# INL Test Reactor Area Nuclear Modeling

Stephen L. Ostrow, PhD

S. Cohen and Associates

**Contractor to:**

**Advisory Board on Radiation and Worker Health/ABRWH**

**Centers For Disease Control and Prevention**

**November 10, 2015**

# Presentation Content

- **Background of Investigation:** Does the methodology of ORAUT-OTIB-0054 (*Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analysis*) adequately model the reactor characteristics and operations of the Test Reactor Area (TRA)?
  - Are the reactors' and operating scenarios adequately enveloped by the OTIB cases so that the isotopic ratios are valid?
  - Have all off-normal operating scenarios been identified and are they adequately enveloped by the OTIB methodology?
- **Approach**:
  - Describe the OTIB
  - Describe the TRA reactors
  - Assess whether the OTIB models the reactors

(Presentation material taken from SC&A report, *NIOSH SEC-00219 Test Reactor Area Modeling*, SCA-SEC-2015-0074-C, Rev. 0, 9/28/15)

# Background of Investigation

- NIOSH SEC Petition Evaluation Reports
  - March 12, 2015: Rev. 00
  - July 21, 2015: Rev. 01
- Dose Reconstruction
  - Inherent in the SEC framework is the assumption that doses can be reconstructed with sufficient accuracy for site areas and time periods that lie outside the SEC class definition
  - This investigation examines one aspect of the dose reconstructability assumption for one of the several major site areas, the Test Reactor Area (TRA)
  - Applicability of ORAUT-OTIB-0054 to the more "exotic" reactors at Test Area North (TAN) is addressed in a separate report

# ORAUT-OTIB-0054 Scope

- *This guidance applies to a broad scope of reactor operations including*:
  - *plutonium production reactors (low enrichment, low burnup, Zircaloy or aluminum cladding)*
  - *research reactors [modest enrichment, modest burnup, stainless-steel (SS) or Zircaloy cladding; e.g., Training, Research, Isotope General Atomics (TRIGA) reactors]*
  - *high-enrichment, high-burnup reactors [e.g., Idaho National Laboratory's Advanced Test Reactor (ATR), fuel from naval reactors], and fast breeder reactors [e.g., the Hanford Site's Fast Flux Test Reactor (FFTF), Argonne National Laboratory–West's experimental breeder reactors].*

(ORAUT-OTIB-0054)

# ORAUT-OTIB-0054 Scope

- *It does not apply to:*
  - *operations involving decay times shorter than 10 days (e.g., radioactive lanthanum (RaLa) processing)*
  - *determination of intakes if radionuclides have been purposely extracted and concentrated, as for heat generation sources, medical uses, or waste handling operations that caused significant alteration to the source term to which workers were exposed*

(ORAUT-OTIB-0054)

# Prior Evaluation of ORAUT-OTIB-0054

- SC&A evaluated the general validity and applicability of the OTIB as part of the review process of the Subcommittee on Procedures Review, and all findings were closed by the Advisory Board
- The details of the evaluation and resolution process will not be recapitulated here

# ORAUT-OTIB-0054 Development

- Frequently, air-sampling or urinalysis data on worker exposure to mixed fission and activation products associated with nuclear reactors or fuel are available only in the form of gross beta or gross gamma activity unattributed to specific radionuclides.
- The OTIB provides guidance on assigning radionuclide-specific intakes using ratios to Cs-137 or Sr-90 indicator radionuclides.
- The goal of the OTIB is to reduce a large amount of reactor fuel isotopic data into a simplified, representative set that could be used by dose reconstructors looking at actual claimant cases.

# ORAUT-OTIB-0054 Development

- The starting point is the radionuclide mix in spent fuel for a number of different reactor types and fuel designs operated under a variety of different conditions (e.g., specific power, irradiation time, and burnup) calculated at several different decay times following removal from the reactor
- It is assumed that an actual claimant case condition will fit somewhere within the parameter space defined by the selected representative reactor types

# Initial Representative Reactors

The OTIB starts with 7 representative reactors

| Reactor | Category |
|---|---|
| **Hanford N-Reactor**<br>**Hanford Single-Pass Reactors** | Plutonium Production Reactors |
| **Fast Flux Test Facility (FFTF)** | Sodium-Cooled Fast Reactors |
| **Advanced Test Reactor (ATR)** | High-flux Reactors |
| **TRIGA® Reactor (Al-clad fuel)**<br>**TRIGA® Reactor (SS-clad fuel)** | Research Reactors |
| **Pressurized Water Reactor (PWR)** | Generic Reactor |

# Reduction of Representative Reactors

- The ORIGEN2 isotope generation and depletion code calculated isotopic inventories for the 7 reactors in 11 runs, decayed for 7 decay times, which produced activity data for 879 fission product nuclides and 688 activation product nuclides
- The activities in each dataset were normalized to that of Cs-137 at 10 days of decay. Four cases were then selected as representative for a wide variety of reactors, fuel types, and operational parameters:

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

# Reduction of Representative Reactors

- ORIGEN-S (SCALE system) runs then produced a total of 9 representative cases for the 4 reactors.
- NIOSH customarily considers all 9 reactor cases when doing a dose reconstruction and, in the absence of individual worker information, might apply data for 4 decay times (10 days, 40 days, 180 days, and 1 year) as well, if required by a specific dose reconstruction case.

