ATTACHMENT (1)

APPLICATION FOR CONSENT TO INDIRECT TRANSFER OF CONTROL OF LICENSE

APPLICATION FOR CONSENT TO INDIRECT TRANSFER OF CONTROL OF LICENSES

May 30, 2012

Submitted by

Aerotest Operations, Inc.

Aerotest Radiography and Research Reactor
Facility Operating License No. R-98
Docket No. 50-228

APPLICATION FOR CONSENT TO
INDIRECT TRANSFER OF CONTROL OF LICENSE

TABLE OF CONTENTS

I. INTRODUCTION

Aerotest Operations, Inc. (Aerotest) hereby submits an application seeking U.S. Nuclear Regulatory Commission (NRC) consent to the indirect transfer of control of the Facility Operating License No. R-98 (the License) for the Aerotest Radiography and Research Reactor (ARRR). (The ARRR is sometimes referred to herein as the licensed facility.) The indirect transfer of control will result from acquisition of Aerotest Operations, Inc. by Nuclear Labyrinth LLC (Nuclear Labyrinth).

As set forth in Nuclear Labyrinth's April 4, 2012 commitment letter, Nuclear Labyrinth will acquire ownership of the stock and business of Aerotest from Autoliv ASP, Inc. (Autoliv) (the ultimate parent corporation of Aerotest). It is expected that the transaction will close on or before September 30, 2012. Aerotest Operations, Inc., the owner of the ARRR, is a wholly owned subsidiary of OEA Aerospace, Inc., a wholly owned subsidiary of OEA, Inc. which is a wholly-owned subsidiary of Autoliv ASP, Inc (collectively, "seller"). The parties intend that all of the shares of the stock of Aerotest would be transferred to Nuclear Labyrinth upon the completion of commercial negotiations (and upon receiving the NRC's consent to transfer the facility operating license). All liabilities regarding the management, storage, removal and disposal of the nuclear fuel located at the facility, as well as liabilities associated with the decommissioning of the site, would be transferred to the Nuclear Labyrinth. Seller will contribute funds for of a decommissioning trust. Additionally, at closing of the transaction, seller will transfer funds sufficient to fund approximately twelve months of operating expenses during the restart of the facility and its resumption of commercial operations. It is the intention of both buyer and seller that management and technical personnel at the ARRR, including those responsible for licensed activities including reactor safety, would be offered employment on terms and conditions consistent with current norms for the type and size of business and employment market.

Nuclear Labyrinth is incorporated in the State of Utah and is managed by its Chief Executive Officer (CEO) and sole owner, Dr. David M. Slaughter, PhD. Dr. Slaughter, has provided education, training, and research in nuclear science and engineering since 1990. Upon closing of the transaction and approval of the license transfer, Aerotest will continue to provide neutron radiography services and will enhance its principle mission to include education, training and research.

Upon closing of the transaction, Dr. Slaughter will replace Dario Brisighella as President of Aerotest. With the exception of this change, the current management and staff at the facility will remain in place. Given Dr. Slaughter's extensive experience in research reactor operation and administration, Aerotest will maintain its technical qualifications.

Aerotest has a long and substantial history as a provider of neutron radiography services. Aerotest plans to restart the reactor and continue with these services as well as add education, training, and research services. Along with the aforementioned funds transferred at closing, fees from these services will provide the source of funds to cover operating and maintenance expenses of the facility. In addition, at closing Autoliv will transfer funds for decommissioning the facility to a trust set up in accordance with regulatory requirements. Thus, Aerotest will remain a financially qualified licensee.

The ARRR is a 250 Kilowatt thermal (kw(t)) research reactor located in San Ramon, California, and owned and operated by Aerotest. Aerotest is the holder of the license, which was issued under the provisions of 10 C.F.R. § 50.21(c), and authorizes Aerotest to possess, use, and operate the ARRR. Aerotest is currently wholly owned by OEA Aerospace, Inc., which is in turn wholly owned by OEA, Inc. OEA, Inc. was purchased by Autoliv ASP, Inc. on May 9, 2000. Autoliv ASP, Inc. is wholly owned by Autoliv, Inc., which is incorporated in Delaware with its headquarters in Stockholm, Sweden.

The information contained in this Application demonstrates that Nuclear Labyrinth will possess the requisite qualifications to own Aerotest and therefore to acquire an indirect interest in the License. The proposed indirect transfer of control of the License will not result in any change in the role of the licensed operators of the facilities and will not result in any changes to their technical qualifications. Following its acquisition by Nuclear Labyrinth, Aerotest will remain financially qualified to conduct its activities under the License. Financial assurance for decommissioning will be assured as described below. Finally, this request for consent to the indirect transfer of control of the License will not result in the licensed facility becoming owned, controlled, or dominated by a foreign entity.

II. STATEMENT OF PURPOSE OF THE TRANSFER AND NATURE OF THE TRANSACTION MAKING THE TRANSFER NECESSARY OR DESIRABLE

Because of Autoliv,Inc.'s foreign status, NRC has requested that Autoliv divest its interest in the ARRR. Pursuant to that request, Autoliv,Inc. has worked to identify a suitable buyer for the ARRR facility. Nuclear Labyrinth and Aerotest have agreed that Nuclear Labyrinth will acquire the ownership of the stock and business of Aerotest. The parties intend that all of the shares of the stock of Aerotest would be transferred to Nuclear Labyrinth upon the completion of commercial negotiations (and upon receiving the NRC's consent to the indirect transfer of the License). All liabilities regarding the management, storage, removal and disposal of the nuclear fuel located at the facility, as well as liabilities associated with the decommissioning of the site, would be transferred to the Nuclear Labyrinth. Seller will contribute funds for of a decommissioning trust. Additionally, at closing of the transaction, seller will transfer funds sufficient to fund approximately twelve months of operating expenses during the restart of the facility and its resumption of commercial operations.

III. GENERAL CORPORATE INFORMATION REGARDING AEROTEST OPERATIONS, INC. AND NUCLEAR LABYRINTH LLC

A. General Information Regarding Aerotest

1. Name and Address

Aerotest Operations, Inc.
3350 Airport Road
Ogden, UT 84405

2. Description of Business

Aerotest is, and after the transaction described herein will continue to be, the owner and operator of the ARRR. This facility has been primarily used as a material interrogation tool using neutron radiography for research and development of personnel safety systems such as air bags, pilot escape systems, turbine blades for military and commercial jet engines, and pyrotechnic hardware for the space shuttle. It has also been used to inspect medical equipment, e.g., cardiac pacemaker batteries, components of devices for the timed release of drugs within the body, and phenolic hips. The ARRR has been used for training in nondestructive examination techniques using neutron radiography, research and development of a patented boron nitride conversion screen and the development and evaluation for new file and no-film systems in the field of neutron radiography. If the transaction described herein is consummated and the indirect transfer is approved, Nuclear Labyrinth intends to enhance the ARRR's mission to also include education, training and research in the area of neutron science and engineering. This "education through research" endeavor not only will seek to improve neutron technologies through better understanding of the basic science and engineering as other reactor programs at educational institutions but will focus on the effective practical laboratory training of graduate students involved with academic research groups and students enrolled in trade programs. The inclusion of these students will provide intensive hands-on experiences that will reinforce the academic principles taught in the classroom.

To that end, Aerotest anticipates faculty and students of a number of West Coast and Intermountain-west universities, colleges, and local trade schools to be involved. Formal collaborations with these institutions will allow educational and research resources to be available in the completion of students' academic pursuits. Multi-institutional involvement allow for a diversity of ideas to be considered and the best of those to flourish.

The proposed expanded mission of the ARRR's reactor will support outstanding, cutting-edge and innovative research and educational methods. The proposed education and research activities include but are not necessarily limited to:

1. perform material diagnostics - neutron radiography and tomography, neutron activation analysis, and ultra sensitive fission track analysis

2. identify and quantify explosives and other contraband by neutron interrogation techniques

3. advance neutron detection, spectroscopy, imaging, and dosimetry

4. model neutron phenomena and diagnostic technologies, radiation exposure, and dose

3. Organization and Management

After the completion of the transaction described herein, Aerotest will not be owned, controlled or dominated by an alien, foreign corporation or foreign government. The names and addresses of the members of Aerotest's board of directors and its principal officer as of the effective date of the transaction are listed in Attachment (4). It is anticipated that additional corporate officers will be identified and approved by the Board of Directors following the effective date of the transaction. All directors and the officer are citizens of the United States.

B. General Information on Nuclear Labyrinth LLC

1. Name and Address:

Nuclear Labyrinth LLC
10874 South Bay Meadow Circle,
Sandy, Utah 84092

2. Description of Business:

Nuclear Labyrinth, through Dr. Slaughter, provides education, training, and research in Nuclear Science and Engineering. Nuclear Labyrinth is a business that evolved from E-Cubed Inc. (E^3) established in 1989 in the State of Utah. While the structure of the business has changed, its purpose of education, research, and technology development has not wavered. E^3 developed and improved nuclear technologies to ensure their function to be efficient in operation, economic in cost and environmentally compatible. E^3 collaborated with educational institutions, e.g., University of Utah, to pursue research funding that is not normally available to a single institution. "A better future through education and research" was the E^3 – University of Utah partnership motto. In approximately 2002, the State of Utah awarded Dr. Slaughter funds for a Center of Excellence, which, among other things, was to identify nuclear based spinoff technologies for commercialization. This effort was the impetus for changing the company name to Nuclear Labyrinth.

3. Organization and Management:

Nuclear Labyrinth is incorporated in the State of Utah where a Board of Directors is not required. It is owned and managed by its CEO, Dr. Slaughter, a US citizen. Dr. Slaughter is the sole owner and investor of Nuclear Labyrinth; therefore, it is not controlled or dominated by an alien, foreign corporation or foreign government. His address is provided in Attachment (4).

IV. TECHNICAL QUALIFICATIONS

The technical qualifications of Aerotest will be enhanced by the proposed indirect transfer of control of the License. There will be no physical changes to ARRR and no changes in its day-to-day operations in connection with the indirect transfer of control. Proposed conforming amendments to the Technical Specifications and License are included at Attachment (10). Attachment 10 provides proposed markups of the draft license previously prepared by the NRC for the X-Ray Industries, Inc. license transfer.

With the exception of Aerotest's new president, Dr. David M. Slaughter, the operating organization for the licensed facility is expected to remain essentially unchanged as a result of the transaction. Nuclear Labyrinth intends to retain the current management team at the facility. Thus, no significant changes in the management or organization are expected to be made as a result of the transaction. Of note, the ARRR has been shut down since October 15, 2010 and is maintaining the staff necessary to meet all licensing requirements. It is anticipated that additional staff will be hired as necessary to support operations. Dr. Slaughter plans to manage the ARRR on-site. Eventually he will undergo operations and administrative training to participate as the Reactor Administrator, a role he has successfully fulfilled in previous endeavors.

Dr. Slaughter's experience includes working in academic settings (Washington State University, University of Utah, University of California ("UC")-Davis) as well as industrial settings (photogenics, general electric, energy and environmental Research) in performing cost effective research and development activities. Under both academic and industrial venues, his responsibilities included, among other things, estimating, securing and managing federal and industrial contracts and agreements while ensuring the work was successfully completed and reports submitted on time. This included understanding and approving payroll and benefits that were associated with the employees under his direction.

Dr. Slaughter has experience in nuclear engineering, radioenvironmental sciences, radioassays, radiotoxicology, chemical engineering, neutron radiography/tomography, and radiation and materials interactions including experience in regulatory matters with NRC, EPA, and OSHA. He is also familiar with environmental monitoring, dose reconstruction, and nuclear forensics techniques and related analyses.

Dr. Slaughter has extensive academic/training, research, and operational/administrative experience with TRIGA Reactors. This broad and diverse background supports his understanding of TRIGA design, operational characteristics, and the NRC regulatory framework.. His practical experience includes holding positions of Reactor Administrator (11 years), Reactor Supervisor (7 years), and Senior Reactor Operator (11years) at NRC regulated TRIGA nuclear reactor facilities (University of Utah and UC –Davis). In addition to academic and research activities conducted at Washington State University (1 MW TRIGA), University of Utah (100 KW TRIGA) , and UC-Davis (2MW TRIGA), Dr Slaughter has conducted research activities using other TRIGA and plate-type research nuclear reactors: FRM II –Antares, Technical University Munich, neutron/x-ray radiography and tomography; 5 MW Reactor, MIT, Fission Track Analysis (FTA); 250 KW TRIGA, Aerotest San Ramon Ca, neutron radiography; 1 MW TRIGA Reactor, U-Mass Lowell, neutron detection and spectroscopy; 20 MW NIST Reactor, neutron detection, spectroscopy and imaging. A statement of Dr. Slaughter's professional Vita is attached as Attachment (5).

V. FINANCIAL QUALIFICATIONS

An initial year projected income estimate after the completion of the transaction is provided in Attachment (6). It includes the participation in four revenue generating areas: (1) Neutron Interrogation by Radiography; (2) Nuclear Science and Engineering Research; (3)

Training; and (4) Neutron Science/Engineering Research (without the use of ARRR). These sources of funds will compensate for reactor operating, training, research, and service activities that will be rendered at the facility. Additionally, at closing of the transaction, seller will transfer funds sufficient to fund approximately twelve months of operating expenses during the restart of the facility and its resumption of commercial operations. Aerotest has a long and substantial history as the single largest source for interrogation and inspection using neutron radiography of safety critical flight hardware for both commercial, military, and government clients. This includes educating and training neutron radiography practitioners to effectively diagnose the quality of manufacturing process and advancing the neutron detection and imaging techniques.

The initial or first year estimate after resuming full commercial operation for Aerotest Operations is based on detailed accounting of seven previous years of operation (2003-2009). As shown in Attachment (6), expenditures are organized into Unadjusted Cost, Fuel Credit (accounting for fuel burn-up and replacement), charges mandated by NRC, and Decommissioning Contributions. Historical data shows that the contributions for fuel burn up and replacement has been approximately five percent of the unadjusted cost of tasks that use the ARRR. Thus, the projections calculate fuel burn up and replacement as five percent of the costs associated with these tasks. Similarly, the NRC fees have historically been approximately 10% of that same unadjusted cost. (This does not include NRC fees associated with relicensing or foreign divesture activities). NRC fees and fuel are two "variable" cost components that are tracked separately so their effects on our ability to conduct research can be directly understood and conveyed to clients, funding agencies and governmental identities. This initial year estimate is included in the projected income statement for Aerotest for the first five years of operation after the transaction.

The projected income statement for Aerotest for the first five years after the transaction is included in Attachment (7). As described above, the first year is estimated based on historical financial accounting as provided in Attachment (6). For the subsequent four years, the projected income statement assumes a three percent cost escalation per year for Tasks 1 and 3. The plan also assumes a 25 percent growth in revenue in the area of research, Tasks 2 and 4, until the completion of the 5th-year where the rate will be lowered to three percent. This allows the research activities to grow at a controlled and reasonable pace consistent with research acquisition processes and size of the organization. The plan is conservative on the stated amounts of sustainable research to ensure funding targets and stated financial obligations can be reasonably met.

VI. DECOMMISSIONING FUNDING

A decommissioning cost estimate was prepared by Energy Solutions (www.energysolutions.com) as set forth in a report included at Attachment (8). The decommissioning method used in this analysis is DECON.

When decommissioning is commenced, the facility may need to be in SAFSTOR until the removal of fuel can be effected under the terms of a contract between Aerotest and the US Department of Energy (DOE). Under the contract, DOE is obligated to take the ARRR fuel, "as expeditiously as practicable." The Energy Solutions report provides a detailed description of the basis for the decommissioning cost estimate of $3,285,800 including labor, energy and waste cost estimates.

A projection for decommissioning financial assurance is provided in Attachment (9). This projection uses a real rate of return of 1.95% based on an imputed tax affected rate of return on 30 year t-bills on average since 2001. The initial funding will be provided by Autoliv Inc. in the form of cash. These funds will be immediately contributed to and held in a decommissioning trust in conformance with the examples provided in Appendix B of Regulatory Guide 1.159, Assuring the Availability of Funds for Decommissioning Nuclear Reactors. UBS will act as custodian and trustee. A UBS wealth advisor signature team licensed in California and Utah will act as investment advisors for the trust. The trustee and advisors will act independently of Aerotest in conformance with the referenced guidance document. Future funding will be provided by Aerotest per the schedule included in Attachment (9). These funds will be from decommissioning fees charged to users of ARRR for the first five years of operation under the new ownership. This fee represents an actual expense in providing research and service and the client will be assessed 5 % of the "unadjusted" cost associated with Tasks 1 and 2. It is anticipated that the collected fees will be deposited quarterly. After five years, the required funding level will be adjusted on a bi-annual basis based on updated decommissioning cost estimates to be prepared by outside experts. For the purposes of projecting future costs, the schedule included in Attachment (9) uses a cost escalation factor of 3%.

VII. RESTRICTED DATA AND CLASSIFIED NATIONAL SECURITY INFORMATION

This application for indirect transfer of control does not contain any Restricted Data or other classified National Security Information, nor does the proposed transfer result in any change in access to any Restricted Data or classified National Security Information. Aerotest's existing restrictions on access to Restricted Data and classified National Security Information are unaffected by the proposed transfer and will remain in effect. In compliance with Section 145a of the Atomic Energy Act, and 10 C.F.R. § 50.37, Aerotest and Nuclear Labyrinth agree that Restricted Data or classified National Security Information will not be provided to any individual until the individual has been approved for access under the provisions of 10 CFR Parts 25 and/or 95.

VIII. ENVIRONMENTAL CONSIDERATIONS

The requested consent to the indirect transfer of the License is exempt from environmental review because it falls within the categorical exclusion contained in 10 CFR § 51.22(c)(21), for which neither an Environmental Assessment nor an Environmental Impact Statement is required. Moreover, the proposed transfer will not directly affect the actual operation of the ARRR in any substantive way. The proposed transfer does not involve an increase in the amounts, or a change in the types, of any radiological effluents that may be allowed to be released off-site, and it does not involve an increase in the amounts, or a change in the types, of non-radiological effluents that may be released off-site. Further, there is no increase in the individual or cumulative operational radiation exposure, and the proposed transfer has no environmental impact.

IX. PRICE-ANDERSON INDEMNITY AND NUCLEAR INSURANCE

Because Aerotest will remain the licensee, the transaction will not result in any change with respect to the Price-Anderson indemnity nor to nuclear insurance. The current indemnity agreement will remain unchanged and all insurance requirements will continue to apply.

X. EFFECTIVE DATE

The parties to the proposed transaction anticipate closing on or before September 30, 2012. Aerotest respectfully requests that the NRC review this Application on a schedule that will permit NRC to issue its consent to the indirect license transfer as promptly as possible. Aerotest and Nuclear Labyrinth are prepared to work closely with the NRC staff to help expedite the Application's review, but request approval in any event no later than September 30, 2012. Aerotest and Nuclear Labyrinth further request that the consent be immediately effective upon issuance and that it permit the transfer of control to be implemented at any time after the date of approval of this Application and closing of the transaction.

XI. CONCLUSION

Based upon the foregoing information, Aerotest requests that the NRC consent to the indirect transfer of control of the Facility Operating License No. R-98 for the Aerotest Radiography and Research Reactor.

ATTACHMENT (2)

COMMITMENT LETTER

Aerotest Operations, Inc.
May 30, 2012

UCLEAR
LABYRINTH

Document Control Desk
US Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, Maryland 20852

April 4,2012
Docket Number. 50-228

Via Express Delivery

RE: 10CFR 50.80 (a)

Buyer Commitment for Aerotest Radiography and Research Reactor
License No R-98

Nuclear Labyrinth LLC ("Buyer)" is considering acquiring the ownership of the stock and business of Aerotest operations, Inc ("Aerotest") from Autoliv ASP, Inc. (the ultimate parent corporation of Aerotest) ("Seller"). A date certain for this transaction has not been determined, although it is expected to occur on or before September 30, 2012. If the Buyer acquires Aerotest, Buyer intends to continue operation of the facility and enhance its principle mission to include education, training and research in the area of neutron science and engineering. It is understood that all operations are subject to Reactor License R-98 (Aerotest Radiography and Research Reactor ("ARRR") License).

Buyer understands that under the terms of the ARRR license, Seller is subject to a number of constraints, conditions, requirements and commitments. Prior to acquisition, Seller will remain responsible for the compliance with the ARRR license and any violations occur prior to the effective date of the acquisition. After the effect date of acquisition, Buyer agrees to abide by all of the constraints, conditions, requirements, and commitments of the ARRR license and any violations that occur after the effective date of acquisition.

Upon License transfer, buyer will establish a decommissioning trust fund consistent with 10 CFR 50.75 (e) (1) under terms acceptable to the NRC. A draft of the trust documents will be submitted to the NRC for review at a later date.

I appreciate your assistance in this matter. Should you have any questions, please feel free to contact me directly.

Sincerely,

David M. Slaughter, PhD,
CEO,
Nuclear Labyrinth LLC

A020
NRR

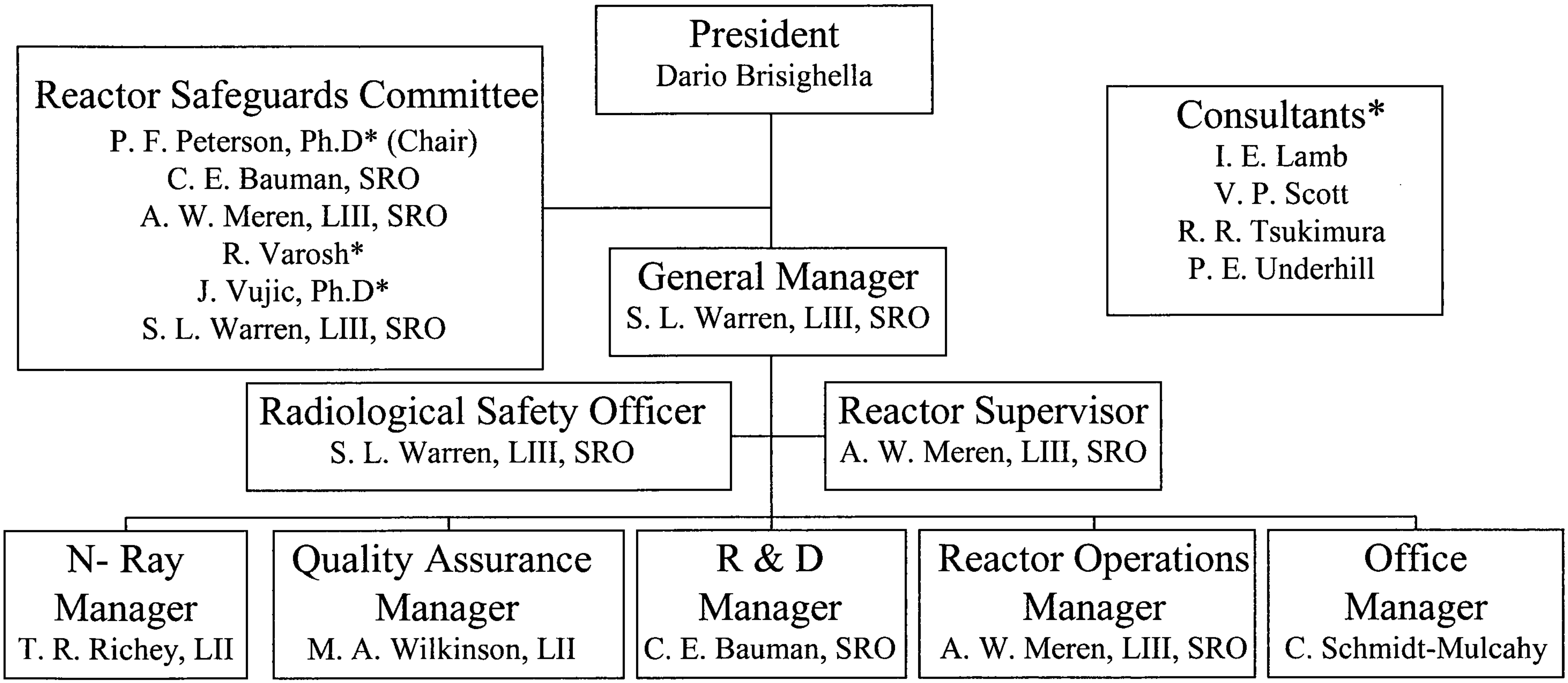
University Research Park ~ P.O. Box 58023 ~ Salt Lake City, UT 84158-0023

ATTACHMENT (3)

