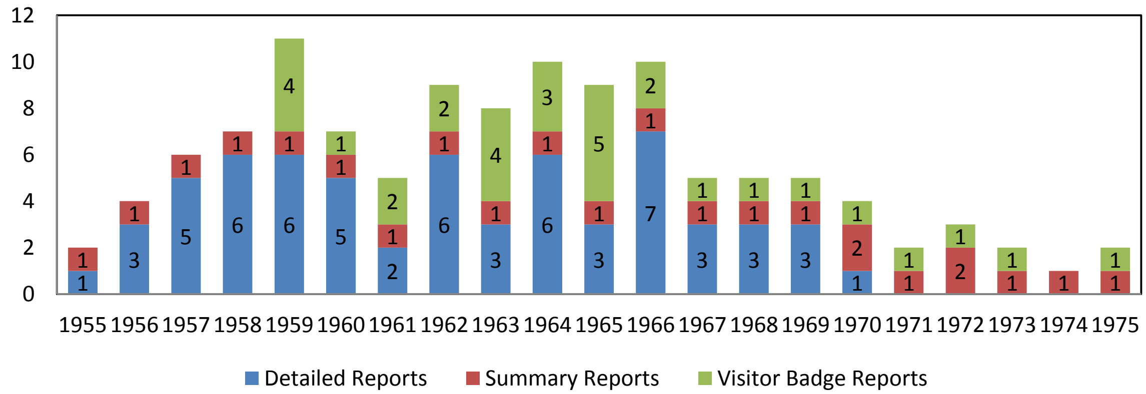

Figure 5. Number and Type of Documents Found in the SRDB According to the Year(s) They Cover

Table 7. Visitor or Summary Documents Found for TAN in SRDB

SRDB Reference ID#	Summary Report or Visitor Badge Report
116024	Visitor
116025	Visitor
116026	Visitor
116027	Visitor
116028	Visitor
116029	Visitor
116040	Visitor
116041	Visitor
118967	Visitor
118970	Visitor
118982	Visitor
118983	Visitor
118989	Visitor
118990	Visitor
116042	Visitor
118965	Visitor
118968	Visitor
118985	Visitor
118986	Visitor
118987	Visitor
118988	Visitor
138150	Visitor
138152	Visitor
138153	Visitor
125917	Summary
125913	Summary
125915	Summary
125920	Summary
125924	Summary
125926	Summary
125927	Summary
125928	Summary
125930	Summary
125934	Summary

Summary Reports

Several of the summary reports give the number of badges, pencil dosimeters, neutron badges (A-film), as well as other types of dosimeters read for a given month for various areas of INL. In one summary report (SRDB Ref ID: #125917), the first mention of an area associated with TAN was in April of 1955 (as ANP). This coincides with the earliest dosimeters associated with ANP, which were also in April of 1955. Use of these summary reports may be useful regarding the assessment of the completeness of the available dosimetry records. For example, in SRDB Ref ID: #125917, the number of neutron badges recorded for October of 1957 for ANP is reported as 186 (p. 236). However, the number of badges counted within the analyzed detailed reports for ANP during this month is 175. Further investigation and comparison between these records might be necessary to know whether more dosimetry records exist, and what types of workers they are attributed to. Some of the summary reports give the number of visitors to each area, but it is unclear whether any visitor badges are included in the reported badge readings for a given month and year. In addition, these summary-style records sometimes also include categories representing dosimeters used for construction workers.

NIOSH does have an electronic database containing external dosimetry records as well. However, the records are given as an annual summary per person, and have no information regarding the area(s) where the worker may have received the dose. From an initial glance at the external dosimetry database, it does appear that the data span from the 1950s through at least the 1990s, and contain hundreds of thousands of rows of data, yet the absence of area information limits the possible uses of these data.

Recommendations for Further Investigations:

1. Figures 2 and 3 identify a rather complete external dosimetry set of records for TAN, with only a limited number of gaps. Additional SRDB searches may help fill these gaps.
2. The external dosimetry records often do not provide information regarding the sub-areas where a given worker experienced exposure during a given change-out. In light of the highly varied activities that took place at the different sub-areas, the complete TAN dataset cannot be used to build a coworker model for unmonitored workers at a given facility within TAN. It is therefore recommended that the data available for each sub-facility be compiled and plotted for use in building a coworker model for each facility, if it is determined that coworker models are needed due to the incompleteness of the external dosimetry database.

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SC&A 2015. *SC&A Evaluation of Idaho Chemical Processing Plant Class Definition Requiring Evidence of External Dosimetry (1963–1974)*, SC&A, Inc., Vienna, Virginia, and Saliant, Inc., Jefferson, Maryland. June 29, 2015.

SRDB Ref. ID: #119002 – This file contains 674 pages, most or all of which appear to be area exposure reports.

SRDB Ref. ID: #125523 – Survey of NRTS Roads, Idaho Nuclear Corporation, December 30, 1969.

SRDB Ref. ID: #125917 – This file contains 414 pages of Dosimetry Branch Activities 1952–1959 (Sheets marked "Personnel Meters Behavior")

SRDB Ref. ID: #138704 – CFA Monthly Reports, 1963 (contamination surveys, burial grounds, HP).

SRDB Ref. ID: #138957 – Possible Internal Intake Report, August 23, 2006.

SRDB Ref. ID: #139224. CFA Contamination Survey Results 1953–1956.

SRDB Ref. ID: #139258. CFA Health Physics Monthly Reports for January 1959–December 1961.

SRDB Ref. ID: #141608 – CFA HP – Selected Safe Work Permits for 1965.

SRDB Ref. IDs: #141859 and #141861 – Internal memorandum from G. Wehmann (Chief, HP Section) to W.P. Gammill (Chief, Site Survey Branch), Subject: Health Physics Practices at the NRTS Burial Ground. May 16, 1962.

SRDB Ref. ID: #141876 – Letter from AEC to Aerojet Nuclear Company, Subject: Personnel Exposure at Burial Ground. September 20, 1973.

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