# ORAUT-OTIB-0054 Representative Cases

| 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<br>Burnup = 314,684 MWd/MTU | Maximum burnup at nominal power. |
| ATR 2 | Specific power = 8,651.2 MW/MTU Irradiation time = 36.4 days<br>Burnup =314,904 MWd/MTU | Maximum burnup at maximum assembly power. |
| ATR 3 | Specific power = 2,379.1 MW/MTU Irradiation time = 56 days<br>Burnup = 133,230 MWd/MTU | Nominal burnup at nominal power. |
| FFTF 1 | Specific power = 163.8 MW/MTHM Irradiation time = 929.4 days<br>Burnup = 152,230 MWd/MTHM | Maximum burnup at nominal power. |
| FFTF 2 | Specific power = 163.8 MW/MTHM Irradiation time = 488.3 days<br>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<br>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. |

# Advanced Test Reactor

- Surrogate for high-flux reactors, as might be encountered in materials testing or experimental reactors
- The ATR, which had a maximum power level of 250 MW, is the latest and largest of three materials testing reactors at INL (the other two are the MTR and the ETR, both of which no longer operate), and has operated for over 40 years to the present
- The ATR is a pressurized, light-water moderated, beryllium-reflected reactor, using highly enriched uranium fuel (93.15% nominal enrichment) arranged in an unusual curved plate configuration
- The primary reactivity control mechanism consists of a unique design of rotating beryllium cylinders with hafnium thermal neutron absorber shells
- The OTIB models three cases with different specific powers, irradiation times, and burnups

# Fast Flux Test Facility

- Represents sodium-cooled fast reactors, which use fast neutron fission to burn both depleted uranium and plutonium. The concept can also breed plutonium fuel through neutron capture in U-238.
- The FFTF was a 400 MW, liquid sodium-cooled reactor located at the DOE Hanford site that operated from April 1982 to April 1992, primarily to explore reactor designs and operations that might be applicable to a commercial fast breeder reactor.
- The fuel was mixed oxide with plutonium (86 wt% Pu-239) amounting to 29 wt% of the heavy metal (uranium plus plutonium); the driver assemblies were clad with 316 SS (stainless steel); and the assemblies were subjected to high burnups.
- The OTIB models two cases with different specific powers, irradiation times, and burnups

# Hanford N-Reactor

- Represents Pu production reactors, where it is desired to have low enrichment fuel (maximizing U-238, which absorbs a neutron and produces Pu-239 following two beta decays) and short irradiation time (to minimize the undesirable "contaminant" Pu-240).
- The N-Reactor, which operated from 1963 to 1987, was a low-enrichment, graphite-moderated, pressurized water-cooled reactor that initially produced Pu for weapons and was later modified to produce steam to generate electricity
- The core of the N-Reactor was a 1800-ton graphite block, 33 feet high by 33 feet wide by 39 feet (12 meters) long
- The OTIB models two cases with different specific powers, irradiation times, and burnups

# TRIGA Reactor

- The TRIGA reactor with stainless steel clad fuel represents research and radiographic reactors.
- The low-power, water-cooled, pool-type, uranium zirconium hydride-fueled reactors are manufactured in a variety of versions by General Atomics (GA) primarily for research and educational purposes, and are considered inherently safe due to their very large prompt negative temperature coefficient of reactivity that quickly drops the reactor subcritical in response to any power excursions. This feature also allows the TRIGA to be pulsed to a high power level, followed by a quick automatic self-shutdown.
- GA has supplied over 65 reactors throughout the world over the past 45 years, ranging in power from 20 kW to 16 MW and containing 60–100 fuel elements. The models have progressed from Mark I through Mark III, with a number of specially modified units also produced. The early models used highly enriched uranium (70%), but the later models reduced the enrichment to 20% and the earlier models were modified to accept lower enrichment fuel.
- The OTIB models two cases with different specific powers, irradiation times, and burnups

# INL Test Reactor Area

Table 5-1 of the Evaluation Report lists the eight TRA reactors, only the first three of which were (or are, in the case of the ATR) high-power, high-flux reactors used for material testing. The remaining critical facilities were low-power reactors used for reactivity measurements and are not considered in this analysis.