PRE- AND POST-SALE SIMPLIFIED ORGANIZATIONAL CHARTS

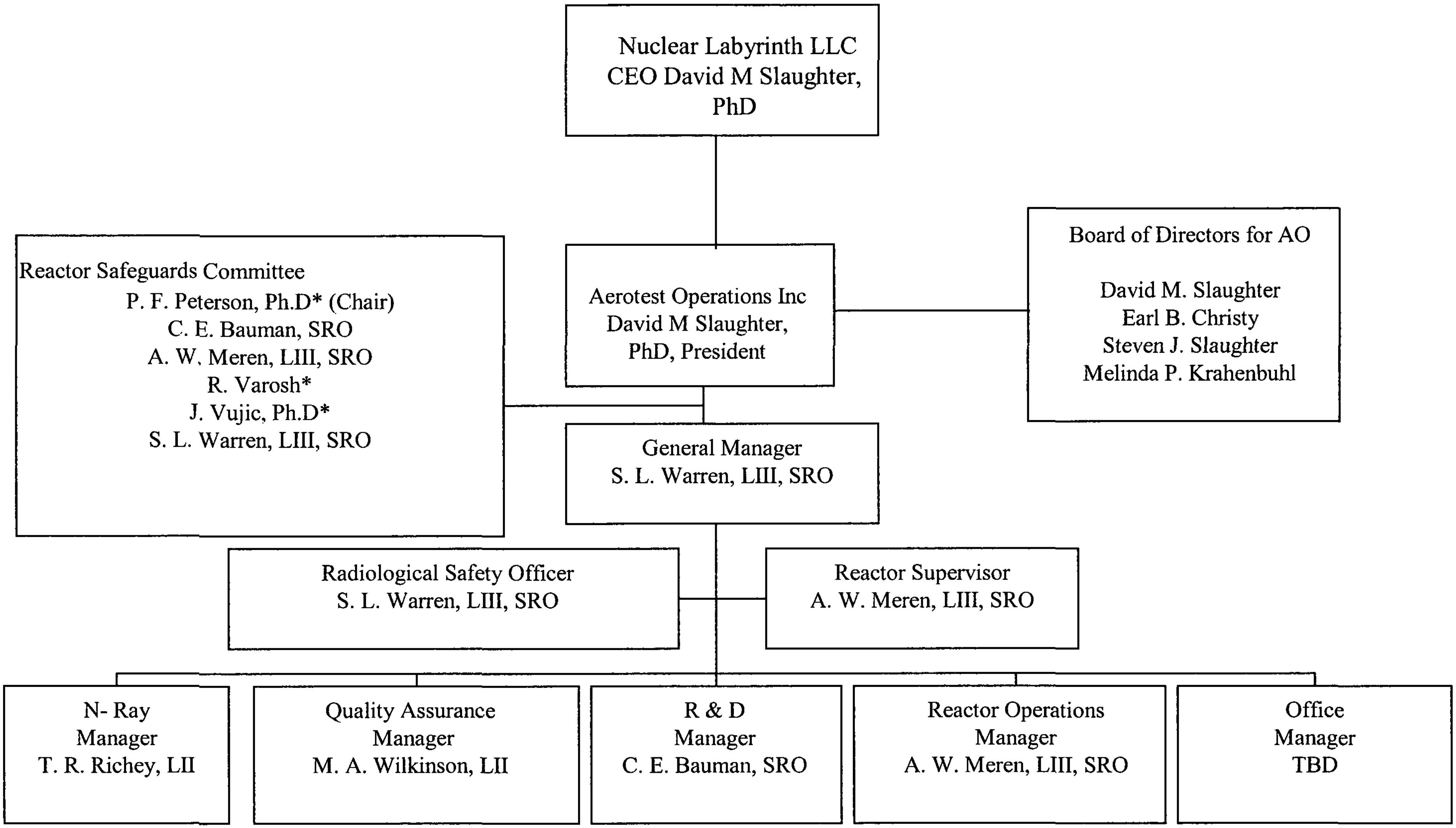
Simplified Organizational Chart of Pre-Sale Reactor Ownership:

Current Aerotest Operations



Simplified Organizational Chart of Post-Sale Reactor Ownership:

Post Transfer Aerotest Operations



ATTACHMENT (4)

DIRECTORS AND PRINCIPAL OFFICERS OF NUCLEAR LABYRINTH LLC AND AEROTEST OPERATIONS, INC

I. Nuclear Labyrinth, LLC

A. Nuclear Labyrinth Board of Directors

Nuclear Labyrinth is incorporated in the state of Utah, which does not require a Board of Directors.

B. Nuclear Labyrinth Principal Officer

Dr. David M. Slaughter, PhD is the Chief Executive Officer. His business address, is as follows:

10874 South Bay Meadow Circle
Sandy, Utah 84092

II. Aerotest Operations, Inc.

A. Aerotest Operations, Inc. Board of Directors

As of the effective date of the transaction, the business addresses and names of the Aerotest Operations, Inc. directors are as follows. All are U.S. citizens.

1. David M. Slaughter, PhD

   10874 South Bay Meadow Circle
   Sandy, Utah 84092

2. Earl Bradford Christy Jr., MBA, JD(Expertise Legal)

   3163 Shamrock Street East
   Tallahassee, FL 32309

3. Steven J. Slaughter (Expertise Financial)

   3025 East 142nd Drive
   Thornton CO 80602

4. Melinda P. Krahenbuhl, PhD (Expertise Science/Engineering)

   19278 Reddaway Avenue
   Oregon City, OR 97045

B. Aerotest Operations, Inc. Principal Officer

As of the effective date of the transaction, Dr. Slaughter will be the President and only principal officer of Aerotest. His business address is provided above.

ATTACHMENT (5)

PROFESSIONAL BACKGROUND OF DR. DAVID M. SLAUGHTER

ATTACHMENT (5)

PROFESSIONAL BACKGROUND OF DR DAVID M. SLAUGHTER

David M. Slaughter, PhD
Curriculum Vita

NAME David M. Slaughter
ADDRESS 10874 South Bay Meadow Circle, Sandy, Utah 84092
E-Mail dmslaughter@wsu.edu
CELLULAR PHONE (801) 631-5919

CURRENT POSITIONS (Administrative, Academic, and Industrial)

Nuclear Labyrinth (CEO)
Research Professor Nuclear Radiation Center, Washington State Univ
MSI-Photogenics (VP of Research)

EDUCATION

Ph.D., University of Utah, Chemical Engineering, Dec. 1986
M.S., University of Utah, Chemical Engineering, Mar. 1983
B.S., University of Utah, Chemical Engineering, June 1980

EXPERIENCE

I have experience in nuclear engineering, radioenvironmental sciences, radioassays, radiotoxicology, chemical engineering and radiation and materials interactions including experience in regulatory matters with EPA, NRC and OSHA. I am also familiar with environmental monitoring, dose reconstruction, and nuclear forensics techniques and related analyses. My experience includes holding positions of Reactor Administrator, Reactor Supervisor, and Senior Reactor Operator at a TRIGA research nuclear reactor.

I am currently planning research studies for neutron radiography/tomography, and for SNM/chemical detection at the 250 KW Aerotest TRIGA Reactor and the University of Kentucky's linear accelerator facility. I have and continue to conduct research activities using: LANSCE (WNR) Facility, LANL, neutron detection and spectroscopy; FRM II – Antares, Technical University Munich, neutron/x-ray radiography and tomography; 5 MW Reactor, MIT , Fission Track Analysis (FTA); 2 MW TRIGA University of California-Davis, neutron radiography, nuclear forensics, material survivability and vulnerability; 1 MW TRIGA, Washington State University, NAA, FTA, Neutron induced autoradiography (NIAR), radiochemistry; 250 KW TRIGA, Aerotest San Ramon Ca, neutron radiography; 100 KW TRIGA, University of Utah, nuclear forensics, material survivability and vulnerability, FTA; Linear Accelerator at EAL, Ohio University, neutron detection and spectroscopy; CM-244 TOF facility, Colorado School of Mines, neutron detection and spectroscopy; 1 MW TRIGA Reactor, U-Mass Lowell, neutron detection and spectroscopy; 20 MW NIST Reactor, neutron detection, spectroscopy and imaging.

I have experience with computational/simulation codes: MCNP5, MCNP4C-Polimi, COG10, Scale 5, MathCad, Fluent, ProMax (process simulation), and Groundwater Vistas/Modflow-Surfact

A brief list of positions and relevant dates follow:

Nuclear Labyrinth -CEO and Research Scientist, (Formally E-Cubed), Feb. 1989-present
Vice-President of Research, Photogenics, Provo Utah, 3/1/2009-Present
Research Professor Nuclear Radiation Center, WSU, 8/2009- Present
Research Associate Radiobiology Division, Univ. of Utah 7/2009-1/2011
Nuclear Engineering Program, Professor-R (Univ. of Utah) 1990-6/2009
Civil and Environmental Engineering, Professor-R (Univ. of Utah) 1996-6/2009
Director of C.E.N.T.E.R., College of Engineering, (93- 04, 6/08-4/09)
Reactor Administrator of 100 Kw TRIGA Nuclear Reactor (95-04, 06/08-4/09)

MNRC Director and Reactor Administrator (Univ. of Ca-Davis), 2 Mw TRIGA Reactor, (6/1/04-9/30/05)
Nuclear Engineering Chair and Graduate Advisor (1995-2004)
Director of Environmental Radiation Toxicology Laboratory, School of Medicine Apr. 1995-Mar. 1999 (Merged with CENTER, Jan. 2000)
Chemical and Fuels Engineering, Univ. of Utah, Associate Professor-Research (1992-2004)
Mechanical Engineering, Univ. of Utah, Associate Professor-Research (1990-2004)
Senior Reactor Operator (NRC Lic.No. SOP-70052) (1992-2004)
Reactor Supervisor of 100 Kw TRIGA Nuclear Reactor, 1992-1999
Chemical Engineer, RAE, Jan. 1990-1995
Health Physics Auditor, Envirocare Inc., Jan. 1993-1995
Senior Engineer, Omega Components, Jan. 1992-2000
Research Director of NEL, Jan 1990-Jun. 1993
Chemical Engineer, GE/CRD, March 1987-Jan. 1990
Research Engineer, Energy & Environmental Research, Sept. 1984-Mar. 1987

COMMITTEES

(Institutional)

Univ.of Utah Reactor Safety Committee (1992-4/2009)
Univ.of Utah Radiation Safety Committee (1991-2004, 6/2008-6/2009)
College of Engr. Emergency Preparedness Committee (93-04, 06/08-4/2009)
Nuclear Engineering Admissions Committee (1995-2006)
Univ. of Ca –Davis Radiation Safety Committee (2004-2005)
Univ. of Ca –Davis MNRC Nuclear Safety Committee (2004-05)
Univ. of Ca –Davis Senior Management, Office of Research (2004-05)
Nuclear Engineering Admissions Committee (1993-2004)
RSC Reactor 1 Mw Upgrade Subcommittee (1994-2004)
Thermal, Fluids, and Energy Systems-ME/Dept. (1991-2004)
Nuclear Engr. Semester Conversion Committee (1997-99)
Environmental Engineering Executive Committee (1997-98)
Environmental Engineering Admissions Committee (1995-97)
Environmental Engr. Semester Conversion Committee (1997-98)
Minority Engineering Program Scholarship Committee (1991-95)
Univ.of Utah RSC Audit Subcommittee (1994-95)
GE-CRD Engineering System Safety Committee (1989-90)

(State/National)

Safety Based Academy- Office of the President Univ. of Ca.(2004-05)
TRTR Executive Committee (1999-2003)
NE Dept. Heads Org. (NEDHO) Committee (1999-2004)
National ANS-Alpha Nu Sigma Honor Society Chair (2003)
Reviewer for DOE Sponsored NEER Program (2000-2007)
Chair-Elect / Chair of TRTR (2001 / 2002)
Pres-Elect / Pres. of Great Salt Lake HP Chapter (1999-00/2000-01)
ANS/TRTR/AIChE/CIRMS Memberships

(Testimony) "Developing New Paradigms to Improve Educational Experiences and Support Unique Infrastructure in Nuclear Engineering and Nuclear-Related Disciplines," Testimony Before the U.S. Congress Science Committee-Energy Subcommittee, Given on June 10, 2003.

GRADUATE STUDENTS
Chairman/Supervisor:

Steven Godfrey, ME/Master of Science, 1993
Rian Smith, ME/Master of Science, 1993
Tony Zhou, ME/Master of Science, 1993
Ross Schmidtlein, NE/Master of Science, 1994
Henry Moeller, NE/Master of Science, 1994
Kevan Weaver, NE/Ph.D., 1998
Melinda Krahenbuhl, ChFE/Ph.D., 1998
Dong-Ok-Choe, NE/Ph.D., 2000
Ross Schmidtlein, NE/Ph.D., 2000
Christy Seiger-Webster, Envir.E/Master of Science, 1999
Stephannie Mecham, ChFE/Master of Science, 2000
Justin Wilde, NE/Master of Science, 2000
Todd Nethercott, NE/Master of Engineering, 2004
Russ Chazell, NE/ Master of Science, 2009

GRADUATE STUDENTS
Committee Member:

William Brown, ME/Master of Science, 1992
Yu Lung Hsieh, ME/Master of Science, 1993
Cindy Henderson, NE/Master of Science, 1993
Sharon Packer, ME/Master of Science, 1993
Randel Fox, ME/Master of Science, 1993
John E. Moore, ME/Ph.D., 1994
Robert Henderson, NE/Master of Science, 1994
K. MuPang, ME/Master of Engineering, 1994
Vern C. Rogers, NE/Master of Science, 1995
John Bennion, NE/Ph.D., 1996
Chi-Hua Tung, ME/Ph.D., 1994
Ke-Hwa Lee, ME/Master of Science, 1996
Byron Hardy, NE/Ph.D., 1996
Yu Lung Hsieh, ME/Ph.D., 1996
Nathan Yanasak, Phys./Ph.D., 1997
Brett Rogers, Envir.E/Master of Science, 1998
Robert Hayes, NE/Ph.D., 1999
Barbara O'Connor, Anthropology/Master of Science, 2000
Tom Resmussen, NE/Master of Science, 2001
Roger Myers, NE/Master of Science, 1999
Heidi Walk, NE/ Master of Science, 2003
Justin Wilde, NE/Ph.D., 2004
Tom Maddock, NE/ Master of Science, 2004
James Parry, NE/Ph.D., 2005
Adina Soaita, Biostatistics/Ph.D., 2006
John Bess, NE/Ph.D., 2008
Sang Ku Lee NE/ Master of Science, 2007
Jesse Reeves NE/ Master of Science, 2007

UNDERGRADUATE THESIS

Chairman/Supervisor: (*- Recipients of the National Academy for Nuclear Training &/or ANS Scholarships)

Melinda Krahenbuhl, ChFE/BS, 1993
Gerry Bowers, ChFE/BS, 1994
Diana Jensen*, ChFE/BS, 1995
Michelle Brown-Walker*, ChFE/BS, 1996
Stephannie Mecham*, ChFE/BS, 1996
Bryon Lawance*, ChFE/BS, 1997
Adam Rogers*, ChFE/BS, 1997
Justin Wilde, ChFE/BS, 1997
Heidi Walk*, ChFE/BS, 2001
David Lignell*, ChFE/BS, 2001
Margaret Fitch, ChE/BS, 2009

UNDERGRADUATE DESIGN

Chairman/Supervisor:

Son Tu, ME/BS, 1991
Viset Ong, ME/BS, 1991
Gordon Willis, ME/BS, 1991
Micheal Willis, ME/BS, 1991
Vinh Tang, ME/BS, 1992
Dien Ngo, ME/BS, 1993
Lam Thai, ME/BS, 1993
Eric Wallentine, ME/BS, 1994
Tomas Cantrell*,ChFE/BS, 1998

COURSES

(Design/Taught)

ME/CE 234 Dynamics I
ME EN 570 Introduction to Nuclear Engineering
ME EN 572 Radioactivity in the Environment
ME EN 577 Nuclear Reactor Engineering Laboratory
ME EN 579 Radiation Monitoring
ME/ChFE 480 Design Project
CVE EN 5700 / 6700 Nuclear Engineering I / II
CVE EN 5710 / 6710 Applied Nuclear Engineering I / II
CVE EN 6720 Nuclear Reactor Physics
CVE EN 6740 Nuclear Environmental Engineering
CVE EN 6750 Nuclear Chemical Engineering
CVE EN 7710 Adv. Nuclear Engineering Design
CVE EN 7720 Adv. Nuclear Phenomena

1. Recipient of 1995-1996 University of Utah's Student Choice Award for Excellence in Teaching

2. Recipient of 1998 INEEL Faculty Summer Fellowship (UT-49-98)

CONTRACTS
(Principal Investigator)

"Advanced Elastic/Inelastic Nuclear Data Development: Optimizing Isotopic Composition of the LGB Scintillator for Monitoring Multi-Energy Neutrons", DOE NEUP, ISU Subcontract, 2009-2012.

"Neutron Spectroscopic Dosimeter", Photogenics, SBIR Phase II, Contract # NNX09CA21C, 2009-2011.

"Diagnosing Structural Health in Advanced Nuclear Energy Systems " DOE Contract DE-FG07-07ID14848, 2007-2009.

"Neutron Irradiation of Electronic Components for the U.S. Air Force (HAFB) by the University of Utah-CENTER-Base, Option 1, 2, 3, and 4 U.S. Air Force, Hill AFB Utah, 2005-2009.

"Health Effects of Radiation Exposures in Russian Workers Subcontract-Univ. of Pittsburgh", NIH Contract No. 1R01 0H007866-01A1, 2004-2008

"University Reactor Instrumentation Program-University Reactor Infrastructure and Education Support," DOE Contract No. DE-FG07-04ID14579, 2004-2009

"DOE University Reactor Sharing Program," DOE Contract No. DE-FG07-02ID14372, 2004-2006

"Development of a Large-Field Cold Neutron Source at the UC Davis McClellan Nuclear Radiation Center," DOE Contract No. DE-FG07-03ID14499, 2004-2006

"Innovations in Nuclear Infrastructure and Education, Subcontract-Oregon State Univ." DOE Contract No. DE-FG07-02ID14422, 2004-2007

"Neutron Irradiation of Electronic Components for the U.S. Airforce (HAFB) by the University of Utah-CENTER-Base, Option 1, 2, and 3" Airforce, Hill AFB Utah, Contract No. F4265001-C0015, 2001-2005.

"University Reactor Instrumentation Program-Sample Characterization and Remote Sensing," DOE Contract No. DE-FG07-03ID14478, 2003-2004.

"DOE University Reactor Sharing Program," DOE Contract No. DE-FG07-02ID14391, 2002-2004.

"Center of Excellence: Center for Nuclear, Medical and Environmental Technologies," State of Utah-Economic Develop Office, Contract No. 020188, 2002-2004.

"University Reactor Instrumentation Program-Reactor Exhaust/Ventilation Upgrade," DOE Contract No. DE-FG07-02ID14281, 2002-2004.

"DOE/Industry Matching Grant to Support Nuclear Engineering," DOE Contract No. DEFG0703ID14521, 2003-2004.

"FTA / Pu Analysis from Bioassay," DOE EH-63-Rongelap Atoll, DOE Contract No. DEFC0397SF21354, 1999-2003.

"University Reactor Instrumentation Program-Heat Exchanger Upgrade The Sequel," DOE Contract No. DE-FG07-01ID14078, 2001-2003.

"University Reactor Instrumentation Program-Heat Exchanger Upgrade and Specially Designed Exterior Barriers," DOE Contract No. DE-FG07-00ID13947, 2000-2003.

"Nuclear Analytical Services" Contract from E-Cubed Inc., Contract No. E31001, 1993-2002.

"DOE/Industry Matching Grant to Support Nuclear Engineering," DOE Contract No. DEFG0300NE38167, 2000-2002.

"Computational and Analytical Support for the HEU/SIO2 Nuclear Criticality Benchmark Experiments Conducted at the Russian Institute of Physics and Power Engineering," Idaho State Univ. / INEEL Contract No. 996898, 1999-00.

"FTA / Pu Analysis from Bioassay," Rongelap Atoll Local Government, Authorization Letter dated 7/26/99, 1999-00.

"Nuclides at Paducah," DOE Contract No. DEFC0397SF21315, 1998-01

"University Reactor Instrumentation Program-Security Systems Upgrade, Enhancement in Surveillance Capabilities," DOE Contract No. DE-FG07-99ID13796, 1999-00.

"DOE University Reactor Sharing Program," DOE Contract No. DE-FG03-95NE38134, 1999-00

"Establishment of a TRIGA Nuclear Reactor Facility at the University," INEEL DOE Contract No. DEAC0776ER02238, 1998-99.

"Evaluate Potential BNCT Filter Designs for WSU Using Monte Carlo Codes (MCNP and COG)", INEEL-AWU, 1998.

"FTA / Pu Analysis for Internal Exposure" National Academy of Science Contract No. RC *80198, 1998.*

"University Reactor Instrumentation Program-Security System Upgrade, Funds for Transport of Donated Reactor Components, and Gamma and Alpha Spectroscopy Systems Up-Grade," DOE Contract No. DE-FG03-92ER79135, 1998.

"DOE University Reactor Sharing Program," DOE Contract No. DE-FG03-95NE38134-A000, 1998.

"Neutron Irradiation of Electronic Components for the U.S. Airforce (HAFB) by the University of Utah-CENTER-Base, Option 1, 2, and 3" Airforce, Hill AFB Utah, Contract No. F4265098-C0035, 1997-2001.

"Evaluation Tests for American Excelsior Aspen Pads," American Excelsior, Jon Traine, Contract No. 470114c, 1998.

"Radioactive Analytical Services" Layton Construction Company, Contract No.165-01813-X, 1998.

Three Type I (3-months) IAEA Fellowships (INS/95018P, 95019P, 95049P), Jan.-Apr. 1997.

"Evaluation Tests for American Excelsior Aspen Pads," American Excelsior, Jon Traine, Contract No. 470114b, 1997.

"FTA /Pu and U Analyses," Colorado State University, Contract No. P151607, 1996-97.

"Baseline Study of Plutonium Excretion in Workers on the NTS Site," Bechtel Nevada (REECO) at NTS, Contract No. 2621 (01967-CUJ), 1995-97.

"Neutron Irradiation of Electronic Components for the U.S. Airforce (HAFB) by the University of Utah Nuclear Engineering Laboratory-Base, Option I and II," Airforce, Hill AFB Utah, Contract No. F4265095-C0044, 1995-97.

"FTA /Pu and U Analyses," Colorado State University, Contract No. P143994, 1995-96.

"Radioactive Analytical Services" Park City Construction, Contract No.PC1001, 1995.

"Determination of Pu-239 by Fission Track Analysis," New Mexico State University, Contract No. PS 57122, 1995.

"DOE University Reactor Sharing Program," DOE Contract No. DE-FG03-95NE38134-A000, 1995-1997.

"Evaluation Tests for American Excelsior Aspen Pads II," American Excelsior, Contract No. 470114, 1995.

"Ultra-Sensitive Detection of Pu-339 by Fission Track Analysis", University of Utah/ DOE, 1995-96.

"Tritium Wipe Strategies for Defense Products," BEMSCO Inc., Contract No. 39313, 1994-97.

"Evaluation Tests for American Excelsior Aspen Pads," American Excelsior, Contract No. 110572, 1994.

"DOE University Reactor Sharing Program," DOE Contract No. DE-FG03-93ER75812-A002,1994.

"Transportation Risk Assessment Services/C.E.N.T.E.R.," US DOE Albuquerque NM, Order No. DE-AP04-94AL99291, 1994.

"Dosimetry for Neutron Irradiation of Electronic Components," Sandia National Laboratory, Sandia Contract No. AK-9297, 1994.

"Performance Testing of Various Materials Used in Drums and Containers," Westinghouse Hanford Co., Award No. MJG-SVV-343433, 1994.

"Sample Irradiations" US Bureau of Mines, Contract No. B4040021, 1994.

"Neutron Irradiation of Electronic Components for the U.S. Airforce (HAFB) by the University of Utah Nuclear Engineering Laboratory (NEL), Airforce, Hill AFB Utah, Contract No. F4265094-M0125 1993-1995.

"DOE University Reactor Sharing Program," DOE Contract No. DE-FG03-93ER75812-A002,1993.

"University Reactor Instrumentation Program-Liquid Scintillation Counter for Contamination Monitoring," DOE Contract No. DE-FG03-93ER79135.001, 1993.

"Tests for Stabilized Waste Forms," Westinghouse Hanford Co. Award No. MJG-SVV-309742, 1993.

"Actinide Removal From Organic Liquids," Westinghouse Hanford Co. Award No. MJG-SVV-343419, 1993.

"Enhanced Low Temperature Oxidation of Organic Waste Treated with Gamma Radiation," Westinghouse Hanford Co. Award No. MJG-SVV-309720, 1993.

"Sample Irradiations" US Bureau of Mines, Contract No. B4030019, 1993.

"DOE University Reactor Sharing Program," DOE Contract No. DE-FG03-92ER75812,1992.

"University Reactor Instrumentation Program-Hand and Foot Radiation Surveying Safety System," DOE Contract No. DE-FG03-92ER79135, 1992.

"Radioactive Characterization Project-A Guide for Low Level Waste Characterization," Westinghouse Hanford Co. Award No. MJG-SVV-204498, 1992.

"Rapid and Safe Pneumatic Driven Sample Delivery System,"Equipment Research Grant, University of Utah, 1992.

"The Radioactive Incineration Project-Organic Material Destruction By Gamma Radiation," Westinghouse Hanford Co. Award No. MJG-SVV-204499, 1992.

"A Parallel Processing-Human Driven Engineering Computing System for Enhanced Process Control of a Nuclear Reactor System," NSF-REG Grant. 1991.)

"Influencing of Trace Mineral Elements on Radiation Induced Cataracts," Biomedical Program Grant, University of Utah 1990.

"Designing a Monochromatic Neutron Condenser," Faculty Grant, University of Utah 1990.

"Influence of Coal Mineral Matter of Effectiveness of Dry Sorbent Injection for SO_2 Control," EPA Contract No. 68-02-3987.1985

CONTRACTS...

(Co-Investigator)

"Health Effects of Radiation Exposures in Russian Workers Univ. of Pittsburgh", NIH Contract No. 1R01 CCR312952, 2004-2008

"JCCRER Project 2.4, Dosimetry Studies in Support of EH-63 Programs-COE Subcontract," DOE Contract No. DE-FC0397SF1354, 1998-2002.

"Training and Technical Support for the Community Radiation Monitoring Program," DOE Contract No.DEFC0890NV10884, 1991-1994.

"Preliminary Investigation of the Transportation Risks and Economic Impact and Benefits Associated with the Siting of an MRS in Utah,"Office of the Nuclear Waste Negotiator, Wash. DC, Contract No. UU/CENTER-1, 1994.

"Preliminary Investigation of the Socioeconomic Impacts Associated with the Potential Siting of an MRS in Tooele Utah,"Office of the Nuclear Waste Negotiator, Wash. D.C. Contract No. UU/CENTER-2, 1994.

"Advanced Coal-Fueled Gas-Turbine Program," DOE Contract No. DE-AC21-86ML23168.

"Coal-Fueled Diesel Technology Development," DOE Contract No. DE-AC21-88MC23174.

"Bench-Scale Combustion Studies of NOx and SOx Formation and Control," EPA Cooperative Agreement CR-809267.

(Project Officer)

"5.4 Protocol Laboratory Film Blackening Study Due to Beta-Irradiation"DOE/CENTER Contract to Northop Grumman, Hill AFB Ogden Utah. 2003.

"Nuclear Engineering Services," Contract to Nuclear Radiation Center, Washington State University, 1993.

"Injection of Calcium-Based Sorbents for SO_2 Control in a Coal-Fueled Diesel Exhaust," DOE/GE Subcontract to Chemical Engineering Department, University of Utah. 1989.

"SO_2 Capture in Advanced Coal-Fired Gas Turbines Systems Using Calcium-Based Sorbents," DOE/GE Subcontract to Physical Science Technology, Inc. 1988.

PUBLICATIONS (Graduate Student in Italics)

(Journal)

Slaughter et al, "Validation of Measured and Reconstructed Individual External and Internal Doses in The Mayak PA Workers' Early Clinical Effects (MWECE) Database", Manuscript being reviewed for submission to Health Physics Journal, 2012.

John D. Bess, Krahenbuhl M.P., Miller S.C., Slaughter DM Khokhryakov, V.F., Suslova K.G., Vostrotin V.V. Uncertainties Analysis for the Plutonium Dosimetry Model, Doses 2005, using bioassay data Health Physics 93(3):207-219; 2007.

Azizova, Tamara V., Niel Wald†, Sergey V. Osovets, Richard D. Day, Maria B. Druzhinina, Margarita V. Sumina, Valentina S. Pesternikova, Igor I. Teplyakov, Aimin Zhang, Michael Kuniak, Evgeny K. Vasilenko, David M. Slaughter, Laura Cassidy Schal, Nadezhda D. Okladnikova, "Dose Assessment and Prediction of Acute Radiation Syndrome Severity," Accepted to Journal of Radiation Research, August 2007.

Kuniak, M., D.O., T. Azizova, M.D., R. Day, Ph.D., N. Wald, M.D., M.V. Sumina M.D., V.S. Pesternikova, M.D., E. Vasilenko, Ph.D., A. Soaita, M.D., D.M. Slaughter, Ph.D. "The Radiation Injury Severity Classification (RISC) System: A Rapid Triage System for Acute Radiation Injury Situations," Accepted in The British Journal of Radiology, June 4, 2007.

Krahenbuhl, M.P., *J.D. Bess, J.L. Wilde*, V.V. Vostrotin, A.E. Schadilov, K.G. Suslova, V.K. Khokhryakov, D.M. Slaughter, S.C. Miller, "Uncertainties Analysis of Doses resulting from Chronic Inhalation of Plutonium at the Mayak Production Association", Health Physics 89(1); 33-45 2005.

Choe, Dong-Ok, Brenda N. Shelkey, Justin L. Wilde, Heidi A. Walk, and David M. Slaughter, "Calculated Organ Doses for Mayak production Association Central Hall Using ICRP and MCNP," Health Physics Jorunal,Vol.84, No. 3, March 2003.

Slaughter, David M., Dong-Ok Choe, Melinda P. Krahenbuhl, Scott C. Miller, Evgenii Vasilenko, Michail Gorelov, "Reconstruction of Individual External Exposure Doses to the Mayak Production Association Workers: External Doses 2000," Submitted to Health Physics Journal 2003.

Krahenbuhl M.P., D.M. Slaughter, J.L. Wilde, J.D. Bess, S.C. Miller, V.F. Khokhryakov, K.G. Suslova, V.V. Vostrotin, S.A. Romonov, Z.S. Menshikh, T.I. Kudryavtseva, "The historical and current application of the FIB-1 model to assess organ dose" Health Physics 82(4):445-454 2002.

Seiger-Webster, Christy. M., David M. Slaughter, and Saskia Duyvesteyn,"Evaluating and Improving Process Methodology for Ion Exchange Columns used in Fission Track Analysis," Journal of Radioanalytical and Nuclear Chemistry, (Manuscript Submitted 2000).

Choe, D.O., D.M. Slaughter and KD Weaver, "Utilizing Distinct Neutron Spectra and a System of Equations to Differentiate Competing Reactions in Activation Analysis," Journal of Radioanalytical and Nuclear Chemistry, Volume 244, No. 3, 2000.

Khokhryakov V.F., K.G. Suslova, E.E. Aladova, E. Vasilenko, S.C. Miller, D.M. Slaughter, M.P. Krahenbuhl, "Development of an Improved Dosimetry System for the Workers at the Mayak Production Association", Health Physics 79 (1): 72-76

Krahenbuhl M.P., J.L. Wilde, D.M. Slaughter, "Using Plutonium Excretion Data to Predict Dose from Chronic and Acute Exposures", Rad. Prot. Dos. Vol. 87 No.3 (2000) 179-185

Krahenbuhl M.P. and D.M.Slaughter,"Improving Process Methodology for Measuring Plutonium Burden in Human Urine Using Fission Track Analysis," Journal of Radioanalytical and Nuclear Chemistry, Volume 230, No. 1-2, 1998.

Sandquist, G.M., D.M. Slaughter, C.Y. Kimura, and G. Brumburgh,"Determining Cutoff Distances for Assessing Risks from Transportation Accident Radiation Releases," Trans American Nuclear Society, Vol. 76, pg. 100, October 1995.

Moore,J.E., G.M. Sandquist, D.M. Slaughter, "A Route and Event Specific System for Risk Assessment of Radioactive and Hazardous Materials Transportation Accidents," Nuclear Technology Journal, Vol. 112, No.1, pg. 63-78, October 1995.

Sandquist,G.M., D.M. Slaughter, V.C. Rogers, "Balancing Health Effects and Economics in the Management of Radioactive Materials," Trans American Nuclear Society, Vol. 74, pg.110, November 1994.

Moore, J.E., G.M. Sandquist, D.M. Slaughter, "Radioactive and Hazardous Materials Transportation Risk Assessment using a Geographic Information System," Radwaste Journal, pg. 75-78, January 1994.

Sandquist, G.M., *J.S. Bennion, J.E. Moore*, D.M. Slaughter, "A Route Specific Transportation Risk Assessment Model," Trans American Nuclear Society, Vol. 68, June 1993.

Sandquist, G.M., *J.S. Bennion, J.E. Moore*, D.M. Slaughter, C.Y. Kimura, "Risk Assessment of Radioactive and Hazardous Materials in DOE Defense Packages in Transportation Accidents," Journal of Radioactive Materials Transport (RATRANS) Vol.3, No.2/3 pg. 121-128, March 1993.

Crawford, K.C., G.M. Sandquist, D.M. Slaughter, "Control Algorithms for Quasi-Steady State Reactor Operation," Trans American Nuclear Society, Vol. 60, November 1991.

Sandquist, G.M., D. M. Slaughter, V.C. Rogers, "Econometric Model for Age and Population Dependent Radiation Exposures," Trans American Nuclear Society, Vol. 60, November 1991.

Taheri, M., G. Sandquist, D. Slaughter, *J. Bennion*, "Graphite Epoxy Composite Degradation by Space Radiation," Trans American Nuclear Society, Vol. 60, November 1991.

Slaughter, D.M., S.L. Chen, D. Kirchgessner, "Promotion of Calcium-Based with Alkali Metals and Chromium for Increased SO_2 Removal," Twenty-second Symposium (International) on Combustion, The Combustion Institute, Pittsburgh, Pa., 1989.

Slaughter, D.M., B.J. Overmoe, D.W. Pershing, "Inert Pyrolysis of Stoker Coals Fines," Fuel, Vol. 67, April 1988.

Silcox, G.D., D.M. Slaughter, D.W. Pershing, "High Temperature Sulfation Studies in an Isothermal Reactor: A Comparison of Theory and Experiment," Twentieth Symposium (International) on Combustion, The Combustion Institute, Pittsburgh, Pa., 1985.

Slaughter, D.M., D.W. Pershing, D.C. Drehmel, G.B. Martin, "Parameters Influencing the Evolution and Oxidation of Sulfur in Suspension-Phase Coal Combustion," Nineteenth Symposium (International) on Combustion, The Combustion Institute, Pittsburgh, PA., 1983.

Starley, G.D., J.M. Munro, D.M. Slaughter, D.W. Pershing, "Formation and Control of NOx Emissions from Coal-Fired Speader-Stoker Boiler," Nineteenth Symposium (International) on Combustion, The Combustion Institute, Pittsburgh, Pa., 1983.

(Pre-Journal)

Choe, D.O. S. Mecham, D.M. Slaughter, "The Theoretical Development of Exposing Biological Specimens to Differing Neutrons Spectrums for Quantifying the Concentration and Distribution of Plutonium-239 and Uranium," Nuclear Science and Technologies (Manuscript being written).

Weaver, K., and D.M. Slaughter, "Preparation of High Specific Activity, Carrier-free Mo-99, P-32, and F-18 in a Non-Power Research Reactor Using Powder Recoil Techniques," Radiochimica Acta (Manuscript being written).

CONF. PROCEEDINGS (Graduate Student in Italics)
(Reprints avail.)

Slaughter, D.M. et al, "Neutron Detectors with Effective Gamma Discrimination using (Inorganic/Organic) Composite Scintillators" accepted for presentation SORMA West, Oakland Ca Sept 2012.

Slaughter, D.M. et al, "An Effective Spectrometer for Neutron Energies between 0.8 MeV and 150 MeV using $^{6}Li^{nat}Gd^{10}B_3O_9$:Ce / Poly-vinyl Toluene Composite Scintillator and Advanced Signal Analysis" accepted for presentation at SORMA West, Oakland Ca Sept 2012.

Slaughter et al, "Neutron Dosimeter with effective Gamma Discrimination using (Inorganic/Organic) Composite Scintillators" Health Physics Society National Meeting Sacramento Ca, July 2012

Slaughter D.M., "Neutron Detectors" (WSU contribution) for Advanced Elastic/Inelastic Nuclear Data Development NEUP Meeting, Dallas Tx, Oct 2011.

Slaughter D.M., "Developing a Neutron Spectrometer for Area Monitoring Applications" NRC Workshop WSU June 2010.

Slaughter, D.M., R. Tsukimura, B. Schillinger, and C.M. Sim . "Military Aircraft Structural Health (MASH) Initiative," Hill AFB/Univ. of Utah Workshop on Joint Research on Sustainment Technologies, College of Engineering, University of Utah, May 25, 2006.

Slaughter, D. M., (and C.M. Sim) "International Aircraft Health Program for the F-15 and F-16" Invited Speaker, Korea Atomic Energy Research Institute Workshop On Non-destructive Testing Methods for Military Aircraft, HANARO, South Korea, Nov. 2005.

Kahenbuhl,M.P., J.D.Bess J.L. Wilde,V.V. Vostrotin, K.G. Suslova,,V.F.Khokhryakov, D.M. Slaughter and S.C. Miller Statistical Analysis of Dose Assignments Resulting from Plutonium Bioassays. Annual Meeting Health Physics Society, Spokane Washington, July 2005

Slaughter, D.M., "Update on MNRC Activities & International Collaboration Development," MNRC Research Advisory Committee Meeting, Institute Nacional de Investigaciones Nucleares (ININ)- Fuente de Gammas Mexico City, May 16, 2005.

Slaughter, D.M., "WNSA Status Update-MNRC," WNSA workshop, Oregon State University, Oregon, May 3, 2005

Slaughter, D.M., "Research opportunities and challenges at the UC Davis McClellan Nuclear Radiation Center (MNRC) and Other University Nuclear Reactor Programs," Lawrence Livermore National Laboratory, California, April 29, 2005.

Slaughter, D.M., "Neutron tomography and radiography status," Neutron Tomography and Radiography Workshop (Worldwide Teleconference), Spallation Neutron Source-Oak Ridge National Laboratory/University of Tennessee, April 11, 2005.

Slaughter, D.M., "MNRC Research Capabilities," UC Davis Research Community Meeting, University of California-Davis, March 4, 2005

Slaughter, D.M., "WNSA Status Update-MNRC," WNSA workshop, Idaho National Laboratory, October 18-19, 2004.

Slaughter, D.M., "WNSA Members (MNRC) & Pacific Northwest National Laboratory Collaborations", Pasco, WA, September 20-21, 2004.

Slaughter, D.M., "Overview of UC Davis MNRC Research and Educational Activities," Northern California Section-American Nuclear Society, University of California-Berkeley, CA, December 1, 2004.

Choe, D.O., B.N. Shelkey, D. M. Slaughter, "An Investigation Comparing The Criticality of Stored DOE Waste Using MCNP with Previously Published Results Obtained with KENO", HPS Joint Midyear meeting Anaheim Ca. Feb. 2001

Alexandrova O.N., E.K. Vasilenko, M.P. Krahenbuhl, D.M. Slaughter," The Statistical Analysis of Occupational Radiation Dose Caused by Professional Exposure to External Gamma Radiation" International Data Analysis Conference, Innsbruck, Austria, Sept. 2000.

Krahenbuhl M.P., D.M. Slaughter, S.C. Miller, V.F. Khokhryakov, E. Vasilenko, " Overveiw of the historical doses at Mayak. Health Physics Society Meeting – Columbia Chapter Nov. 2000

Krahenbuhl M.P., D.M. Slaughter, S.C. Miller, V.F. Khokhryakov, "Summary of the Mayak PA internal dosimetry program", Bioassay, Analytical and Environmental Radiochemistry Conference, Seattle WA. November, 2000

Slaughter D.M., M.P. Krahenbuhl, "Redefining an Administrator's Perception of a Reactor Program Through New Paradigms (Just reactor Guys)", National Organization of Test, Research and Training Reactors Raleigh, North Carolina Oct. 2000

Krahenbuhl M.P., D.M. Slaughter, "Radiation doses associated with the early operation of Mayak" National Organization of Test, Research and Training Reactors Raleigh, North Carolina Oct. 2000

Choe, D.O., M.P. Krahenbuhl, and D.M. Slaughter,"Dose Reconstruction from Pu Exposure Using Fission Track Analysis (FTA) with Two Neutron Energy Spectra ," Workshop on Standards, Intercomparison and Performance Evaluations of Low-Level Radionuclides by Mass Spectrometry and Atom Counting. Gaitherburg, Maryland 1999.

Weaver, K.D and D.M. Slaughter,"Producing Fluorine-18 in Research Nuclear Reactors," MARC V Conference, Hawaii, 2000.

Choe, D.O., D.M. Slaughter, *K.D. Weaver*, "Utilizing Distinct Neutron Spectra and a System of Equations to Differentiate Competing Reactions in Activation Analysis," Tenth International Conference on Modern Trends in Activation Analysis, NIST, Bethesda Maryland, April 19-23, 1999.

Krahenbuhl, M.P., and D.M. Slaughter, "Recent Advancements in the Ultra-Sensitive Detection of Pu Using Fission Track Analysis," Workshop on Standards, Intercomparison and Performance Evaluations of Low-Level Radionuclides by Mass Spectrometry and Atom Counting. Gaitherburg, Maryland 1999.

Sandquist, G.M., D.M. Slaughter, C.Y. Kimura, and D.L.Sanso, "Risk Assessment of High Altitude Free Flight Commercial Aircraft Operations," International Conference on Probabilistic Safety Assessment and Management, New York NY, September 1998.

Kunze, J.F., R. Kudchadker, G.M. Sandquist, D.M. Slaughter, "The Future of Nuclear Power with Respect to Heath Effects of Radiation," ICONE-6, 6th International Conference on Nuclear Engineering, San Diego California., May 10-15, 1998.

Krahenbuhl M.P., and D.M. Slaughter, "Developing Fission Track Analysis to Determine Plutonium Excreted and Reconstruct Uptake to Predict Exposure and Dose," 1998 ANS Radiation Protection Shielding Division, Topical Conference Technologies for the New Century, Nashville Tennessee, April 1998.

Choe.D.O., and D.M. Slaughter, "Two Distinct Energy Spectra for the Measurement of Pu-239 Burden in Human Excreta Using Fission Track Analysis," 1998 ANS Radiation Protection Shielding Division, Topical Conference Technologies for the New Century, Nashville Tennessee, April 1998.

Sandquist, G.M., D.M. Slaughter, and C.Y. Kimura, "Improved Assessment of Aviation Hazards to Ground Facilities using a Geography Information System," ANS International Topical Meeting on PSA 96: Probabilistic Safety Assessment" Park City Utah, Sept-Oct. 1996.

Slaughter, D.M., G.M. Sandquist, "Route Specific Risk Assessments for Spent-Fuel Transportation Through Utah and Nevada," Patram-5, Las Vegas NV, December 1995.

Sandquist, G.M., D.M. Slaughter, C.Y. Kimura, "Risk Assessment for Aircraft Flight Operations at Salt Lake International Airport," ASME-IMECE Conference, November 1995.

Sandquist, G.M., D.M. Slaughter, C.Y. Kimura, C.T. Bennett, S. Smith, "Evaluation of Aviation Hazards to Ground Facilities using a

Geography Information System," ASME Pressure and Piping Conference, Honolulu HI, July 23-27, 1995.

Kimura. C.Y., C.T. Bennett, G.M. Sandquist, D.M. Slaughter, S. Smith, "Aircraft Accident Data Development for Aircraft Risk Evaluation to Ground Facilities Through the Use of a GIS System," ASME Pressure and Piping Conference, Honolulu HI, July 23-27, 1995.

Slaughter, D.M., G.M. Sandquist, *T. Zhou*, " HITRA-A Specific Integrated Transportation Risk Assessment System." U.S. DOE Transportation Management Workshop, Rockville MD., August 8-11, 1994.

Sandquist, G.M., D.M. Slaughter, S.B. Fellows," Risk Assessment of adioactive and Hazardous Materials in DOE Defense Materials Transportation Accidents" 1994 Defense Program Packaging Workshop, Knoxville, Tn., May 16-20, 1994.

Bowers,B.,(and D. Slaughter), "Electrochemical Separation of Actinides from PCB" AICHE Western Regional Conference, Golden Co., March 1994.

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ATTACHMENT (6)

INITIAL YEAR INCOME ESTIMATE FOR AEROTEST OPERATIONS.

[redacted]

ATTACHMENT (6)

INITIAL YEAR INCOME ESTIMATE FOR AEROTEST OPERATIONS

[redacted]

In US $:

Annual Revenue [redacted]

Cost Recoverable Activities	Unadjusted Cost	Fuel Credit	NRCFees	Decon Fee	Total Cost
Task 1. Neutron Interrogation by Radiography	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Task 2. Nuclear Science and Engineering Research	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Task 3. Training	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Task 4. Neutron Sci/Engr Research (w/o use of ARRR)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

Annual Revenue	[redacted]
Total Annual Cost	[redacted]
Profit	[redacted]
St Inc. Tax [redacted]	[redacted]
	[redacted]
Fed Inc. Tax [redacted]	[redacted]
Net Profit	[redacted]

Aerotest Operations, Inc.
May 30, 2012

ATTACHMENT (7)

[redacted]

5 YEAR PROJECTED INCOME STATEMENT FOR AEROTEST OPERATIONS, INC.

Aerotest Operations, Inc.
May 30, 2012

ATTACHMENT (7)

5 YEAR PROJECTED INCOME STATEMENT FOR AEROTEST OPERATIONS, INC.

[redacted]

In US $:

	1 year	2 year	3 year	4 year	5 year
Annual Revenue	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Cost Recoverable Activities	Total Cost				
	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Task 1. Neutron Interrogation by Radiography	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Task 2. Nuclear Science and Engineering Research	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Task 3. Training	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Task 4. Neutron Sci/Engr Research (w/o use of ARRR)	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Annual Revenue	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Total Annual Cost	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Profit	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
St Inc.Tax-[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Fed Inc.Tax-[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Net Profit	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

ATTACHMENT (8)

ENERGY SOLUTIONS ARRR DECOMMISSIONING COST STUDY.

ATTACHMENT (8)

ENERGY SOLUTIONS ARRR DECOMMISSIONING COST STUDY

ENERGYSOLUTIONS

CS-HP-PR-007

Decommissioning Plan for the Aerotest Radiography and Research Reactor

San Ramon, California

Project No. 313150 **Revision 0**

Prepared for:
Aerotest Operations, Inc.

Prepared by:
Energy*Solutions*, LLC
Commercial Services Division
1009 Commerce Park Drive, Suite 100
Oak Ridge, TN 37830

Authored By:	Paul Ely, Engineer	Date 11/17/11
Reviewed By:	Steve Croslin, CHP	Date 11/17/11
Approved By:	Ben Sklar, Project Manager	Date 11/17/11

[X] New Report
[ ] Title Change
[ ] Report Revision
[ ] Report Rewrite

Effective Date ____________

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1.0 ACRONYMS AND ABBREVIATIONS

°	Degrees
°F	Degrees Fahrenheit
Aerotest	Aerotest Operations, Inc.
AGN	Aerojet-General Nucleonics, a division of General Tire
AGNIR	Aerojet-General Nucleonics Industrial Reactor
ARRR	Aerotest Radiography and Research Reactor
CA	State of California
Ci	Curies
Co	Cobalt
COC	Contaminants of Concern
D&D	Deactivating and Decommissioning
DCGL	Derived Concentration Guideline Level
DOE	U.S. Department of Energy
FSS	Final Status Survey
FSSP	Final Status Survey Plan
HSA	Historical Site Assessment
L	Liter
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
mCi	MilliCuries
mR/hr	MilliRoentgen per hour
NRC	U.S. Nuclear Regulatory Commission
pCi	PicoCuries
pCi/g	PicoCuries per Gram
pCi/L	PicoCuries per Liter
RHB	California Radiologic Health Branch of California Department of Health
ROC	Radionuclide(s) of Concern
SAFSTOR	A method of decommissioning in which a nuclear facility is placed and maintained in a condition that allows the facility to be safely stored and subsequently decontaminated (deferred decontamination) to levels that permit release for unrestricted use.
TRIGA	Teaching Research Isotope General Atomic reactor
TS	Technical Specification
USNRC	U.S. Nuclear Regulatory Commission

2.0 EXECUTIVE SUMMARY

EnergySolutions performed an independent cost estimate for decommissioning the Aerotest Radiography and Research Reactor located in San Ramon, California. This estimate was prepared at the request of Aerotest Operations, Inc., (Aerotest).

This cost estimate was developed using a systematic approach. Decommissioning criteria were identified and survey data were reviewed. Specific and general information regarding equipment and structures was used in determining decontamination and demolition methodologies in order to minimize overall decommissioning costs.

This estimate includes itemized costs for manpower and equipment resources, radioactive waste volume reduction, packaging, shipping and burial activities, the performance of final status surveys for buildings and structures. The estimated decommissioning cost is $3,285,800 in terms of 2011 dollars. This estimate **does not** include the costs associated with fuel removal and transport from ARRR to the Department of Energy (DOE). However, ARRR has a DOE contract amendment that establishes the fee for the disposal of the spent nuclear fuel generated by ARRR (U.S. Department of Energy Contract Amendment to DE-CR01-83NE44484). This decommissioning estimate is for budgetary purposes only and is not a proposal for EnergySolutions to perform the decommissioning work.