SC&A examined whether ORAUT-OTIB-0054 adequately envelopes the parameters of the three materials testing reactors, including the fuel and cladding composition, power level profile, burnup, and decay time following removal from the core.

# INL Test Reactor Area

| 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 (ATRC) | 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 |

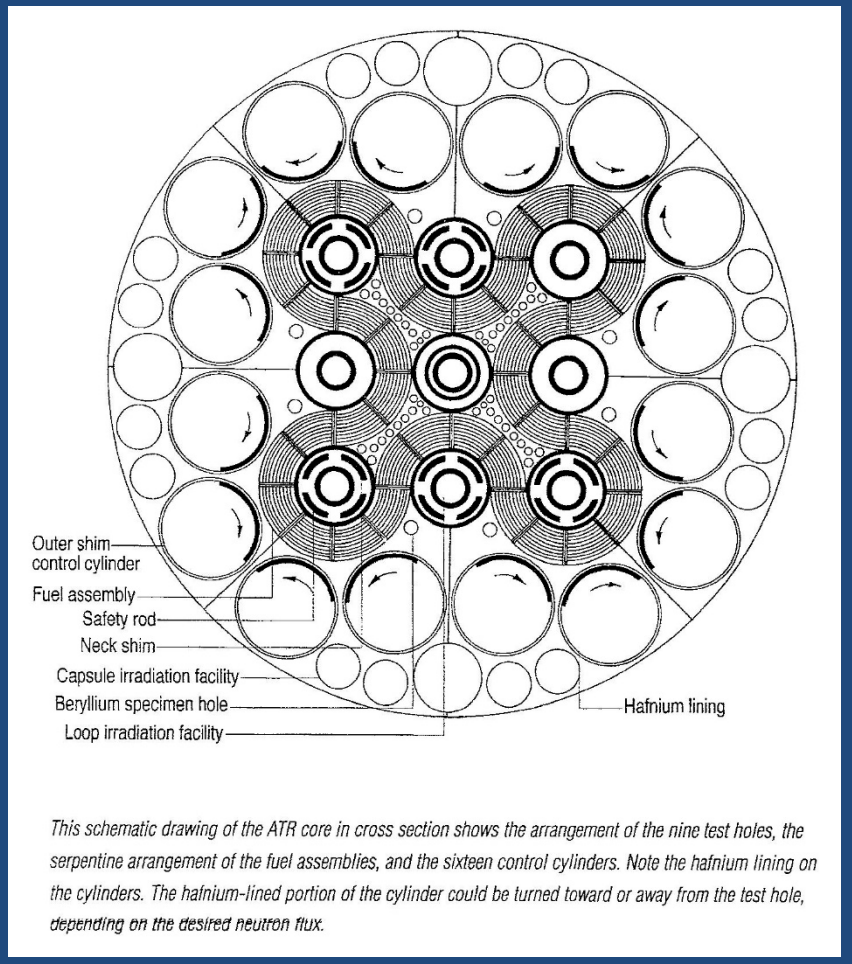
# SC&A Evaluation – ATR

## Advanced Test Reactor

- OTIB-0054 explicitly models the fuel characteristics and operations of the ATR, so it is expected that any workers exposed to ATR fuel during reactor operations or out of the reactor, but before any isotopic separation, would be adequately treated by the methodology in the OTIB in order to determine internal exposures.
- SC&A did not find material instances of the ATR operating outside of its design envelope.

# ATR Illustrations

(Wikipedia)




*This schematic drawing of the ATR core in cross section shows the arrangement of the nine test holes, the serpentine arrangement of the fuel assemblies, and the sixteen control cylinders. Note the hafnium lining on the cylinders. The hafnium-lined portion of the cylinder could be turned toward or away from the test hole, depending on the desired neutron flux.*

(Stacy, S., *Proving the Principle: A History of the Idaho National Engineering and Environmental Laboratory 1949-1999*, DOE/ID-10799, 2000)

# SC&A Evaluation – MTR

**Materials Test Reactor**

- The MTR, designed by Argonne National Laboratory and Oak Ridge National Laboratory and sited at INL, was the second reactor built at that location.
- It achieved first criticality on March 31, 1952, and attained full power of 30 MW on May 22, 1952. The power level was later increased to 40 MW in September 1955.
- It was last used in 1970 with a plutonium core, and was succeeded by the larger and higher-flux ETR and ATR in the same TRA facility complex.