A significant portion of the overall decommissioning costs is attributed to the disposal of radioactive waste. The radioactive waste disposal rate used for most of the waste in this estimate was based on shipping to the disposal site at Clive Utah.

3.0 SUMMARY OF THE PLAN

3.1 INTRODUCTION

3.1.1 Overview

Aerotest Operations has provided Neutron Radiographic (N-Ray) Inspection Services since 1969 using the Aerotest Radiography and Research Reactor (ARRR) for the neutron source.

Aerotest Operations, Inc., (Aerotest) is the holder of Facility Operating License No. R-98 which authorizes the possession, use and operation of the Aerotest Radiography and Research Reactor (ARRR) located in San Ramon, California. Aerotest is a wholly owned subsidiary of OEA Aerospace, Inc., which is wholly owned by OEA, Inc. OEA, Inc., was purchased by Autoliv ASP, Inc., (Autoliv). Autoliv is owned by Autoliv, Inc., a Delaware corporation with a Board of Directors and Executive Officers the majority of whom are non-U.S. citizens and as a result Aerotest became a subsidiary of Autoliv and Autoliv, Inc.

The NRC's position and regulations in 10 CFR 50.38 do not allow issuing a license for a production or utilization facility to an alien or an entity that is owned , controlled, or dominated by foreign interests. There was a good faith effort over several years to sell the facility and transfer the license to a non-foreign entity. This effort failed and Aerotest was forced to close its neutron radiography testing facility in the latter part of 2010 and plans to submit a request for a possession only license (POL) in 2011. The POL period will continue until all fuel is removed from site at which time the process for decommissioning and license termination will commence. The TRIGA is currently operated for short periods of time at low power levels in order to maintain operator qualifications and to satisfy the maintenance requirements in the Technical Specifications. There are no plans to resume reactor operations.

Aerotest plans to proceed with its decommissioning and the termination of the associated reactor license after fuel removal from site. After fuel removal Aerotest will file the appropriate decommissioning amendment requests together with this decommissioning plan with the NRC. As with other facilities of this nature, the ARRR Facility is contaminated with varying amounts of radioactive material and small amounts of hazardous material. The characterization study (Ref. 3-1) indicates that practices employed by Aerotest to minimize the spread of contamination were effective and contamination is relatively modest. Decontamination and Decommissioning (D&D) of the ARRR will eliminate the potential for future inadvertent environmental releases. The goal of the proposed D&D activities is termination of the ARRR TRIGA Reactor Nuclear Regulatory Commission (NRC) License R-80, Docket No. 50-228 and release of the reactor portions of the ARRR for "unrestricted use." The term "unrestricted use" means that there will be no future restrictions on the use of the site other than those imposed by the City of San Ramon zoning ordinances.

ARRR also maintains radioactive materials license number 2010-07 with the State of California. This license governs possession and use of several radioactive materials independent of the reactor. The State of California, as an Agreement State, has regulatory authority and responsibility for these radioactive materials. The state is the primary authority responsible for decommissioning the Site with respect to these materials. In addition the State of California is authorized to implement RCRA corrective action requirements, and through this means can implement more restrictive cleanup requirements for the ARRR decommissioning than required by the NRC. The State rules for termination of a radioactive materials license are provided in Title 17 California Code of Regulations, Division 1, Chapter 5, Subchapter 4, Section 30256.

Currently there is no dose based release criteria in California and there is a case by case evaluation of decommissioning plans performed by the California Department of Public Health (CDPH). Experience indicates that release limits that equate to a few mrem/yr are accepted.

The decommissioning is projected to start within a year, but not be completed until fuel is removed from the site which is currently projected to be sometime after 2055.

The decommissioning scenario to be implemented will depend on fuel storage. If on-site wet fuel storage is implemented, then the water circulation and cleanup systems would need to remain functional until the fuel is removed from site in 2055. If on site dry fuel storage is implemented, then all reactor operating systems could be removed. In both of these cases the site would still be licensed until after fuel removal when the final site decontamination, survey and license termination could take place. Remediation and removal of clean and contaminated materials, equipment and systems could proceed prior to fuel removal. It is anticipated that as much decommissioning as possible would take place prior to fuel removal in 2055, but the amount of early decommissioning performed will be at the discretion of Aerotest and regulators.

If off site fuel storage is a possibility, the decommissioning and license termination can proceed promptly.

The regional location of the ARRR is shown in Figure 3-1 and Figure 3-2 depicts the ARRR site and adjacent structures; Figure 3-3 shows an aerial image of the facility and Figure 3-4 presents a general arrangement plan of the ARRR Facility. This Decommissioning Plan has been prepared using the guidance and format of NUREG-1537 Rev. 0, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* (Ref. 3-1) and includes portions of the additional guidance from NUREG–1757, *Consolidated Decommissioning Guidance: Decommissioning Process for Materials Licensees* (Ref. 3-3). A summary profile for the ARRR TRIGA is provided in Table 3-1.

3.1.2 Decommissioning Plan Provisions

This Decommissioning Plan provides the following:

- A description of the present radiological condition of the ARRR Facility and site environs.
- A description of planned ARRR Facility radiological conditions and facility configuration during a SAFSTOR period.
- A description of the planned approach to be employed to decommission the ARRR Facility.
- Descriptions of the methods that will be utilized to ensure protection of the health and safety of the workers and to protect the environment and the public from radiological

hazards associated with ARRR Facility SAFSTOR and Decommissioning Project activities.

- A description of ARRR Facility physical security and material accountability controls that will be in place during the various phases of SAFSTOR and Decommissioning Project activities.
- A description of fuel management options and ultimate disposition.
- A description of radioactive waste management and disposal.
- A cost estimate for decommissioning the ARRR Facility and the source of funding for these activities.
- A schedule for the ARRR Facility Decommissioning Project.
- A description of the quality assurance program applicable to the ARRR Facility Decommissioning Project.
- A description of the training program to be established for personnel performing work in support of the ARRR Facility Decommissioning Project.
- An Environmental Report concerning the expected impact of performing the activities involved in the ARRR Facility Decommissioning Project.

3.2 BACKGROUND

Site and Facility History

The property, on which the ARRR is situated, was designated for construction in 1963. The ARRR was constructed between 1963 and 1964. The land area is well defined, as there is fence around the facility except for a well defined facility parking lot. The Reactor Building footprint is about 3,200 square feet and has two floor levels, and the total footprint for all buildings is 9,250 square feet. A layout of the ARRR buildings was provided in Figure 3-4, Figure 3-5 provides a plan view of the high bay reactor building and Figure 3-6 provides a cross section view of the Reactor Building.

The areas listed in this section of the ARRR decommissioning plan included all rooms in all buildings whether remediation could be required or not based upon the characterization study survey performed in May 2011.

Table 3-2 summarizes the results of the characterization survey and identifies areas containing residual activity. The characterization survey report should be reviewed in detail when planning for the scope of remediation in each area.

TRIGA Reactor

In 1963 Aerotest began construction of a facility to house the TRIGA Reactor, and supporting systems (e.g., Instrumentation and Control Systems, Forced Cooling System, Water Demineralization System, Ventilation/Exhaust System, Radiation Monitoring Systems, etc.). Following construction and reactor hardware installation, the TRIGA Reactor was brought to initial criticality in July of 1964. The TRIGA was routinely operational from that date until October 15, 2010. The TRIGA is currently operated only for short periods of time at low power levels in order to maintain operator qualifications. Aerotest plans to request that the USNRC issue an amendment to the TRIGA facility license to place the reactor in a Possession-Only-Status. The specific detailed conditions of this status are not known at this time.

Current Facility Status

It is anticipated that the TRIGA Reactor will be placed in "Possession-Only-Status" (POS), through an amendment to the USNRC License No. R-98, in 2011. The following conditions are anticipated for POS status:

- ARRR utility services required for facility operation and maintenance under POS status conditions will remain active.
- Manually actuated and automated fire alarm systems in the ARRR will remain operational.
- All building utility services required for facility operation and maintenance are active.
- The license-required radiological monitoring and instrumentation systems remain operational.
- Existing physical security and material control and accounting plans approved by the Nuclear Regulatory Commission (as may be amended) will continue to be implemented.
- The water demineralization system serving the ARRR is currently operational although the status may change depending on requirements that are implemented in the amended license.

3.2.1 Reactor Decommissioning Overview

Prior to implementing the decommissioning actions described herein, the ARRR will have been cleared of all extraneous fixtures, equipment and materials. The facilities will have been decontaminated to meet the site criteria for unrestricted access, radioactive waste will have been removed from site and secondary support systems outside the reactor building that are no longer needed will have been removed. The facility will be placed in SAFSTOR until the fuel is removed and then the facility will be decontaminated (deferred decontamination) to levels that permit release for unrestricted use. The majority of the remediation performed during the decommissioning will focus on components with the reactor tank, the shielding surrounding the reactor tank area, the primary cooling system and the demineralizer system. In other areas of the facility only minor remediation requirements are anticipated. The general activities to complete the Plan objectives are:

- Remove the neutron beam catcher, the wood shield over the tank and the concrete block shielding around the reactor tank area.

- Remove the components within the TRIGA reactor's tank.
- Remove the primary and secondary cooling systems.
- Perform additional decontamination and dismantlement of the structure and equipment in accordance with this plan.
- Remove the TRIGA reactor's tank if activated and any surrounding activated concrete.
- Demolish buildings that will not be left on site such as the Demineralizer Building, the Heat Exchange Building and the Chemical Shed.
- Prepare the waste generated by decommissioning activities for release or disposal (as appropriate). Either decontaminate and release the material as non-radioactive waste, or package it for disposal as radioactive waste.
- Ship all radioactive waste off-site to a licensed waste processor or disposal facility.
- Perform and document the final status survey(s) and submit a request to the USNRC for termination of the reactor license and submit a request to the state of California for termination of radioactive materials license number 2010-07.

3.2.2 Estimated Cost

The cost estimate is consistent with the scope of work covering D&D of the ARRR. D&D of the ARRR includes dismantlement of the reactor and reactor systems but retaining the reactor building and most other structures on site. The detailed estimated cost to decommission the ARRR is presented in the Decommissioning Cost Estimate for the Aerotest Radiography and Research Reactor, San Ramon CA (Ref. 3-4). This project is estimated to cost $3,285,800. A cost breakdown is given in

Table 3-3: Decommissioning Cost Summary – ARRR.

3.2.3 Availability of Funds

Estimates of the costs of decommissioning of ARRR USNRC licensed facility are provided in this plan. Autoliv, Inc is committed to providing the funding for decommissioning of the ARRR.

3.2.4 Program Quality Assurance

A Quality Assurance Project Plan (QAPP) will be developed to incorporate the applicable portions of 10 CFR 50, Appendix B and 10 CFR 71, Subpart H. In addition, the QAPP will utilize a graded approach that bases the level of controls on the intended use of the results and the degree of confidence needed in their quality. ANSI/ASQC E4-1994 (ASQC 1995) and Appendix K of MARSSIM will be used to provide guidance in quality systems, the collection and evaluation of environmental data, and for developing a QAPP.

A quality assurance program will be implemented throughout the ARRR decommissioning effort to assure that work does not endanger public safety and to assure the safety of the decommissioning staff.

Quality assurance efforts during the ARRR decommissioning period will include the following:

- Performing QA functions for procurement
- Qualifying suppliers
- Auditing project activities
- Monitoring worker performance for compliance with work procedures
- Verifying compliance of radioactive shipments with appropriate procedures and regulations
- Performing dimensional, visual, nondestructive examinations or other required inspection services to assure compliance with work plans
- Maintaining auditable files

The QAPP will be issued and approved by Aerotest and it will be documented by written procedures and implemented throughout the decommissioning project in accordance with those procedures. The management of those organizations participating in the QAPP shall regularly review the status and adequacy of that part of the plan that they are implementing.

All changes to the Plan shall be governed by measures commensurate with those applied to the original issue.

The Quality Assurance Project Plan will incorporate the items discussed in the following subsections.

3.2.4.1 Quality Assurance Responsibilities

The Quality Assurance organizations of Aerotest and the decommissioning contractor have the responsibility, authority and organizational freedom to:

- Identify quality problems
- Take action to stop unsatisfactory or unsafe work and control further processing, delivery, installation or use of nonconforming items
- Initiate, recommend or provide solutions
- Verify implementation of solutions.

Aerotest, as the reactor license holder, has the ultimate responsibility for the implementation of the QA Plan for the ARRR, both by Aerotest and its decommissioning subcontractor(s). Aerotest shall verify compliance through periodic audits.

3.2.4.2 Quality Requirements for Instrumentation

Calibration

Field instruments and associated detectors shall be calibrated on an annual basis using National Institute of Standards and Technology (NIST) traceable sources and appropriate calibration equipment and laboratory instruments shall be calibrated on an annual basis.

Calibration labels showing instrument identification number, calibration date and calibration due date shall be attached to all field and laboratory instrumentation.

Response Testing

All instrumentation will be inspected and source checked daily prior to use to verify calibration status and proper operation. Control checks and/or source check criteria will be established prior to the initial use of the instruments

Maintenance

Limited maintenance, such as changing Mylar windows, high voltage cables, etc., may be performed on-site by qualified personnel. Following the change of essential components for maintenance, limited calibration may be performed on site by qualified personnel.

Record Keeping

Calibration and maintenance records, or copies of these records, shall be maintained on site where they will be available for review. The results of the daily instrument functional checks will be recorded on separate log sheets for each instrument and maintained on-site.

3.2.4.3 Sampling and Analysis Quality Control

Sample Collection

Direct surface beta measurements, removable contamination measurements, gamma exposure rates, soil sampling and any specialized measurements will be performed to provide data required to meet the guidance provided in 10 CFR 20.1402, *Radiological Criteria for Unrestricted Use* (Ref. 3-5), NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (Ref. 3-6), and NUREG-1757, *Consolidated MSS Decommissioning Guidance, Decommissioning Process for Materials Licenses* (Ref. 3-3).

Sample QC

Quality Control (QC) samples will be obtained for minimum of 5% of all samples collected for radionuclide specific analysis. QC samples for direct measurements and smears are not required. The QC samples will be a combination of split, duplicate, blank, and/or spiked samples.

Sample Identification

Direct surface beta measurements, removable contamination samples, exposure rates, and any specialized measurements will be identified as to location, type of measurement, specific instrument and probe used, sample time and date (as appropriate) and name of the person collecting the data.

Soil samples will be identified with a unique sample number, sample location, depth of sample, sample time and date as appropriate, and the name of the person collecting the sample.

Sample Chain-of-Custody

Sample chain-of-custody shall be initiated for those samples being sent off site for analysis or transferred to another organization for analysis. A sample Chain-of-Custody Record will be generated which will document the sample identification and sample transfer and will accompany the sample during shipping to the new custodian of the sample.

Sample Analysis

Vendor laboratories shall be on a QA Approved Suppliers List for the decommissioning contractor or Aerotest for the type of analytical services being provided. Aerotest has the ultimate responsibility for ensuring that decommissioning sample analysis specifications and laboratory capabilities meet data quality requirements.

Sample Documentation

Sample identification information, sample Chain-of-Custody Records, sample analysis results, vendor laboratory qualification records, or copies of these records, shall be maintained on site where they will be available for review.

3.2.4.4 Record Keeping

Measures shall be established to control the issuance of documents and changes to documents that prescribe activities affecting quality, such as procedures, drawings and specifications. These measures shall ensure that documents and changes to documents are reviewed for adequacy, approved for release by authorized personnel and distributed to and implemented at the location where the prescribed activity is performed.

Procedure Control

Procedures shall be controlled to ensure that current copies are provided to personnel performing the prescribed activities. Procedures shall be independently reviewed by a qualified person and shall be approved by a management member of the organization responsible for the prescribed activity. Significant changes to procedures shall be reviewed and approved in the same manner as the original.

Radioactive Shipment Package Documents

All documents related to a specific shipping package for radioactive material shall be controlled by appropriate procedures. All significant changes to such documents shall be similarly controlled.

Final Survey Documents

All documents related to the final status survey shall be controlled by appropriate procedures. All significant changes to such documents shall be similarly controlled. This documentation would normally include a Survey Plan, Survey Packages, Survey Results and Survey Report.

3.2.4.5 Handling, Storage and Shipping

Approved procedures shall be utilized to control the handling, storage and shipping of radioactive materials.

Radioactive Material Storage

Areas shall be provided in the Reactor Complex for storage of radioactive material to ensure physical protection and control of the stored material. The handling, storage and shipment of radioactive material shall be controlled through the following requirements:

- Procedures shall be provided for handling, storage and shipping operations.
- Safety requirements established for the handling, storage and shipping of packages for radioactive material shall be followed.
- Shipments shall not be made unless all tests, certifications, acceptances and final inspections have been completed.

Shipping and Packaging

Shipping and packaging documents for radioactive material shall be consistent with pertinent regulatory requirements.

3.2.4.6 Quality Assurance Records

Sufficient records shall be maintained to furnish evidence of activities important to safe decommissioning as required by code, standard, and specification or project procedures. Records shall be identifiable, available and retrievable. The records shall be reviewed to ensure their completeness and ability to serve their intended function. Requirements shall be established concerning record collection, safekeeping, retention, maintenance, updating, location, storage, preservation, administration and assigned responsibility. Requirements shall be consistent with applicable regulations and the potential for impact on quality and radiation exposure to the workers and the public.

Typical records would include:

- Proposed Decommissioning Plan
- Procedures
- Reports
- Personnel qualification records
- Radiological and environmental site characterization records, including final site release records
- Dismantlement records
- Inspection, surveillance, audit and assessment records

Health and Safety Related Activities

Records that have a potential for impact on quality and radiation exposure to the workers and the public include the following:

- Work Permits
- Work Procedures
- Contamination Survey Reports
- Radiation Survey Reports
- Airborne Survey Reports
- Counting data or air samples and gamma spectrum analysis

- Instrument calibrations
- Source inventory and storage
- Radioactive material inventory and storage
- Shipment records
- Incidents and accidents
- Confined space entry permits
- Monitoring records for oxygen deficient and explosive atmospheres

Personal Records

Typical records containing personal information that may impact quality and radiation exposure to the workers and the public are as follows:

- Dosimetry Records
- Bioassay analysis
- Respiratory protection qualifications (medical/clearance and fit test)
- Training records
- Visitor logs and exposure information

3.2.4.7 Audits

Audits shall be implemented to verify compliance with appropriate requirements of the Quality Assurance Project Plan and to determine the effectiveness of the plan. The audits shall be performed in accordance with written procedures or checklists by trained and qualified personnel not having direct responsibility in the areas being audited.

Audit Reports

Reports of the results of each audit shall be prepared. These reports shall include a description of the area audited, identification of the individual responsible for implementation of the audited provisions and for performance of the audit, and identification of discrepant areas. The audit report shall be distributed to the appropriate level of management and to those individuals responsible for implementation of audited provisions.

Audit Corrective Action

Measures shall be established to ensure that discrepancies identified by audits are resolved. These measures shall include notification of the manager responsible for the discrepancy and verification of satisfactory resolution. Discrepancies shall be resolved by the manager responsible for the discrepancy. Higher levels of management shall resolve disputed discrepancies.

Follow-up action, including re-audit of deficient areas, shall be taken as indicated.

Figure 3-1: Aerotest Operations Site Location

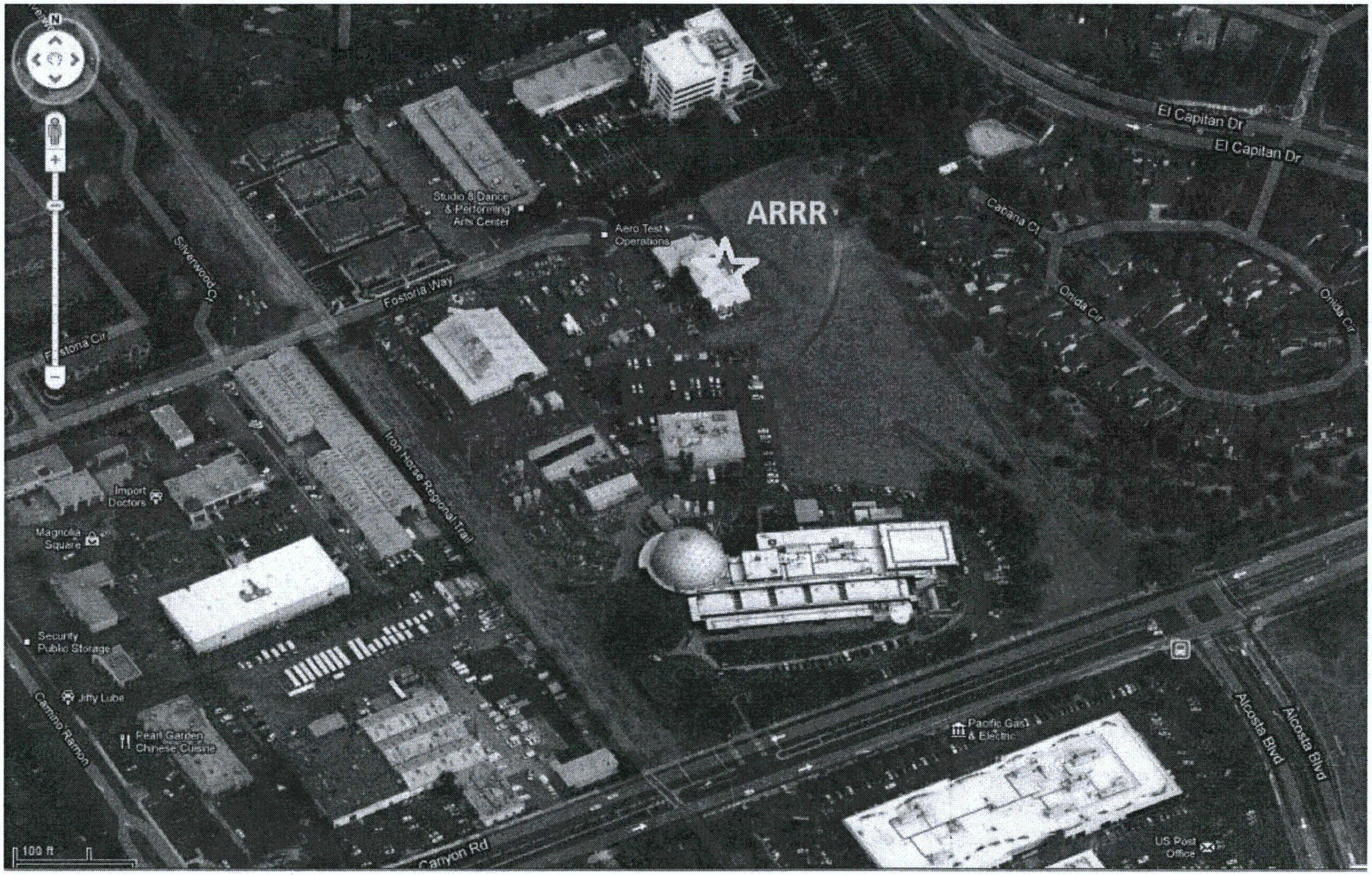

Figure 3-2: ARRR Local Area View

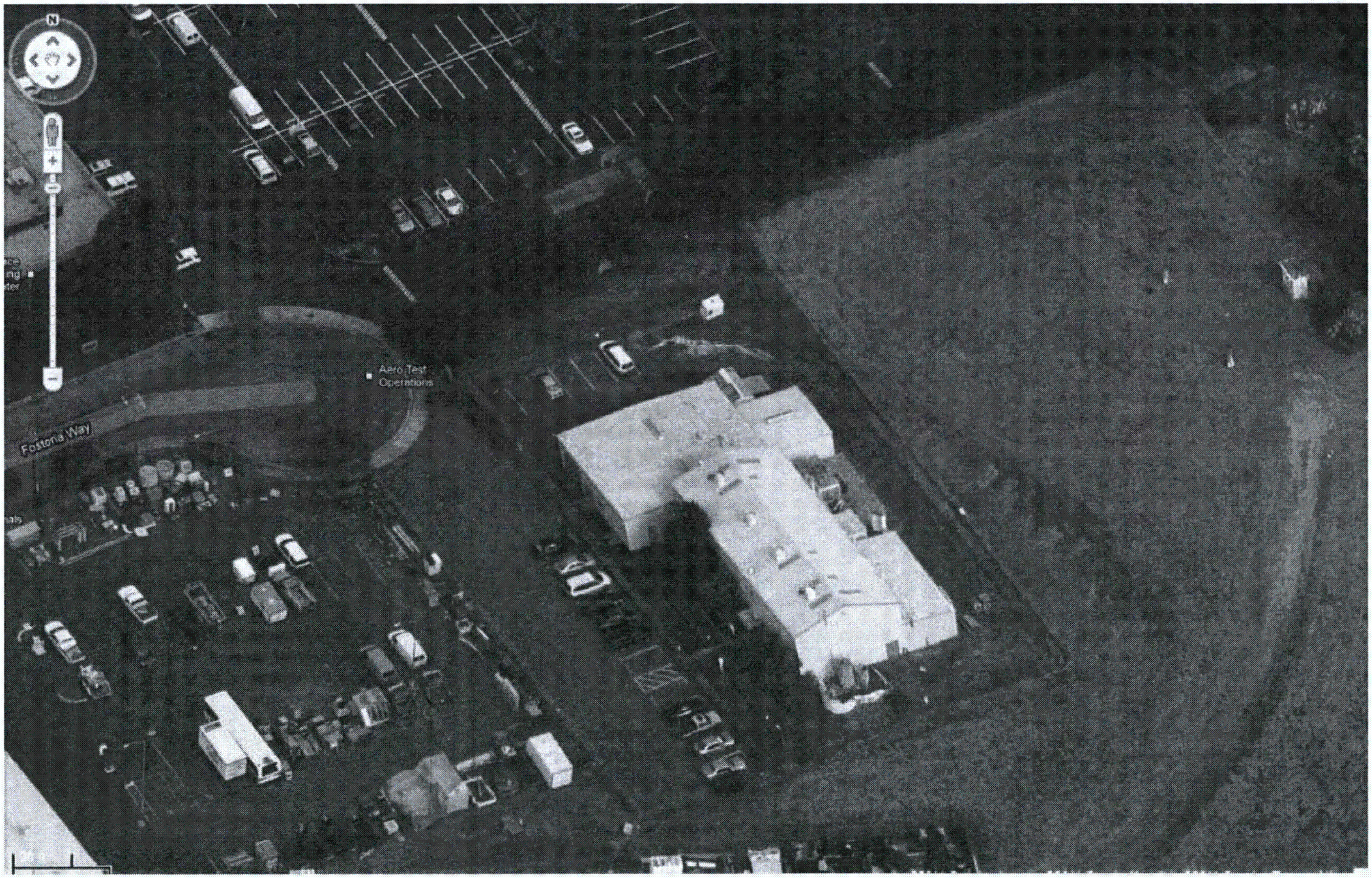

Figure 3-3: Aerotest Aerial Image

1. Office Space
2. Customer Viewing Room
3. Quality Control Room
4. Dark Room
5. Explosive Storage & Safe
6. Film Storage
7. Shipping & Receiving
8. N-Ray Setup Area
9. Computer and Counting Room
10. Lunch Room
11. High Bay N-Ray Exposure Area
12. Reactor Enclosure
13. Control Room
14. Men's Room
15. Ladies' Room
16. Employee's Lockers
17. General Manager's Office
18. Business Office
19. Accounting Office
20. Machine Shop
21. Office Supply Room
22. Tagging Area
23. South End Radiography
24. Demineralizer Building
25. Maintenance Office
26. Heat Exchange Building
27. Backup Cooling Tower
28. Compressor Building
29. Safe
30. Waste Storage Tanks & Sump
31. N-Ray Gauge Office
32. Preparation Lab
33. Chemical Lab
34. Storage
35. Instrument Calibration Area
36. Sheet Metal Fabrication Area
37. Electronics Lab
38. Class 1.1 Explosive Storage
39. Main Cooling Tower
42. Storage Building
43. Parking Area
44. Parking Area
45. Radioactive Material Storage Room
46. Chemical Shed

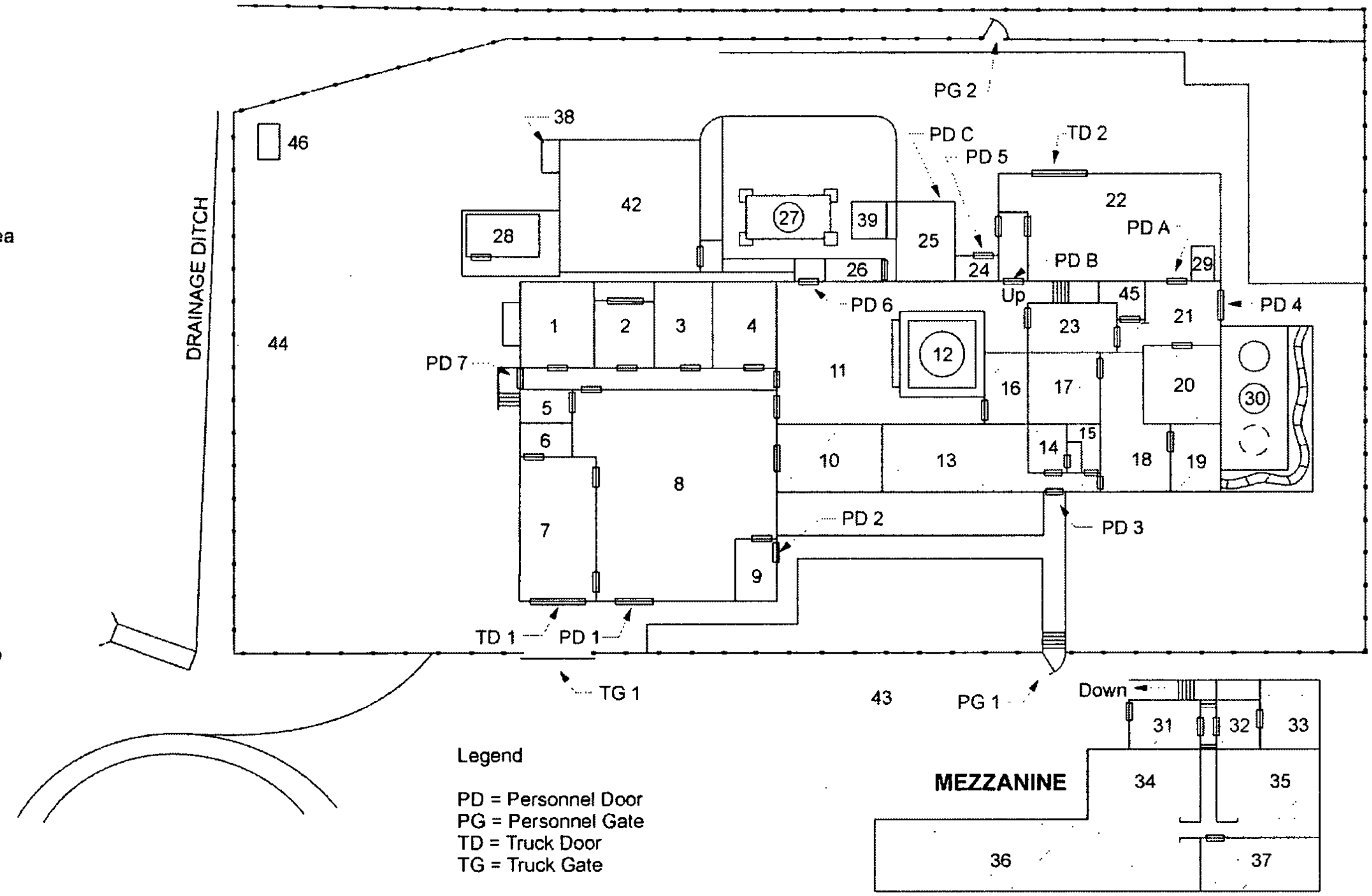

Figure 3-4 General ARRR Arrangement Plan

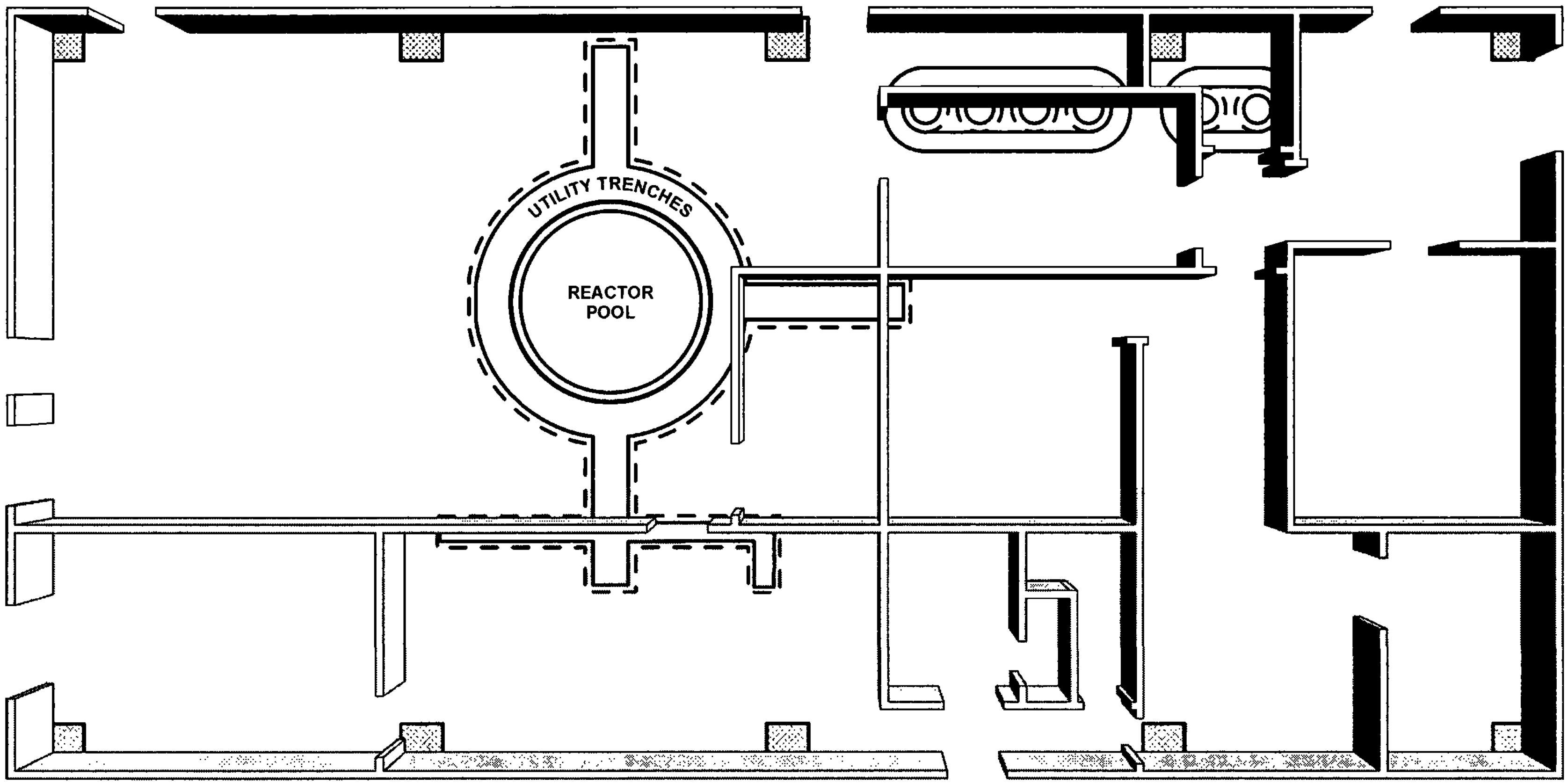

Figure 3-5 Plan Layout of ARRR High Bay Reactor Building

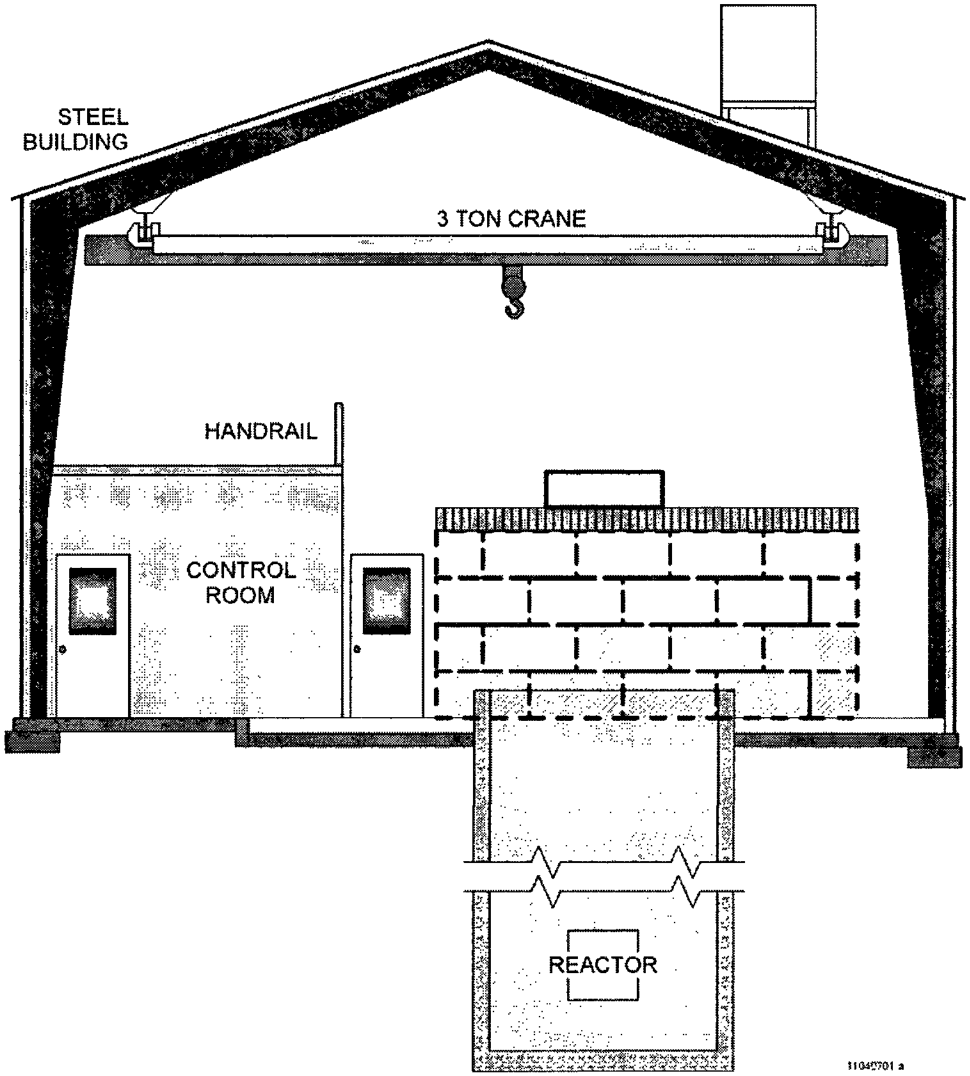

Figure 3-6: Cross Section of ARRR High Bay Reactor Building

Table 3-1: Profile of the ARRR TRIGA Reactor

General Reactor information:	
Type:	TRIGA CONV
Owner/Operator:	OEA, INC./AEROTEST OPERATIONS INC
Licensee:	Aerotest Operations, Inc.
Nuclear Design:	General Atomic Division of General Dynamics
Principal Uses:	Radiographic Irradiations
Reactor Operation and Authorization:	
Initial Criticality:	July 9, 1964
USNRC Utilization Facility License #:	R-98
USNRC Facility Docket #:	50-228
Reactor Specifications:	
Thermal Power, Steady (kW):	250
Maximum Flux SS, Thermal (n/cm^2-s):	3.0×10^{13}
Equilibrium Core Size (Fuel Elements)	84
Coolant:	Light Water
Moderator:	H_2O, ZrH
Reflector:	Graphite
Control Rod Material	B_4C
Control Rods Number	3

Table 3-2: Characterization Survey Summary

Survey Package Number	Location Description	Survey Results
1	Reactor Building Inside Bioshield	Residual Activity Detected
2	Reactor Building Rad Material Storage Area	Residual Activity Detected
3	Demineralizer Building	Residual Activity Detected
	Maintenance Office	Residual Activity Not Detected
	Heat Exchanger Building	Residual Activity Detected
4	Waste Storage Tank Area	Residual Activity Detected
5	Building Addition 1 Counting Room	Residual Activity Detected
	Reactor Building Conference/Lunch Room	Residual Activity Not Detected
	Reactor Building Control Room	Residual Activity Not Detected
	Reactor Building Employee's Lockers	Residual Activity Not Detected
	Reactor Building General Manager's Office	Residual Activity Not Detected
	Reactor Building Machine Shop	Residual Activity Not Detected
	Reactor Building Office Supply Room	Residual Activity Not Detected
	Reactor Building South End Radiography	Residual Activity Not Detected
6	Tagging Building Safe	Residual Activity Not Detected
7	Reactor Mezzanine Preparation lab	Residual Activity Not Detected
	Reactor Mezzanine Chemical lab	Residual Activity Not Detected
	Reactor Mezzanine Instrument Calibration Area	Residual Activity Not Detected
	Reactor Mezzanine Electronics Lab	Residual Activity Not Detected

Survey Package Number	Location Description	Survey Results
	Reactor Mezzanine Stairway	Residual Activity Not Detected
	Reactor Mezzanine N-Ray Gauge office	Residual Activity Not Detected
	Reactor Mezzanine Sheet Metal Fab Area	Residual Activity Not Detected
	Reactor Mezzanine Storage area	Residual Activity Not Detected
8	Building Addition 1 Office Space	Residual Activity Not Detected
	Building Addition 1 Customer Viewing Area	Residual Activity Not Detected
	Building Addition 1 Quality Control Room	Residual Activity Not Detected
	Building Addition 1 Dark Room	Residual Activity Not Detected
	Building Addition 1 Hallway	Residual Activity Not Detected
	Building Addition 1 Explosive Storage Safe	Residual Activity Not Detected
	Building Addition 1 Film Storage Room	Residual Activity Not Detected
	Building Addition 1 Shipping & Receiving	Residual Activity Not Detected
	Building Addition 1 N-Ray Setup Area	Residual Activity Not Detected
9	Reactor Building Men's Room	Residual Activity Not Detected
	Reactor Building Ladies' Room	Residual Activity Not Detected
	Reactor Building Business Office	Residual Activity Not Detected
	Reactor Building Accounting Office	Residual Activity Not Detected
10	Tagging Building Entry Vestibule	Residual Activity Not Detected
	Tagging Area	Residual Activity Not Detected
	Tagging Area Back Room	Residual Activity Not Detected
	Storage Building	Residual Activity Not Detected
	Compressor Building	Residual Activity Not Detected
	Chemical Shed	Residual Activity Not Detected
11	Reactor Building Exterior Walls	Residual Activity Not Detected
	Building Addition 1 Exterior Walls	Residual Activity Not Detected
	Tagging Building Exterior Walls	Residual Activity Not Detected
	Storage Building Exterior Walls	Residual Activity Not Detected
12	All Other Buildings Exterior Walls	Residual Activity Not Detected
13	Parking Area Outside Fence	Residual Activity Not Detected
14	Paved Areas Inside Fence	Residual Activity Not Detected
15	Soil Areas Inside Fence	Residual Activity Not Detected
16	Main Cooling Tower	Residual Activity Not Detected
	Backup Cooling Tower	Residual Activity Not Detected
17	Waste Storage Tanks	Residual Activity Detected

Table 3-3: Decommissioning Cost Summary – ARRR

Operation	Man-hours	Labor Plus Trav. & Liv.	Equipment, Contracts & Supplies	Radwaste Shipping & Disposal	Total Cost
TRIGA Reactor	6,147	$589,449	$97,714	$309,432	$996,596
TRIGA shielding & N-Ray Components	76	$	$26,816	$71,639	$98,455
Buildings	2,727	$260,233	$64,981	$295,003	$620,217
Outdoor Areas	530	$48,757	$43,935	$342,268	$434,959
Decommissioning Planning	240	$22,800	$0	$0	$22,800
Characterization Surveys	455	$47,849	$2,335	$0	$50,183
Final Surveys	1,820	$191,395	$9,338	$0	$200,733
Planning, Training, & Mobilization	404	$36,244	$0	$0	$36,244
Owner Oversight & Licensing	1,527	$138,453	$0	$0	$138,453
NRC Verification Survey		$0	$0	$0	$30,000
Totals	**13,926**	**$1,335,180**	**$245,118**	**$1,018,342**	**$2,628,640**
			25% CONTINGENCY		**$657,160**
				GRAND TOTAL	**$3,285,800**

* The estimate for LLW disposal is based upon the assumption that all radioactive waste will be buried at the Clive Utah disposal site.

REFERENCES FOR SECTION 3

3-1 CS-HP-PR-007, "Characterization Report for the Aerotest Radiography & Research Reactor, San Ramon, California", Revision 0, August 2011.

3-2 NUREG- 1537 Rev. 0, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors".

3-3 NUREG–1757, "Consolidated MSS Decommissioning Guidance, Decommissioning Process for Materials Licenses", September 2002.

3-4 CS-HP-PR-006, "Decommissioning Cost Estimate for the Aerotest Radiography and Research Reactor, San Ramon", Revision 0, July 2011, prepared by Energy*Solutions*.

3-5 10 CFR 20.1402 Radiological Criteria for Unrestricted Use.

3-6 NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM).

4.0 DECOMMISSIONING ACTIVITIES

The decommissioning alternative selected for the ARRR is the removal of the facility from service, establishment of "Possession-Only-Status" (POS), reduction of the residual radioactivity to levels that allow unrestricted access to all facility areas, entering a SAFSTOR period followed by fuel removal and final decontamination that will permit termination of the reactor licenses and beneficial reuse of the property. The facility will be surveyed and left in place.

4.1 DECOMMISSIONING ACTIVITIES

The objective of ARRR Decommissioning is the regulatory release of the TRIGA reactor to unrestricted use. On this basis the entombment (ENTOMB) decommissioning options were considered inappropriate to Aerotest's future plans.

The Proposed Action and the Alternatives are as follows:

Proposed Action (Modified SAFSTOR) - In safe storage, the Aerotest Reactor would be placed and maintained in a condition that allows it to be safely stored and subsequently decontaminated to USNRC and state of California levels permitting release of the property. This would involve retention of the fuel onsite until the Department of Energy (DOE) is able to take the spent fuel. The DOE has agreed to take the fuel in 2055 at the earliest.

Alternative 1 (DECON) - Decontamination and Decommissioning of the ARRR, including the reactor, followed by license termination and subsequent release of the site for unrestricted use. This is not currently a viable option as there is not a possibility of fuel removal from the site in the near future.

Alternative 2 (ENTOMB) - In entombment, radioactive materials are encased in a structurally long lived material such as concrete. The entombed structure is appropriately maintained and surveillance is continued until the radioactivity decays to a USNRC or state of California level permitting release of the property. This is not currently a viable option as the fuel must first be removed from the site and there is not a possibility of fuel removal from the site in the near future.

No Action Alternative - A no-action alternative would leave the facility in its current status with the current support staff having to maintain the facility under the existing license conditions. This action would not be allowed without a license transfer to a new facility owner. This action would involve maintaining:

- The facility reactor operating license
- Personnel to support facility maintenance and surveillance
- Surveillance and maintenance of Reactor Pool Water Level, Purity and pH
- The Reactor Facility physical security plan

The reactor tank still contains fuel and activated hardware with some items reading over an estimated 500 R/hr on contact. The reactor tank does not have a history of leakage. However keeping the facility in this status over a long period of time may lead to a degradation of the tank. That degradation will require either the repair or the decommissioning of that portion of the facility. Aerotest would incur expenses for maintenance of the facility without making beneficial use of the facility.

The NRC requirement in 10 CFR 50.82(b)(1)(ii) providing for non-power reactor decommissioning without significant delay following permanent shutdown would have to be waived.

Implementation of the Proposed Action would include retention of the fuel on site until the DOE is able to take the spent fuel. This could probably be accomplished using one of two scenarios, either (1) storing the fuel in the pool where it is currently located or (2) removal of the reactor fuel from the tank to an on-site dry storage location.

The scenario where the fuel is maintained in the pool would likely include the following tasks:

- Move some or all of the fuel out of the core to storage racks on the pool floor or on the pool wall.
- Continued operation and maintenance of the pool water demineralizer system.
- Optional removal of the pool water cooling system including heat exchanger and cooling towers.
- Decontamination of any contaminated areas outside of the reactor pool.
- Shipment of the low level radioactive waste (LLRW) currently on site or generated as a result of decommissioning activities.
- Performance of surveys to confirm the facility status and submission of a request to the USNRC for a Possession Only Status (POS) through an amendment to the USNRC License No. R-98, in 2011.
- Daily site monitoring of operations, similar to current site requirements, to ensure systems are performing correctly, and performing maintenance of the facility for continued occupancy.
- Existing physical security, radiological control, material control and accounting plans approved by the Nuclear Regulatory Commission (as may be amended) will continue to be implemented.
- Once the DOE has taken the fuel off site, the facility would be decontaminated and decommissioned, including the performance of Final Status Surveys and release of the subject areas for unrestricted use and termination of the ARRR license.