# SC&A Evaluation – MTR

- The MTR was cooled and moderated with light water, and used Al-clad, curved plate, enriched uranium fuel the majority of the time. Beryllium, graphite, and light-water neutron reflectors surrounded the core and control rods entered from above.
- The core was quite small by modern standards, measuring 9 in × 28 in in cross section × 24 in high and containing only a maximum total of 4.9 kg of U-235.
- The MTR provided a high neutron and gamma flux for the accelerated testing of materials for reactor development, and had about 100 beam holes that penetrated into the core and its surroundings from several different directions. Samples for irradiation and experiments were provided by several different laboratories.

# SC&A Evaluation – MTR

- The MTR operated for > 125,000 hrs and performed > 19,000 irradiations.
- The MTR fuel enrichment, cladding, and plate design were similar in important nuclear aspects to those of the ATR, and both reactors were light-water cooled and moderated, with beryllium and aluminum reflectors. Its power level and burnup, however, were considerably less than those of the ATR. Hence, the OTIB-0054 methodology, which explicitly models the ATR, should also adequately envelope the MTR in considering internal exposures when the later reactor operated with uranium fuel.

# SC&A Evaluation – MTR

**Radioactive Lanthanum (RaLa)**

- The RaLa extraction campaign from 1956 through 1963 was an "unusual" operating scenario. 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 since, 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 Evaluation – MTR

**Plutonium Core**

- Although the MTR initially used uranium fuel, in 1958, it became the first reactor run with a 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 wt% Pu and 79 wt% Al.
- The reactor reached initial power on January 28, 1970, and operated at various power levels up to 24 MW until April 23, 1970, at which time it had accumulated a burnup of 923 MWd. This compared to the standard U-235 core, which operated at 40 MW.

# SC&A Evaluation – MTR

**Plutonium Core**

- Although the MTR plutonium fuel operations used fuel plates similar to those used with 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 this report.

# SC&A Evaluation – MTR

**Plutonium Core**

- The Hanford N Reactor cases, although containing plutonium, are likely not applicable for several reasons. One obvious difference is that the MTR was water-moderated, while the Hanford N Reactor was graphite-moderated. In addition, the N-Reactor 1 case (production of weapons-grade Pu with nominal 6% Pu-240) and the N-Reactor 2 case (production of fuel-grade Pu with nominal 12% Pu-240) started out with a low-enrichment uranium core and sought to breed plutonium as the end product, while the MTR started out with a plutonium core and used fertile Pu-240 (23%) to breed fissile Pu-241.

- It is not clear which, if any, of the nine OTIB-0054 cases would adequately envelope the case of the MTR with a plutonium core. Hence, whether the MTR with a plutonium core can be adequately modeled with ORAUT-OTIB-0054 has not been determined at this time.

# Engineering Test Reactor – ETR

- The ETR, located south of, and adjacent to, the MTR, was designed taking advantage of information gathered and lessons learned from the operations and limitations of the MTR.
- The ETR was larger than the MTR, with four times the neutron flux (both thermal and fast), with its main control rods entering from the bottom (decluttering the operating floor), without troublesome beam holes (thereby lessening the potential for neutron streaming out of the reactor), and with placement of test materials and experiments directly in the core.
- The experimental spaces were sufficiently large to accommodate experiments with their own closed coolant loops.

# Engineering Test Reactor – ETR

- The ETR achieved first criticality on September 19, 1957, and operated at a maximum power level of 175 MW until December 1981.
- The reactor was light-water cooled and moderated and uranium fueled. The reactor core was in a square configuration with an average diameter of 32 inches and an active height of 36 inches. Beryllium and aluminum neutron reflectors were placed outside the core. Initially, a new core contained about 14 kg of U-235, which was increased to 23.7 kg in January 1963.
- The average thermal flux in the core exceeded $10^{14}$n/(cm$^2$-s). The 49 fuel assemblies (increased to 52 in January 1963) contained 19 Al-clad fuel plates per assembly. The core life before refueling was initially 3,500 MWd, which increased to 4,500 MWd in January 1963.

# Engineering Test Reactor – ETR

- A notable extension of the reactor occurred in 1972, when a Sodium Loop Safety Facility with accompanying cooling and other systems was added. This was in support of the DOE's breeder reactor program.

- As with the MTR operating with uranium fuel, the OTIB-0054 methodology, which explicitly models the ATR, should also adequately envelope the ETR in considering internal exposures.

# Summary and Path Forward

- Resolve potential issues of applicability of ORAUT-OTIB-0054 to MTR operating with plutonium fuel.

- The analysis of applicability of ORAUT-OTIB-0054 to more “exotic” reactors at Test Area North (TAN), with different fuel compositions and arrangements and operations than the OTIB representative reactors, under normal and off-normal conditions, is addressed in the SC&A report on TAN. Other experimental reactors located in several different areas on the INL site are yet to be addressed.

# Comments and Questions?