The scenario where the fuel is removed from the pool to dry storage on site would likely include the following tasks:

- Removal of the reactor fuel from the tank to an on-site dry storage location. This may include utilizing the existing dry storage locations, new storage locations or a new dry storage cask.
- Dismantlement, decontamination or packaging as low-level radioactive waste (LLRW) the ARRR Reactor components including the demineralizer system, the cooling system, the reactor area shielding and the operating and control systems, but not the tank.
- Decontamination of any contaminated areas.
- Shipment of the low level radioactive waste (LLRW) currently on site or generated as a result of decommissioning activities.
- Performance of surveys to confirm the facility status and submission of a request to the USNRC for a Possession Only Status (POS), through an amendment to the USNRC License No. R-98, in 2011.
- Existing physical security, radiological control, material control and accounting plans approved by the Nuclear Regulatory Commission (as may be amended) will continue to be implemented.
- Once the DOE has taken the fuel off site, the facility would be decontaminated and decommissioned, including the performance of Final Status Surveys and release of the subject areas for unrestricted use and termination of the ARRR license.

4.2 FACILITY RADIOLOGICAL STATUS

4.2.1 Facility Operating History

The initial fuel loading for the AGNIR utilized previously used aluminum clad fuel elements provided by GA. The reactor achieved initial criticality on July 9, 1965 with a licensed steady-state thermal power limit of 250 kW.

The ARRR has provided a neutron source for research and development and services, mainly neutron radiology. Irradiation services for activation analyses have included: crude oil and hydrocarbon samples for oil companies; plastic slides impregnated with microscopic quantities of fissionable materials; ocean silt samples for the Bureau of Mines; and, silver iodide in snow samples from cloud seeding. Other irradiation services have included: calibration of power reactor fission detectors; radiation damage effects studies of solid state electronic components; detection of gunshot residue in paraffin; lattice deformation studies in ammonium perchlorate; and, spallation experiments with uranium dioxide. The reactor has accumulated a total of 14,107,874 kw hours after 69,582 hours of operation (Ref. 4-1).

Reactor operations ceased on October 15, 2010.

There were no instances of releases of radioactive material (activity levels greater than public release limits for air or water to sewers). There was however, one instance of a leaking fuel element reported during the operation of the ARRR. The reactor achieved initial criticality July 9, 1965 and the ruptured fuel element was located and removed from the core in November of 1965. The fuel element was immediately shipped offsite for disposal (Ref. 4-1). The current presence of fission products such as ^{137}Cs in the pool water demin system indicates there may be other leaking fuel elements.

There were no instances of surface water flooding that may have contributed to an uncontrolled release of radioactive water. There were however several instances of work area flooding from internal water sources as described below (Ref. 4-1):

> In 2002 a pipe burst and flooded the N-Ray set-up room and garage. This was a clean area and it involved normal water cleanup and disposal methods. No radioactive material was involved.
>
> There was a sewer backup that got into the trench system but there was never any overflow to the floors. The backup was resolved when the drain line was cleaned out by the city. No radioactive material was involved.
>
> There were pipe breaks several times in the Conference/Lunch Room area and also in the QC wing area and in the High Bay area. No radioactive material was involved.
>
> The sink in the Chemistry Room has overflowed several times where water ran down the stairs before someone realized that the water had been left on in the sink and shut off the water. No radioactive material was involved.

A capsule of Na-24 was spilled at the south end of the facility before carpet was laid in the area. The spill was cleaned up immediately after it occurred.

Building gaseous effluents were continuously monitored by sampling air from the area above the reactor water tank. The air intake was either just above the reactor pool, which was the normal position, or in the ceiling of the reactor room. These locations ensured that if gaseous fission products were released they would be detected. No gaseous fission products were detected (Ref. 4-1).

In addition a building particulate sampler is used to continuously withdraw an air sample from the reactor room and collect particulate material on a filter paper. The reactor room particulate samples are analyzed monthly and no activity in excess of limits was ever noted (Ref. 4-1).

4.2.2 Current Radiological Status of the Facility

4.2.2.1 General

Routine radiological surveys show that the radiation levels and contamination levels measured at the ARRR have consistently been low. A characterization survey performed in May 2011, and summarized in Appendix A, confirmed that only minor quantities of residual radioactivity or radioactive contamination are present. The information indicates that the radioactive portions of the facility are primarily confined to the radioactive waste room and the reactor pool.

Estimates of the radioactivity inventory can be determined by considering the constituent elements of the material in question and calculating the duration of exposure to the neutron flux and the energies of the incident neutrons. Direct measurements, however, are generally more reliable and will be used during actual removal and/or dismantlement of components. These direct measurements will further define the basis for specifying the necessary safety measures and procedures for the various dismantling, removal, decontamination, and waste packaging and storage operations so that exposure to personnel is maintained ALARA.

4.2.2.2 Radionuclides

The radionuclides known to be present, or possibly present in detectable levels within the ARRR, are listed in Table 4-1. The list of expected radionuclides is based on the assumption that reactor operation resulted in neutron activation of reactor core components and other integral hardware or structural members situated adjacent to, or in close proximity to, the reactor core. Specific items to be considered exposed to neutron activation include materials composed of aluminum, steel, stainless steel, graphite, cadmium, lead, concrete and possibly others. The determination of residual activity in reactor components and structures surrounding the reactor will be based upon the theoretical neutron activation calculations performed by a subcontractor at the Georgia Institute of Technology (Georgia Tech) (Ref. 4-2) and verified by direct measurements and sampling.

4.3 DECOMMISSIONING TASKS

4.3.1 Activities and Tasks

4.3.1.1 Preparation of the ARRR for Decommissioning

4.3.1.1.1 Characterization Surveys

As part of Decommissioning Project planning actions, studies have been conducted to determine the type, quantity, condition and location of radioactive materials, which are, or may be, present in the ARRR and surrounding areas. A contractor, Energy*Solutions*, performed a comprehensive characterization survey of the ARRR in May 2011. A summary of these surveys are provided in this document as Appendix A: *Summary of Characterization Results*. Detailed survey results are presented in the Characterization Report (Ref. 4-2).

4.3.1.1.2 General Cleanup of ARRR

In preparation for decommissioning activities, non-reactor related equipment and materials situated throughout the Reactor Complex will be collected, surveyed, packaged and appropriately dispositioned in accordance with established procedures.

4.3.1.1.3 Decontamination of the Facility

This Decommissioning Plan pertains to the dismantling of the reactor and associated systems in a safe manner and in accordance with ALARA principles, and the decontamination and survey of the entire ARRR.

Disposition of Decommissioning Equipment and Materials

The equipment, materials, instrumentation, and tools that are used or encountered during the decommissioning will be handled as described below:

- The above items may be surveyed and released on site as clean waste if the residual radioactivity is less than the current site release limits of 200 dpm/100 cm^2 beta above background and 30 dpm/100 cm^2 alpha above background.
- The above items may be shipped directly for disposal as radioactive waste.
- The above items may be shipped to a licensed radioactive material processing facility for survey and release, decontamination followed by survey and release, or shipment for disposal as radioactive waste.
- The above items may be shipped to a licensed facility for holding until they are utilized on another project involving radioactive materials.
- No contaminated items as listed above will be left on site.

Reactor High Bay Building

- The equipment, materials, instrumentation, and tools that are used during the decommissioning will be handled as described in the previous section.
- All contaminated equipment will be decontaminated, surveyed and left in place or removed for disposal or processing and disposal.
- High Bay Building HVAC system filters will be removed and the remaining system will be surveyed and left in place.
- Concrete floors will be decontaminated if required by removing a portion of the upper concrete surface. Pipes and drains will be surveyed and left in place or decontaminated as required, or removed for disposal or processing and disposal.
- Building roof exhausts will be surveyed and left in place.
- The ARRR Crane will be utilized during the decommissioning activities. It will be surveyed, decontaminated if required and left in place and in operating condition.

Reactor and Pool

- Remove beam catcher and wood shield over reactor tank, characterize them and ship them for disposal.
- Remove concrete blocks around reactor tank area, decontaminate them and survey them for disposal as clean waste.

- Remove the reactor core, core stand, thermal column, fuel racks, vertical beam tube, and other items inside the reactor pool and ship them for disposal as LLRW.
- Remove reactor tools, characterize them and ship them for disposal.
- The reactor bridge and items attached to the bridge will be removed and shipped for disposal.
- When no longer useful as a radiological shield, the reactor pool water will be processed through a demineralizer system, sampled and discharged as non-radioactive waste using the existing site discharge procedures.
- Any debris or hardware remaining present in the pool will be removed and shipped for disposal.
- An activation analysis performed by Georgia Tech and reported in Appendix B of the "Characterization Survey Report" (Ref. 4-2) indicates that near the core there will be activated concrete and activated rebar in the concrete. Radionuclides currently in excess of the anticipated California limits, Table 4-7, include C-14, Ca-41, Ca-45, Mn-54, Fe-55 and Co-60. However, if there is a lengthy storage period prior to decommissioning (~50 years), the only remaining radionuclide of concern will be C-14. The C-14 will be less than NRC limits but close to anticipated California limits for soil and a good case could be made to leave it in place as it is encapsulated in concrete deeply below grade. In this case the tank and concrete would be removed down to 6-feet below grade and the rest left in place as the concrete is not contaminated. In the case of prompt decommissioning, about one foot of concrete near the core will need to be removed. The prompt decommissioning activated concrete to be removed is shown in Figure 4-1.
- The activation analysis also indicated slightly elevated activity in the soil; Ca-45 and Mn-54 exceed the anticipated California limits but not the NRC limits. If the tank and concrete is being removed during a prompt decommissioning scenario, then up to a foot of soil near the core will be removed. In the case of delayed decommissioning (~50 years), the Ca-45 and Mn-54 will have decayed to less than the anticipated California limits.
- In the case of delayed decommissioning (~50 years), the reactor pool tank will be decontaminated and the tank left in place. In this case the tank and concrete would be removed down to 6-feet below grade and the rest left in place as the tank and concrete are not contaminated. The Co-60 in the tank walls will have decayed away and the activity in the concrete behind the aluminum tank will have decayed away also as indicated above. In the case of prompt decommissioning, the pool liner will be removed for disposal as radioactive waste.
- The unused beam port is not activated and will be left in place.
- The reactor pool trenches will be decontaminated and the concrete left in place.

- The remaining tasks are removal of residual surface contamination in the rooms, and performance of the final status survey. The packaged waste is to be shipped to a licensed processing or disposal facility.

Remaining Rooms and Structure

The equipment, materials, instrumentation, and tools that are used or encountered during decommissioning of the remaining rooms and structures will be handled as the same as indicated above for areas known to be potentially contaminated.

4.3.1.2 Dismantling Sequence

Dismantling will occur sequentially by the schedule shown in Figure 4-2. Items to be removed are grouped as follows:

Group 1 Equipment that does not have induced radioactivity but which may have surface contamination.

Group 2 Core components and other components that have induced radioactivity, including pool concrete that has been activated.

Group 3 Reactor support systems.

Group 4 Equipment, tools and systems that become contaminated during decommissioning operations.

Components and equipment in the four groups are identified in

Table 4-2, Table 4-3, Table 4-4 and Table 4-5.

4.3.1.2.1 Reactor Area

- The Beam Catcher and wood shield over the reactor tank will be removed initially followed by removal of the N-Ray exposure tray system and a portion of the shielding blocks surrounding the pool top. This will allow good access to the reactor bridge and pool contents while still providing personnel shielding for removal of the activated pool contents. The Beam Catcher and wood shield will be disposed of as radioactive waste.

- The control rods in the TRIGA pool are expected to have the highest dose rates from induced radioactivity. The control rods and other Group 2 items will be hoisted from the pool within shielded containers that will have been prepared to accept the items.

- After pool components, equipment and parts listed in Table 2-3 and Table 2 4 have been removed; the remaining shielding around the pool will be removed. This may involve an initial surface decontamination prior to removal of the shielding depending on the surface activity levels found. If the piping in trenches that passes from the pool to the heat exchanger and the demineralizer systems is still in place it will be removed at this time.

- The control panels in the Control Room will be removed and the associated wiring in the trenches removed. The Trenches will be decontaminated by a combination of concrete surface removal (scabbling) and manual wipe down with cleaning agents.

- The reactor tools will be removed from the storage cabinets, characterized and shipped for disposal. The reactor bridge and the items attached to the bridge will be removed and shipped for disposal.

- The reactor pool will be emptied and the water processed through a demineralizer system, sampled and discharged as non-radioactive waste using the existing site discharge procedures.

- An activation analysis performed by Georgia Tech and reported in Appendix B of the "Characterization Survey Report" (Ref. 4-2) indicates that near the core there will be activated concrete and activated rebar in the concrete. Radionuclides currently in excess of the anticipated California limits, Table 4-7, include C-14, Ca-41, Ca-45, Mn-54, Fe-55 and Co-60. However, if there is a lengthy storage period prior to decommissioning (~50 years), the only remaining radionuclide of concern will be C-14. The C-14 will be less than NRC limits but close to anticipated California limits for soil and a good case could be made to leave it in place as it is encapsulated in concrete deeply below grade. In this case the tank and concrete would be removed down to 6-feet below grade and the rest left in place as the concrete is not contaminated. In the case of prompt decommissioning, about one foot of concrete near the core will need to be removed. The prompt decommissioning activated concrete to be removed is shown in Figure 4-1.

- The activation analysis also indicated slightly elevated activity in the soil; Ca-45 and Mn-54 exceed the anticipated California limits but not the NRC limits. If the tank and concrete is being removed during a prompt decommissioning scenario, then up to a foot of soil near the core will be removed. In the case of delayed decommissioning (~50 years), the Ca-45 and Mn-54 will have decayed to less than the anticipated California limits.

- In the case of delayed decommissioning (~50 years), the reactor pool tank will be decontaminated and the tank left in place. In this case the tank and concrete would be removed down to 6-feet below grade and the rest left in place as the tank and concrete are not contaminated. The Co-60 in the tank walls will have decayed away and the activity in the concrete behind the aluminum tank will have decayed away also as indicated above. In the case of prompt decommissioning, the pool liner will be removed for disposal as radioactive waste.

- The unused beam port is not activated and will be left in place.

- There are two potential radiological safety concerns during performance of the above tasks: 1) external exposure from the activated components in the pool, and 2) inhalation of airborne material. To minimize the risk, during occupancy, the work areas will be monitored frequently and radiation levels will be monitored continuously, to determine sudden changes in the radiological conditions.

- Post-remediation surveys of the remaining building floor concrete may include concrete coring sampling and analysis. As the removal of activated material proceeds, the radioactive material will be packaged for shipment and disposal.

- If the water treatment and demineralizer systems are still in place, they will be removed at this time. The water treatment system includes the filter and demineralizer along with associated pipe, valves and instrumentation. The pool water cooling system includes a heat exchanger that utilizes chilled water from a cooling tower, pumps and associated piping valves and controls. The demineralizer system is housed in the Demineralizer Building and the primary side of the cooling system is housed in the Heat Exchanger Building. The secondary side of the cooling system is outdoors next to the Cooling Tower. The lead shielding in the Demin Building will be sent offsite for processing and disposal. The spare demineralizer in the building will be characterized and shipped offsite as radioactive waste. The concrete pads in the Demin and Heat Exchange Buildings are contaminated and the pads will either be decontaminated by concrete surface removal and the remaining pads disposed of as clean waste or the pads disposed of as radioactive waste along with the rest of the building.

4.3.1.2.2 <u>Reactor High Bay Area</u>

All equipment in the Reactor High Bay Building that is not part of the structure or utilities will be removed for disposal and contaminated building surfaces will be decontaminated. It is anticipated that this activity will include the following items:

- Radwaste Room: The room waste will be characterized and sent offsite for disposal as radioactive waste. The lead will be sent offsite for processing and disposal. The asbestos floor tile will be removed and sent for disposal as radioactive waste. The floor will be decontaminated by surface concrete removal and 10% of the lower walls will be decontaminated by surface concrete removal. About 25% of the walls will be decontaminated by a hand wipe down.

- Machine Shop: Any remaining small tools, supplies cabinets and tables will be surveyed and removed from site. The large equipment including Drill Press, Milling Machine and Lathe will also be surveyed, decontaminated if necessary and removed from site. The asbestos floor tile will be surveyed, removed and sent for disposal as asbestos waste or radioactive waste.

- Office Areas, Locker Room and Supply Room: Any remaining furniture, supplies cabinets and tables will be surveyed and removed from site. Any asbestos floor tile will be surveyed, removed and sent for disposal as asbestos waste or radioactive waste.

- Restrooms: In general these rooms do not require any remediation. Any items and materials in the supply closet in the men's restroom will be surveyed and removed from site.

- Control Room and Lunch Room: The control room cabinets will be surveyed and removed from site. Any remaining furniture, monitors, supplies, sink, cabinets and tables will be surveyed and removed from site. The asbestos floor tile under the sink area will be surveyed, removed and sent for disposal as asbestos waste or radioactive waste.

- Main High Bay Area: An area of about 40-ft by 40-ft around the reactor pit will be decontaminated by surface concrete removal. The HVAC ductwork and heating and cooling units will be surveyed and any contaminated items removed for disposal as radioactive waste. The roof insulation material will be surveyed and removed for disposal if any contamination is found. In addition the overhead lights will be surveyed and removed for disposal if any contamination is found. The florescent light bulbs and ballasts will be removed for separate disposal as they contain hazardous materials. Mercury is in the bulbs and there are probably PCB's in the ballasts.

4.3.1.2.3 <u>High Bay Mezzanine Area</u>

- N-Ray Gauge Office, Preparation Lab, Hallway and Chemical Lab: In the Chemical lab the hood and support bench along with the lab benches and sinks will be surveyed and removed from site. In the Preparation Lab the marble weighing benches will be surveyed and removed from site. Any remaining equipment, furniture, supplies and cabinets in any of these rooms and will be surveyed and removed from site. The asbestos floor tile will be surveyed, removed and sent for disposal as asbestos waste or radioactive waste.

- Sheet Metal Fabrication Area, Storage Area, Calibration Area and Electronics Lab: The radioactive items on the table top in the Sheet Metal Fabrication Area and the shield cave made from cement blocks on the floor will characterized and shipped for disposal as radioactive waste. Any remaining equipment, furniture, supplies and cabinets in any of these areas will be surveyed and removed from site. The asbestos floor tile will be surveyed, removed and sent for disposal as asbestos waste or radioactive waste.

4.3.1.2.4 Other Buildings

- Tagging Building: The Tagging room furnace, stainless tables, sink, tank and dryer will be surveyed and removed from site. Any remaining equipment, furniture, supplies and cabinets in the room will be surveyed and removed from site.
- Storage Building: The Storage Building is sometimes referred to as the Forklift Garage. The materials and supplies in this building have not been in a radioactive material area and will be removed from site. The equipment could have been used in a radioactive materials area and will be surveyed and removed from site.
- Maintenance Office: The maintenance Office and the contents have never been in a radioactive materials area. The contents will be removed from site or to an appropriate hazardous material disposition site as appropriate. On the south side of the Office there is an added structure that contains a large demineralizer undergoing decay. This demineralizer will be characterized and shipped for disposal as radioactive waste.
- Chemical Shed: The Chemical Shed and the shed contents have never been in a radioactive materials area will removed to an appropriate hazardous material disposition site as appropriate. The building will be demolished and sent to an industrial landfill.
- Compressor Building: The compressors and vacuum pump have never been in a radioactive materials area. The compressors will be surveyed and removed from site. The vacuum pump oil will be sampled to insure that the vacuum pumps are not contaminated internally and they will be surveyed and removed from site.

4.3.1.2.5 Outdoor Areas

- Cooling Towers: The main and secondary cooling towers with associated pumps and controls have never been used with contaminated water and are maintained outside of radioactive materials areas. The cooling towers and the rest of the system will be demolished and removed from site.
- Spare Demin Units: There are three large spare demineralizers that were obtained from the University of California at Berkeley. One unit was in the small addition to the Maintenance Office. Two of these units were never used at ARRR and are stored on the north side of the Storage Building. These units will be characterized and disposed of as radioactive material.
- Waste Sump and Tanks: The waste sump and waste tanks are potentially contaminated. The two waste tanks will be drained if required, surveyed for potential contamination, decontaminated if necessary and shipped offsite for disposal. The Waste Sump pump, valves, and controls will be characterized and shipped for disposal as radioactive waste.

The concrete sump will be excavated, including bottom pad, and shipped for disposal as radioactive waste.

- Shield Blocks: There are quarter circle shield blocks in the vicinity of the waste tanks and the demin system that were used to shield the fence line from high dose rates. These blocks were never in a contaminated area and will be shipped offsite.
- The sanitary and storm sewers will be surveyed at accessible areas and released for continued use.

4.3.1.3 Surveys

Following decontamination and remediation activities of the reactor, a final status survey (FSS) of the entire ARRR site will be performed and documented.

4.3.1.4 Schedule

The project schedule is presented as Figure 4-2, Aerotest estimates that a formal request for termination of Facility License No. R-98 will be submitted to the USNRC approximately eighteen months after removal of fuel from the site in 2055. The preliminary portion of the ARRR Decommissioning (prior to fuel removal) is projected to run from April 2012 through June 2012. Changes to the schedule may be made at Aerotest's discretion as a result of changes to fuel disposition plans, resource allocation, availability of a radioactive waste burial site, ALARA considerations, and further characterization measurements.

4.4 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

Aerotest is committed to, and retains ultimate responsibility for, full compliance with the existing USNRC reactor license and the applicable regulatory requirements during decommissioning. Corporate policies and goals will be followed to ensure high standards of performance in accomplishing the decommissioning tasks.

The Aerotest Director with support from the TRIGA Reactor D&D Oversight Committee and the Aerotest Project Manager, will monitor decommissioning operations to ensure they are being performed safely and according to federal, state, and local regulatory requirements (NRC, EPA, DOT, etc.) and will approve of DOC procedures used during the decommissioning as described in this plan. Consistent with Aerotest policy, the Radiation Safety Committee (RSC) has certain responsibilities to review and approve policies, procedures, programs and facilities pursuant to the safe use of radiological materials and radiation producing equipment. The RSC's jurisdiction will extend to all decommissioning activities dealing with radioactive material and radiological controls.

The planned organization for the ARRR Decommissioning as shown in Figure 4-3 will be maintained, however individuals performing the functions may vary over the project duration. Specialized contractors may be utilized under the direction of the Aerotest Director when necessary and appropriate.

4.4.1 Contractor Assistance

Aerotest management will select a qualified contractor to perform the ARRR Decommissioning Project. The team will consist of Aerotest personnel and the selected contractor. Aerotest will be in charge of overall project management; a Decommissioning Operations Contractor (DOC) will manage the physical decommissioning work, provide health physics support, radiation surveys, and waste packaging, processing, and shipping.

The DOC has not been selected yet. Aerotest will select a contractor through established corporate procurement procedures and standards requiring a rigorous source evaluation and review process. The review and evaluation specifications define scope and method of selection and criteria for contractor qualifications, experience, and reputation. The contractor qualifications and experience required include the following:

a. Demonstration of experience in the performance of the following tasks:

   - Integration of decommissioning, dismantlement and demolition plans.
   - Waste management and other methods used to minimize final waste disposal costs.
   - Decontamination and remediation of facilities and equipment.
   - Use of survey equipment and techniques suitable for compliance with current NRC and MARSSIM survey criteria.
   - Use of inventory and tracking mechanisms to assure accurate waste tracking.
   - Development and execution of radiological and industrial safety programs that will be used during the D & D.
   - Selection, design and/or procurement of appropriate containers and packaging for radioactive and hazardous waste, and transportation to approved treatment and disposal facilities.
   - Performing license termination surveys on a project of similar size and scope.
   - Package, manifest, transport, process and dispose of radioactive waste.

b. The decommissioning contractor selected must have a QA program that meets the requirements under 10 CFR 71 "Packaging and Transportation of Radioactive Material", Subpart H "Quality Assurance". In addition, the contractor's QA program must meet the applicable criteria from 10 CFR 50, Appendix B and the American Society of Mechanical Engineers (ASME) NQA-1. One of the applicable criteria that must be included is a QA Approved Suppliers List.

c. The decommissioning contractor must have an NRC or agreement state license authorizing decommissioning activities at remote sites. The contractor must also have the programs and procedures necessary to perform such activities.

d. The contractor should be prepared to provide qualified personnel, including but not limited to the following:

- Project Manager
- Certified Health Physicist with MARSSIM survey experience
- Waste management
- Industrial hygienist
- Civil and mechanical engineer
- Quality assurance engineer
- Construction supervisor
- Planning and scheduling specialist
- Decontamination and waste technicians
- Radiological safety engineer, foreman and technicians

Official decommissioning project records will be hardcopy documents maintained in secure cabinets. The contractor will maintain working documents, and Aerotest will maintain completed or final project records and documents. These decommissioning files will be available at the Reactor Facility for inspection by the NRC.

Contractors and subcontractors performing work under this Decommissioning Plan will be required to comply with applicable Aerotest policies and procedures.

4.4.2 Aerotest Director

The Aerotest Director functions include:

- Controlling and maintaining safety during decommissioning activities and protecting of the environment
- Reporting of performance
- Approving minor changes to the decommissioning plan and procedures (which do not change the original intent and do not involve an unreviewed safety question)
- Oversight and coordination of Aerotest functional groups and decommissioning contractors
- Ensuring that the conduct of decommissioning activities complies with applicable regulations and is in accordance with Aerotest license.

The minimum qualifications for the ARRR Director are:

- Current or previously certified reactor operator
- Ten years of management experience in the nuclear industry
- Familiarity with the ARRR Facility
- Appropriate training in radiation protection, nuclear safety, hazard communication and industrial safety.

4.4.3 Reactor Supervisor

The functions of the ARRR Reactor Supervisor include:

- Maintaining the TRIGA Reactor in a safe and proper condition during the evolution of Decommissioning Project activities, in accordance with the requirements set forth in the applicable USNRC facility licenses
- Review of plans and procedures
- Providing engineering support for the decommissioning activities

The minimum qualifications for this position are:

- Current or previously certified reactor operator
- At least 4 years of experience in Reactor Operation of a nuclear facility, and a BS degree in a Engineering or Physical Science.

4.4.4 Radiation Safety Officer

The Radiation Safety Officer shall be responsible for providing radiological support in the decommissioning of the ARRR. This function ensures that the activities involving potential radiological exposure are conducted in compliance with the applicable licenses, Federal and State regulations, and ARRR standard operating procedures. The position includes responsibility for maintaining the TRIGA surveillance and monitoring program and for HP radiological protection procedures.

The Radiation Safety Officer for Aerotest will have oversight of all D&D operations. The scope of his oversight will include all D&D operations that involve work with systems or materials that have a radiological component.

The minimum qualifications for this position are:

- A Bachelor's degree in Biological or Physical Science

- A minimum of 2 years experience in personnel and environmental radiation monitoring programs at a nuclear facility
- May be certified as a Health Physicist by the Health Physics Society in lieu of the education and experience requirements given above

The RSO is responsible for ensuring that:

a. Radiological controls are in place prior to and during any work involving radiation

b. Applicable license conditions are satisfied

c. Applicable state and federal regulations are met.

The Radiation Safety Officer has the authority to:

a. Implement any actions necessary to ensure that radiological controls are implemented and followed including review and approval of all procedures involving radiological safety

b. Ensure that routine surveys are performed

c. Stop or modify radiological work immediately and then make changes to RWPs within 24 hours.

4.5 TRAINING PROGRAM

Individuals (employees, contractors and visitors) who require access to the work areas or a radiologically restricted area will receive training commensurate with the potential hazards to which they may be exposed.

Radiation protection training will be provided to personnel who will be performing remediation work in radiological areas or handling radioactive materials. The training will ensure that decommissioning project personnel have sufficient knowledge to perform work activities in accordance with the requirements of the radiation protection program and accomplish ALARA goals and objectives. The principle objective of the training program is to ensure personnel understand the responsibilities and the required techniques for safe handling of radioactive materials and for minimizing exposure to radiation.

Records of training will be maintained which will include trainee's names, dates of training, type of training, test results, authorization for protective equipment use, and instructor's name. Radiation protection training provides the necessary information for workers to implement sound radiation protection practices. The following are examples of the training programs applicable to remediation activities.

4.5.1 General Site Training

A general training program designed to provide orientation to project personnel and meet the requirements of 10 CFR Part 19 will be implemented. General Site Training (GST) will be required for all personnel assigned on a regular basis to the remediation project. This training will include:

- Project orientation/access control
- Introduction to radiation protection
- Quality assurance
- Industrial safety
- Emergency procedures

4.5.2 Radiation Worker Training

Radiation Worker Training (RWT) will be required for all individuals directly associated with the ARRR Decommissioning, and the training will include the following topics:

- Fundamentals of Radiation
- Biological Effects of Radiation
- External Radiation Exposure Limits and Controls
- Internal Radiation Limits and Controls
- ALARA Program (Program, Objectives, Investigational Limits, Keeping Doses ALARA)
- Contamination Limits and Controls
- Management and Control of Radioactive Waste.

Personnel who have documented equivalent RWT from another site may be waived from taking training except for training on Aerotest administrative limits and emergency response, and will be required to pass the written examination and demonstration exercises.

4.5.3 Respiratory Protection Training

Personnel whose work assignments require the use of respiratory protection devices will receive respiratory protection training in the devices and techniques that they will be required to use. The training program will follow the requirements of 10 CFR 20 Subpart H (Ref. 4-3), Regulatory Guide 8.15 (Ref. 4-4), NUREG 0041 (Ref. 4-5) and 29 CFR 1910.134 (Ref. 4-6).

Training will consist of a lecture session and a simulated work session. Personnel who have documented equivalent respiratory protection training may be waived from this training.

4.6 DECONTAMINATION AND DECOMMISSIONING DOCUMENTS AND GUIDES

Health physics, industrial health criteria and other standards that guide the activities described in this Decommissioning Plan are discussed in 5.1 RADIATION PROTECTION, 5.2 RADIOACTIVE WASTE MANAGEMENT, 5.3 GENERAL INDUSTRIAL SAFETY PROGRAM, and 5.4 RADIOLOGICAL ACCIDENT ANALYSES. Relevant documents and guides used are noted therein.

4.7 FACILITY RELEASE CRITERIA

The proposed decommissioning alternative that has been presented in this Decommissioning Plan does not necessitate the major dismantlement of the MAJOR ARRR Buildings. The results of the site and facility radiological characterization have indicated that the building structures are directly releasable without need for extensive decontamination.

This section provides the specific criteria for release of the ARRR. The Final Release survey will use the Derived Concentration Guideline Levels (DCGL's) developed from the characterization survey data (Ref. 4-2) and the current NRC guidance for license termination in Subpart E, Radiological Criteria for License Termination, of 10 CFR Part 20, Standards of Protection Against Radiation (Ref. 4-7). Subpart E, 10 CFR 20.1402, Radiological Criteria for Unrestricted Use (Ref. 4-8), allows termination of a license and release of a site for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent to an average member of a critical group that does not exceed 25 millirem (0.25 millisevert) per year and the residual radioactivity has been reduced to levels that are as low as is reasonably achievable (ALARA). The current NRC guidance for acceptable license termination screening values (meeting the 10 CFR 20.1402 criteria) of common radionuclides for building surface contamination and surface soil contamination are presented in NUREG-1757, Volume 1 Consolidated NMSS Decommissioning Guidance, Decommissioning Process for Materials Licenses, Appendix B, (Ref. 4-9). This information is duplicated in Table 4-6 and Table 4-7. An ALARA analysis is not needed as stated in NUREG-1757, Volume 2, Appendix N (Ref. 4-10). "In light of the conservatism in the building surface and surface soil generic screening levels developed by the NRC staff, the staff presumes, absent information to the contrary, that licensees or responsible parties that remediate building surfaces or soil to the generic screening levels do not need to provide analyses to demonstrate that these levels are ALARA. In addition, if residual radioactivity cannot be detected, it may be assumed that it has been reduced to levels that are ALARA. Therefore, the licensee may not need to conduct an explicit analysis to meet the ALARA requirement" However, for the state of California, more restrictive cleanup requirements will be required.

The State rules for termination of a radioactive materials license are provided in Title 17 California Code of Regulations, Division 1, Chapter 5, Subchapter 4, Section 30256. Currently no dose based release criteria in California and there is a case by case evaluation of decommissioning plans performed by the California Department of Public Health (CDPH). Experience indicates that release limits that equate to a few mrem/yr are accepted. For the purposes of this plan, it was assumed that license termination screening values of 12% (3/25th) of the 10 CFR 20.1402 criteria would be acceptable to California (equivalent to 3 mr/yr above background).

Upon completion of the decontamination and remediation activities (e.g. see Section 4.3 Decommissioning Tasks), a final status survey of the ARRR will be performed using the method described in NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (Ref. 4-11). The results of the survey(s) will be summarized in a report which will be submitted to NRC, as required by the U.S. Nuclear Regulatory Commission NUREG 1537 (Ref. 4-12), in support of a license termination request.

The characterization did not indicate that there was any surface soil contamination. The release criteria for surface soil will be based upon the relative concentrations of isotopes on the material and their respective release criteria, if more than one category of nuclide for beta-gamma emitters applies from Table 4-7.

If additional screening values are required for nuclides not included in Table 4-6 or Table 4-7, they will be calculated using the NRC's D and D Code with default values.

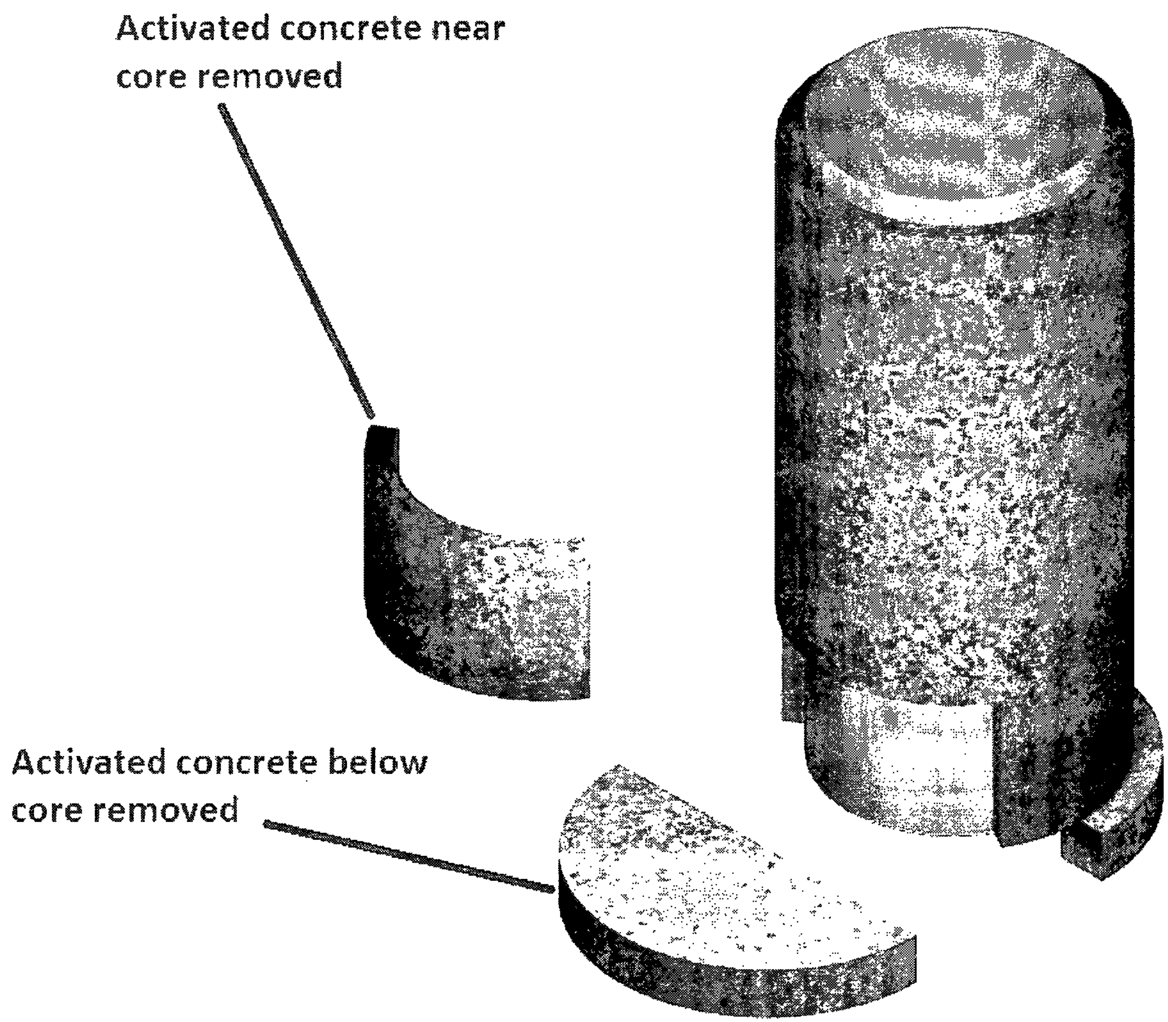

Figure 4-1: ARRR Activated Concrete

ARRR Delayed Decommissioning Schedule

WBS	Task	Start Date	Duration	1-Apr-12	1-May-12	1-Jun-12		1-Apr-55	2-May-55	2-Jun-55	2-Jul-55	2-Aug-55	1-Sep-55	2-Oct-55	2-Nov-55	30-Nov-55	31-Dec-55	30-Jan-56	29-Feb-56	30-Mar-56
1	**Preparatory Work**	**4/3/2012**	**15,928**																	
1.1	Possession Only License	4/3/2012	1																	
1.2	Site cleanup	4/4/2012	90																	
1.3	Storage period	7/3/2012	15,612				→													
1.4	Fuel removal	4/1/2055	30																	
1.5	Update Decommissioning Plan	8/3/2012	60																	
1.6	NRC decommissioning authorization	10/2/2012	90																	
1.7	Project planning & contracting	12/31/2012	45																	
2	**Decommissioning Activities**	**2/20/2013**	**105**																	
2.1	Mobilization and training	2/20/2013	5																	
2.2	Remove shielding around reactor area	2/25/2013	14																	
2.3	Remove reactor and N-Ray components	3/11/2013	14																	
2.4	Empty pool and process water	3/25/2013	7																	
2.5	Decontaminate pool tank	4/1/2013	21																	
2.6	Remove auxiliary systems	4/22/2013	14																	
2.7	Decontaminate facility surfaces	5/6/2013	7																	
2.8	Remove 6-ft of tank and backfill	5/13/2013	7																	
2.9	Perform Final Status Survey	5/20/2013	14																	
2.1	Demobilization	6/3/2013	2																	
3	**Final Activities**	**6/5/2013**	**81**																	
3.1	Submit FSS Report to NRC and California	6/3/2013	1																	
3.2	NRC Review and License Termination	6/4/2013	40																	
3.3	California Review and License Termination	7/14/2013	40																	

Figure 4-2: ARRR Decommissioning Schedule

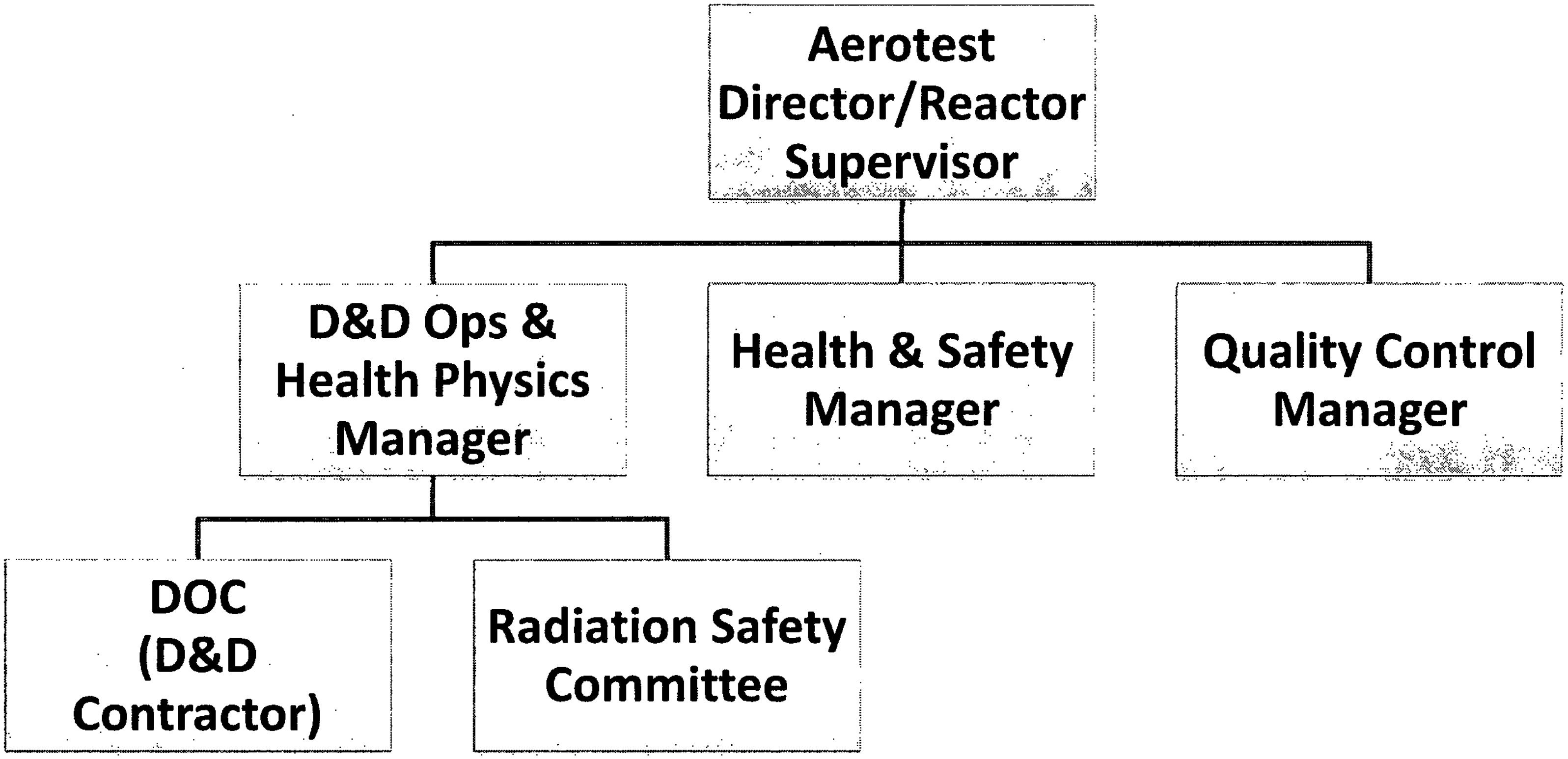

Figure 4-3: ARRR Decommissioning Organization

Table 4-1: List of Expected Radionuclides

Nuclide	Half-Life (yr)	Decay Mode	Inventory Ci
^{3}H*	12.28	β^-	<28.7
^{10}Be	1,510,000	β^-	7.25 x 10-7
^{14}C	5,730	β^-	6.48 x 10-3
^{22}Na	2.60	β^+	1.67 x 10-2
^{35}S	0.2392	β^-	2.99 x 10-4
^{36}Cl	301,000	β^-	2.85 x 10-4
^{39}Ar*	269	β^-	2.61 x 10-7
^{41}Ca	103,000	ε	1.10 x 10-4
^{45}Ca	0.446	β^-	2.54 x 10-3
^{46}Sc	0.233	β^-	2.04 x 10-5
^{54}Mn	0.86	ε	2.26 x 10-3
^{55}Fe	2.73	ε	0.385
^{59}Fe	0.1222	β^-	1.56 x 10-4
^{58}Co	0.194	ε	1.00 x 10-4
^{60}Co	5.27	β^-	0.896
^{59}Ni	76,000	ε	1.34 x 10-4
^{63}Ni	100	β^-	1.75 x 10-2
^{65}Zn	0.67	ε	0.203
^{90}Sr	29.1	β^-	0.454
^{93m}Nb	13.6	IT	3.68 x 10-9
^{94}Nb	20,000	β^-	2.15 x 10-8
^{93}Mo	4,000	ε	1.19 x 10-8
^{95}Zr	0.175	β^-	1.21 x 10-2
^{108m}Ag	418	IT, β^+	0.159
^{110m}Ag	418	β^-, IT	0.878
^{109}Cd	1.27	ε	0.120
^{113m}Cd	14.1	β^-, IT	1.07 x 10-3
^{115m}Cd	0.122	β^-	8.13 x 10-6
^{113}Sn	0.315	β^+	6.74 x 10-4
^{119m}Sn	0.803	IT	2.17 x 10-2
^{121m}Sn	55	IT, β^-	2.82 x 10-5
^{123}Sn	0.354	β^-	2.40 x 10-4
^{124}Sb	0.16	β^-	8.51 x 10-3

Nuclide	Half-Life (yr)	Decay Mode	Inventory Ci
^{125}Sb	2.76	β^-	3.00 x 10-3
^{123m}Te	0.328	IT	3.69 x 10-5
^{125m}Te	0.157	IT	7.31 x 10-4
^{134}Cs	2.7	β^-	9.16 x 10-7
^{137}Cs	30.17	β^-	0.283
^{133}Ba	10.51	ε	6.17 x 10-4
^{139}Ce	0.377	ε	6.94 x 10-9
^{144}Ce	0.78	β^-	0.185
^{152}Eu	13.48	β^-, β^+, ε	1.59
^{154}Eu	8.8	β^-	0.161
^{155}Eu	4.96	β^-	1.73 x 10-2
^{153}Gd	0.659	ε	8.93 x 10-5
^{181}W	0.332	ε	1.12 x 10-7
^{185}W	0.206	β^-	5.12 x 10-7
^{203}Hg	0.128	β^-	4.42 x 10-6
^{204}Tl	3.78	β^+	5.82 x 10-5
^{205}Pb	15,3000,000	ε	9.80 x 10-5
^{210}Po	0.3791	α	1.75 x 10-2
^{238}Pu	87.7	α	4.35 x 10-4
$^{239/40}Pu$	24,110	α	6.95 x 10-4
^{241}Pu	14.35	β^-	1.32 x 10-2
^{241}Am	432.2	α	3.59 x 10-3
^{242}Cm	0.446	α	5.62 x 10-4

Symbols/Abbreviations:

α = Alpha
β^- = Beta
β^+ = Positron
ε = Electron Capture
IT^- = Isomeric Transition

Radionuclide Half-Life values and Decay Mode information used above are taken from Reference 4-13.

Table 4-2: Components with Potential Surface Contamination –Group 1

Area or System	Component
TRIGA Reactor	Demineralizer resin, tanks, pipe loop and floor drains
	Heat exchanger, heat exchanger piping loop
	N-Ray Tray System
	Beam Catcher
	Reactor Bridge
	Shielding blocks
Laboratory Areas	Fume hoods, sink drains, and samples
High Bay Building	Radwaste Room Items
	Pool Tools
	Mezzanine High-Rad storage items
	Storage Wells
Pool Area	Concrete floor Surrounding Pool

Table 4-3: Components with Induced Radioactivity - Group 2

Induced Radioactivity Component
Vertical Beam Tube
Neutron Beam Catcher
Neutron Shutter
N-Ray Facility
Grid Plates
Control Support Structure
Graphite Reflector Elements
Instrument Guide Tubes
Control Rod Guide Tubes
Control Rods
Neutron Source Holder
Ion Chambers and Mounting Assembly
Fasteners and connectors
Graphite Thermal Column
Activated concrete

Table 4-4: Reactor Support Systems – Group 3

Reactor Support System
Heat Exchange System
Demineralizer System
Purge System
Drains

Table 4-5: Equipment Used In Decommissioning Operations - Group 4

Decommissioning Supplies and Equipment
General ventilation system
Temporary localized ventilation system
Confinement barrier
Contaminated tools and equipment
Contaminated clothing

Table 4-6: License Termination Screening Values for Building Surface Contamination

Radionuclide	Symbol	NRC Acceptable screening levels[1] for unrestricted release (dpm/100 cm^2)[2]	California screening levels[3] for unrestricted release (dpm/100 cm^2)[2]
Hydrogen-3 (Tritium)	^{3}H	1.2E+08	1.4E+07
Carbon-14	^{14}C	3.7E+06	4.4E+05
Sodium-22	^{22}Na	9.5E+03	1.1E+03
Sulfur –35	^{35}S	1.3E+07	1.6E+06
Chlorine-36	^{36}Cl	5.0E+05	6.0E+04
Manganese-54	^{54}Mn	3.2E+04	3.8E+03
Iron-55	^{55}Fe	4.5E+06	5.4E+05
Cobalt-60	^{60}Co	7.1E+03	8.8E+02
Nickel-63	^{63}Ni	1.8E+06	2.2E+05
Strontium-90	^{90}Sr	8.7E+03	1.0E+03
Technetium-99	^{99}Tc	1.3E+06	1.6E+05
Iodine-129	^{129}I	3.5E+04	4.2E+03
Cesium-137	^{137}Cs	2.8E+04	3.4E+03
Iridium-192	^{192}Ir	7.4E+04	8.8E+03

[1]Screening levels are based on the assumption that the fraction of removable surface contamination is equal to 0.1. For cases when the fraction of removable contamination is undetermined or higher than 0.1, users may assume, for screening purposes, that 100 percent of surface contamination is removable, and therefore the screening levels should be decreased by a factor of 10. Alternatively, users having site-specific data on the fraction of removable contamination, based on site-specific resuspension factors, (e.g., within 10 percent to 100 percent range) may calculate site-specific screening levels using D and D Version 2.

[2]Units are disintegrations per minute (dpm) per 100 square centimeters (dpm/100 cm2). One dpm is equivalent to 0.0167 becquerel (Bq). Therefore, to convert to units of Bq/m2 multiply each value by 1.67. The screening values represent surface concentrations of individual radionuclides that would be deemed in compliance with the 0.25 mSv/yr (25 mrem/yr) unrestricted release dose limit in 10 CFR 20.1402. For radionuclides in a mixture, the "sum of fractions" rule applies; see Part 20, Appendix B, Note 4.

[3]For the purposes of this estimate, it was assumed that license termination screening values of 12% (3/25th) of the 10 CFR 20.1402 criteria would be acceptable to California (equivalent to 3 mr/yr above background).

Table 4-7: License Termination Screening Values for Surface Soil

Radionuclide	Symbol	NRC Surface Soil Screening Values for unrestricted release (pCi/g)[2]	California Surface Soil Screening Values for unrestricted release (pCi/g)[3]
Hydrogen-3 (Tritium)	^{3}H	1.1E+02	1.3E+01
Carbon-14	^{14}C	1.2E+01	1.4E+00
Sodium-22	^{22}Na	4.3E+00	5.2E-01
Sulfur –35	^{35}S	2.7 E+02	3.2E+01
Chlorine-36	^{36}Cl	3.6 E-01	4.3E-02
Calcium-41	^{41}Ca	6.4 E+01	7.7E+00
Calcium-45	^{45}Ca	5.7 E+01	6.8E+00
Scandium-46	^{46}Sc	1.5E+01	1.8E+00
Manganese-54	^{54}Mn	1.5E+01	1.8E+00
Iron-55	^{55}Fe	1.0E+04	1.2E+03
Cobalt-57	^{57}Co	1.5E+02	1.8E+01
Cobalt-60	^{60}Co	3.8E+00	4.6E-01
Nickel-59	^{59}Ni	5.5E+03	6.6E+02
Nickel-63	^{63}Ni	2.1E+03	2.5E+02
Strontium-90	^{90}Sr	1.7E+00	2.0E-01
Niobium-94	^{94}Nb	5.8E00	7.0E-01
Technetium-99	^{99}Tc	1.9E+01	2.3E+00
Iodine-129	^{129}I	5.0E-01	6.0E-02
Cesium-134	^{134}Cs	5.7E+00	6.8E-01
Cesium-137	^{137}Cs	2.8E+04	3.4E+03
Europium-152	^{152}Eu	8.7E+00	1.0E+00
Europium-154	^{154}Eu	8.0E+00	9.6E-01
Iridium-192	^{192}Ir	4.1E+01	4.9E+00
Lead-210	^{210}Pb	9.0E-01	1.1E-01
Radium-226	^{226}Ra	7.0E-01	8.4E-02
Radium-226+C	^{226}Ra+C	6.0E-01	7.2E-02
Actinium-227	^{227}Ac	5.0E-01	6.0E-02
Actinium-227+C	^{227}Ac+C	5.0E-01	6.0E-02
Thorium-228	^{228}Th	4.7 E+00	5.6E-01

Radionuclide	Symbol	NRC Surface Soil Screening Values for unrestricted release (pCi/g)[2]	California Surface Soil Screening Values for unrestricted release (pCi/g)[3]
Thorium-228+C	^{228}Th+C	4.7 E+00	5.6E-01
Thorium-230	^{230}Th	1.8 E+00	2.2E-01
Thorium-230+C	^{230}Th+C	6.0 E-01	7.2E-02
Thorium-232	^{232}Th	1.1 E+00	1.3E-01
Thorium-232+C	^{232}Th+C	1.1 E+00	1.3E-01
Protactinium-231	^{231}Pa	3.0 E-01	3.6E-02
Protactinium-231+C	^{231}Pa+C	3.0 E-01	3.6E-02
Uranium-234	^{234}U	1.3 E+01	1.6E+00
Uranium-235	^{235}U	8.0 E+00	9.6E-01
Uranium-235+C	^{235}U+C	2.9 E-01	3.5E-02
Uranium-238	^{238}U	1.4 E+01	1.7E+00
Uranium-238+C	^{238}U+C	5.0 E-01	6.0E-02
Plutonium-238	^{238}Pu	2.5 E+00	3.0E-01
Plutonium-239	^{239}Pu	2.3 E+00	2.8E-01
Plutonium-241	^{241}Pu	7.2 E+01	8.6E+00
Americium-241	^{241}Am	2.1 E+00	2.5E-01
Curium-242	^{242}Cm	1.6 E+02	1.9E+01
Curium-243	^{243}Cm	3.2 E+00	3.8E-01

[1]These values represent surficial surface soil concentrations of individual radionuclides that would be deemed in compliance with the 25 mrem/y (0.25 mSv/y) unrestricted release dose limit in 10 CFR 20.1402. For radionuclides in a mixture, the "sum of fractions" rule applies; see Part 20, Appendix B, Note 4.

[2]Screening values are in units of (pCi/g) equivalent to 25 mrem/y (0.25 mSv/y). To convert from pCi/g to units of becquerel per kilogram (Bq/kg) divide each value by 0.027. These values were derived using D and D screening methodology (NUREG/CR-5512, Volume 3). They were derived based on selection of the 90th percentile of the output dose distribution for each specific radionuclide (or radionuclide with the specific decay chain). Behavioral parameters were set at the mean of the distribution of the assumed critical group. The metabolic parameters were set at "Standard Man" or at the mean of the distribution for an average man.

[3]"Plus Chain (+C) " indicates a value for a radionuclide with its decay progeny present in equilibrium. The values are concentrations of the parent radionuclide, but account for contributions from the complete chain of progeny in equilibrium with the parent radionuclide (NUREG/CR-5512 Volumes 1, 2, and 3).

REFERENCES FOR SECTION 4

4-1 CS-HP-PR-004, *Historical Site Assessment of the Aerotest Radiography and Research Reactor, San Ramon, California*, Revision 0, July 2011.

4-2 CS-HP-PR-007, *Characterization Report for the Aerotest Radiography & Research Reactor, San Ramon, California*, Revision 0, August 2011.

4-3 10 CFR 20 Subpart H, *Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas*.

4-4 Regulatory Guide 8.15, *Acceptable Programs for Respiratory Protection*; Revision 1, October, 1999

4-5 NUREG 0041, *Manual of Respiratory Protection Against Airborne Radioactive Materials*

4-6 29 CFR 1910.134, Respiratory Protection

4-7 10 CFR 20 Subpart E, Radiological Criteria for License Termination

4-8 410 CFR 20.1402 Radiological Criteria for Unrestricted Use

4-9 NUREG-1757, *Consolidated MSS Decommissioning Guidance, Decommissioning Process for Materials Licenses*, September 2006

4-10 NUREG-1757, *Consolidated MSS Decommissioning Guidance, Decommissioning Process for Materials Licenses, Appendix N, ALARA Analyses*, September 2006

4-11 NUREG-1575, Revision 1, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, August 2000

4-12 NUREG 1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, February 1996

4-13 *The Health Physics and Radiological Health Handbook*, Revised Edition 1992, Editor by B. Shleien.

5.0 OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

5.1 RADIATION PROTECTION

5.1.1 Ensuring As Low As Reasonably Achievable (ALARA) Radiation Exposures

Decommissioning activities at the ARRR involving the use and handling of radioactive materials will be conducted in a manner such that radiation exposure will be maintained As Low As Reasonably Achievable (ALARA), taking into account the current state of technology and economics of improvements in relation to the benefits.

ALARA Program

The Aerotest practice during this project will be as follows:

A documented ALARA evaluation will be required for specific tasks if a Project HP determines that 5% of the applicable dose limits (collective dose) for the following may be exceeded:

- Total Effective Dose Equivalent (TEDE) (5 rem)
- The sum of the Deep-Dose Equivalent (DDE) and the Committed Dose Equivalent (CDE) to any individual organ or tissue other than the lens of the eye (50 rem)
- Eye Dose Equivalent (EDE) (15 rem)
- Shallow-Dose Equivalent to the skin or any extremity (SDE) (50 rem)

Decommissioning Project management positions responsible for radiation protection and maintaining exposures ALARA during decommissioning include the ARRR Director and Radiation Safety Officer.

Methods for Occupational Exposure Reduction

Various methods will be utilized during the Decommissioning Project work to ensure that occupational exposure to radioactive materials is kept ALARA. The methods include the Radiological Work Permit (RWP), special equipment, technique, and practices as described in the following subsections. Work will be performed in accordance with reactor licenses and/or this Decommissioning Plan.

Radiological Work Permits (RWPs)

A Radiation Work Permit (RWP) will be used for the administrative control of personnel entering or working in areas that have radiological hazards present. Work techniques will be specified in such a manner that the exposure for all personnel, individually and collectively, are maintained ALARA. RWPs will not replace work procedures, but will act as a supplement to procedures. Radiation work practices will be considered when procedures are developed for work that will take place in a radiologically controlled area.

Project RWPs will describe the job to be performed, define protective clothing and equipment to be used, and personnel monitoring requirements. RWPs will also specify any special instructions or precautions pertinent to radiation hazards in the area including listing the radiological hazards present, area dose rates and the presence and intensity of hot spots, loose surface radioactivity, and other hazards as appropriate. The HP organization will ensure that radiation, surface radioactivity and airborne surveys are performed as required to define and document the radiological conditions for each job.

RWPs for jobs with low dose commitments will be approved at the HP technician or HP supervisory level while RWPs for jobs with potentially high dose commitment or significant radiological hazards will be approved by the RSO. Examples of topics covered by implementing procedures for the Radiation Work Permits are:

- Requirements, classifications and scope for RWPs;
- Initiating, preparing and using RWPs;
- Extending expiration dates of an RWP; and
- Terminating RWPs

Respiratory Protection and TEDE ALARA Evaluations

The use of engineering controls to mitigate the airborne radiological hazard at the source will be the first choice with respect to controlling the concentrations of airborne radioactive material. There may be, however, circumstances where engineering controls are not practical or may not be sufficient to prevent airborne concentrations in excess of those that constitute an airborne radioactivity area. In such circumstances where worker access is required, respiratory protective equipment will be utilized to limit internal exposures. Any situation wherein workers are allowed access to an airborne radioactivity area, or allowed to perform work that has a high degree of likelihood to generate airborne radioactivity in excess of 0.1 DAC, the decision to allow access will be accompanied by the performance of representative measurements of airborne radioactivity to assess worker intake. The results of DAC-hour tracking and air sample results for intake will be documented in accordance with appropriate regulations. Workers will provide nasal smears for HP evaluation following the use of respiratory protective equipment for radiological purposes, as necessary.

Control and Storage of Radioactive Materials

The Aerotest HP Program establishes radioactive material controls that ensure:

- Deterrence of inadvertent release of licensed radioactive materials to unrestricted areas.
- Confidence that personnel are not inadvertently exposed to licensed radioactive materials.
- Minimization of the volume of radioactive wastes generated during the decommissioning.

All material leaving the Restricted Area will be surveyed to ensure that radioactive material is not inadvertently released from the ARRR. See Section 5.1.3 "Radioactive Materials Controls" for a description of the specific survey methods that will be used.

5.1.2 Health Physics Program

Project Health Physics Program - General

Aerotest has procedures in place that will be implemented during the ARRR Decommissioning Project. If additional Health Physics procedures are required at some point in the work to support the decommissioning, they will be developed and approved in accordance with Aerotest Health Physics policy and procedure.

Aerotest senior management is readily accessible to ensure timely resolution of difficulties that may be encountered. The RSO, while organizationally independent of the Project staff, has direct access to the ARRR Director on a daily basis, and have full authority to act in all aspects of protection of workers and the public from the effects of radiation. Conduct of the ARRR Decommissioning Project HP program will be evaluated according to Aerotest policy.

Audits, Inspections, and Management Review

During Decommissioning Project work, aspects of the Project may be assessed and reported by the Contractor's Quality Assurance Department, through audits, assessments and inspections of various aspects of decommissioning performance, including HP, as described in Section 3.2.4 Program Quality Assurance.

Audits of the Aerotest Health Physics program are conducted in accordance with the requirements of 10 CFR 20. These audits will include aspects of the ARRR Decommissioning Project.

Additional assessments or management reviews may be performed when deemed appropriate by the Aerotest.

Health Physics Equipment and Instrumentation

HP equipment and instrumentation suitable to permit ready detection and quantification of radiological hazards to workers and the public will be chosen to ensure the validity of measurements taken during remediation and final release surveys. The selection of equipment and instrumentation to be utilized will be based upon detailed knowledge of the radiological contaminants, concentrations, chemical forms and chemical behaviors that are expected to exist as demonstrated during radiological characterization, and as known from process knowledge of the working history of the ARRR. Equipment and instrumentation selection also takes into account the working conditions, contamination levels and source terms that are reasonably expected to be encountered during the performance of decommissioning work, as presented in this Plan.

The following sections present details of the equipment and instrumentation planned for use during the decommissioning. It is anticipated that through retirement of worn or damaged equipment/instrumentation or increase in quantities of available components or instruments, that new technology will permit upgrades or, at a minimum, like-for-like replacements. Aerotest is committed to maintaining conformance to minimum performance capabilities stated in this Plan whenever new components or instruments are selected.

Criteria for Selecting Equipment and Instrumentation for Conduct of Radiation and Contamination Surveys and Personnel Monitoring

A sufficient inventory and variety of instrumentation will be maintained on site to facilitate effective measurement of radiological conditions and control of worker exposure consistent with ALARA, and to evaluate the suitability of materials for release to unrestricted use. Instrumentation and equipment will be capable of measuring the range of dose rates and radioactivity concentrations expected to be encountered during the decontamination and decommissioning activities associated with the ARRR, including implementation of a final status survey.

Project HP staff will select instrumentation that is sensitive to the minimum detection limits for the particular task being performed, but also with sufficient range to ensure that the full spectrum of anticipated conditions for a task or survey can be met by the instrumentation in use. Consumable supplies will conform to manufacturer and/or regulatory recommendation to ensure that measurements meet desired sensitivity and are valid for the intended purpose.

Storage, Calibration, Testing and Maintenance of Health Physics Equipment and Instrumentation

Survey instruments will be stored in a common location under the control of ARRR Decommissioning Project HP personnel. A program to identify and remove from service inoperable or out-of-calibration instruments or equipment as described in HP procedures will be adhered to throughout the ARRR Decommissioning Project. Survey instruments, counting equipment, air samplers, air monitors and personnel contamination monitors will be calibrated at license-required intervals, manufacturer-prescribed intervals (if shorter frequency) or prior to use against standards that are NIST traceable in accordance with approved calibration laboratory procedures, HP procedures, or vendor technical manuals. Survey instruments will be operationally checked daily when in use. Counting equipment operability will be verified daily when in use. The personnel contamination monitors are operationally tested on a daily basis when work is being performed.

Specific Health Physics Equipment and Instrumentation Use and Capabilities

Table 5-1 provides details of typical HP equipment and instrumentation that is planned for use in the ARRR Decommissioning Project. This list is neither inclusive nor exclusive.

Policy, Method, Frequency and Procedures

The ARRR Decommissioning Project will utilize the existing Aerotest HP Program for the Project. Aerotest's existing program will be augmented as necessary using plans and procedures provided by the decommissioning contractor.

Airborne Effluent Monitoring — During the decommissioning effort where a temporary barrier with an exhaust system is in use, the ventilation system exhaust points from the temporary barrier will be sampled continuously downstream of the HEPA filtration system.

Radiation Surveys — Radiation, airborne radioactivity and contamination surveys during decommissioning will be conducted in accordance with approved HP procedure(s). The purposes of these surveys will be to (1) protect the health and safety of workers, (2) protect the health and safety of the general public, and (3) demonstrate compliance with applicable license, federal and state requirements, as well as Decommissioning Plan commitments. HP personnel will verify the validity of posted radiological warning signs during the conduct of these surveys. Surveys will be conducted in accordance with procedures utilizing survey instrumentation and equipment suitable for the nature and range of hazards anticipated. Equipment and instrumentation will be calibrated and, where applicable, operationally tested prior to use in accordance with procedural requirements. Routine surveys are conducted at a specified frequency to ensure that contamination and radiation levels in unrestricted areas do not exceed license, federal, state or site limits. HP staff will also perform surveys during decommissioning whenever work activities create a potential to impact radiological conditions.

Personnel Monitoring - Internal and External — External monitoring will be conducted in accordance with approved procedures. Prospective external exposure evaluations will be performed prior to initiating decommissioning activities and whenever changes in conditions warrant. Visitors to the ARRR will be monitored in accordance with requirements specified in Aerotest HP procedures and according to the radiological hazards of areas to be entered.

Internal monitoring will be conducted in accordance with approved procedures. This prospective internal exposure evaluation will be evaluated on an annual basis, at a minimum, or whenever significant changes in planned work evolutions warrant it. A comprehensive air-sampling program will be conducted at the ARRR to evaluate worker exposures regardless of whether internal monitoring is specified. The results of this air-sampling program will be utilized to ensure validity of specified internal monitoring requirements for decommissioning personnel. If, at any time during the decommissioning, hazards that may not be readily detected by the preceding measures are encountered, special measures or bioassay, as appropriate, will be instituted to ensure the adequate surveillance of worker internal exposure.

Monitoring will be required if the prospective dose evaluation shows that an individual(s) dose is likely to exceed 10% of the applicable limits, and for individuals entering a high or very high radiation area.

Respiratory Protection - The Decommissioning Project respiratory protection program will include direction for use of National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) certified equipment. This program will be reviewed and approved by Aerotest HP to ensure adherence to the requirements of 10CFR20. The Aerotest industrial hygienist may be consulted to advise the decommissioning contractor on issues about air quality and the use of respiratory protection. The Director of Environmental Health and Safety has supervisory control over the IH position.

NIOSH/MSHA approved air purifying respirators include full face piece assemblies with air purifying elements to provide respiratory protection against hazardous vapors, gases, and/or particulate matter to individuals in airborne radioactive materials areas. Individuals may be required to use continuous or constant flow full-face airline respirators for work in areas with actual or potential airborne radioactivity. The RSO will also ensure that the respiratory protection program meets the requirements of 10 CFR Part 20, subpart H.

Maintenance — When respiratory protection equipment requires cleaning, the filter cartridges will be removed. The respirator will be cleaned and sanitized after every use with a cleaner/sanitizer and then rinsed thoroughly in plain warm water in accordance with HP procedures.

Storage — Respiratory protective equipment will be kept in proper working order. When any respirator shows evidence of excessive wear or has failed inspection, it will be repaired or replaced. Respiratory protective equipment that is not in use will be stored in a clean dry location.

<u>Contamination Control</u> - Contamination control measures that will be employed include, as appropriate, the following:

- Worker training will incorporate methods and techniques for the control of radioactive materials, and proper use and donning/doffing of protective clothing
- Procedures will incorporate HP controls to minimize spread of contamination during work
- Radiological surveys will be scheduled and conducted by HP
- Containment devices such as designed barriers, containers and plastic bags will be used to prevent the spread of radioactive material
- Physical decontamination of ARRR areas or items
- Physical barriers such as Herculite sheeting, strippable paint, and tacky mat step-off pads to limit contamination spread
- Posting, physical area boundaries and barricades
- Clean step-off pads at the entrance point to contaminated areas

Personnel entries into radiological contaminated areas will require the use of protective clothing. This clothing will consist of a suitable combination of items such as the following, dependent upon the conditions outlined in the RWP:

- Heavyweight lab coat

- Heavyweight canvas, cotton, or cotton/polyester coveralls
- Heavyweight hoods
- Plastic calf-high booties
- Rubber, plastic or cloth shoe covers
- Plastic or rubber gloves which may require cloth liners.
- Tyvek paper coveralls or plastic rain suit disposable outer clothing
- Face shield or other protective device

Access Control - A Restricted Area (RA) will be established and properly posted and monitored to prevent unauthorized access.

Engineered Controls - Personnel exposure to airborne radioactive materials will be minimized by utilizing engineering controls such as the following:

- Ventilation devices — in-place or portable HEPA filters or ARRR ventilation systems, local exhaust by use of vacuums
- Containment devices — designed containment barriers, containers, plastic bags, tents, and glove-bags
- Source term reduction — application of fixatives prior to handling, misting of surfaces to minimize dust and resuspension

Airborne Radioactivity Monitoring - Monitoring for the intake of radioactive material is required by 10 CFR 20.1502(b) if the intake is likely to exceed 0.1 ALI (annual limit on intake) during the year for an adult worker, or if the committed effective dose equivalent is likely to exceed 0.10 rem (1.0 mSv) for the occupationally exposed minor or declared pregnant woman. Air sampling will be performed in areas where airborne radioactivity is present or likely.

Prospective estimates of worker intakes and air concentrations used to establish monitoring requirements will be based on consideration of the following:

- The quantity of material(s) handled
- The ALI for the nuclides of interest
- The release fraction for the radioactive material(s) based upon its physical form and use
- The type of confinement being used for the material(s) being handled
- Other factors that may be applicable

HP personnel will use technical judgment in determining the situations that necessitate air sampling regardless of generalized, prospective evaluations done for the ARRR.

Prior to identifying the location for an air sampler, the purpose of the radiological air sample will be identified. Various reasons exist for collecting air samples. The following are a few examples:

- Estimation of worker intakes
- Verification of confinement of radioactive materials
- Early warning of abnormal airborne concentrations of radioactive materials
- Determining the existence of criteria for posting an Airborne Radioactivity Area (ARA).

Smoke tubes and buoyant markers may then be used to determine airflow patterns in the area. Airflow patterns may be reevaluated if there are changes at the ARRR that may impact the validity of the sampling locations. Such factors might include the following:

- Changes in the work process
- Changes in the ventilation system
- Use of portable ventilation that might alter earlier assessments

After identifying the purpose for the air sample and establishing flow patterns, air sample locations are chosen as follows:

- For verification of confinement of radioactive materials:
    - Locate samplers in the airflow near the potential or actual release point.
    - More than one sampling point may be appropriate when there are more than one potential or actual release points.
- For estimation of a worker intake, the sampler intake will be located as close to the worker's breathing zone as practical without interfering with the work or worker

General workplace air sampler intakes will not be placed in or near ventilation exhaust ducts unless their purpose is to detect system leakage during normal operation, and if quantitative measurements of workplace concentrations are not required. Locations or number of air samplers will be changed when dictated by modifications to facility structure, changes in work processes, or elimination of potential sources.

A sufficient inventory and variety of operable and calibrated portable and semi-portable air sampling equipment will be maintained to allow for effective collection, evaluation, and control of airborne radioactive material and to provide backup capability for inoperable equipment. Air sampling equipment will be calibrated at prescribed intervals or prior to use against certified equipment having known valid relationships to nationally recognized standards. Table 5-1 includes anticipated air-sampling equipment.

When the work being performed is a continuous process, a continuous sample with a weekly exchange frequency is appropriate. For situations where short-lived radionuclides are important considerations, the exchange frequency will be adjusted accordingly. Longer sample exchange frequencies may be approved by HP management for situations where airborne radioactive material and nuisance dust are expected to be relatively low. Grab sampling for continuous processes may also be approved by HP management based upon consideration of variability of the expected source term for the facility and process. Grab sampling is the appropriate means of airborne sampling for processes conducted intermittently, and for short duration radiological work that involves a potential for airborne release.

Potential Sources of Radiation or Contamination Exposure to Workers and Public as a Result of Decommissioning Activities

Sources of radiation or contamination exposure may be assessed by process knowledge, radiological survey data, surveys performed during characterization, previous and current job coverage surveys, or daily, weekly and monthly routine surveys.

Classification of potential sources may also be identified by radionuclide, physical properties, volatility and radioactivity.

Worker exposure to significant external deep-dose radiation fields is considered unlikely during this project due to the nature of the contaminants and/or the work precautions and techniques employed. Worker exposure to airborne radioactivity may occur during decontamination operations/work evolutions that may involve abrasives or methods that volatilize loose and/or fixed contamination.

Exposure of the public to external or internal radiation from this Decommissioning Project is not considered credible because of the confinement provided by the facility and the access control provided for the facility and the area surrounding it.

The types of exposure controls used take into account the current state of technology and the economics of improvements in relation to the benefits. Control of potential sources of radiation exposure to workers and public as a result of decommissioning activities will be achieved through, but not limited to, the use of administrative, engineering and physical controls.

Administrative controls consist of, but are not limited to:

- Administrative dose limits that are lower than regulatory limits
- Training
- Radiological surveys.

Physical barriers such as radiological warning rope/ribbon, in combination with radiological warning tape, lockable doors/gates as well as information signs and flashing lights or other applicable barriers may also be used.

Engineering controls may consist of but are not limited to:

- HEPA ventilation/enclosures
- Protective clothing/equipment

- Access restrictions/barriers
- Confinement.

Health Physics Policies for Contractor Personnel

Contractor personnel will be used during the ARRR Decommissioning Project. Contractors who will work with licensed radioactive materials will be required to:

- Attend and complete appropriate radiation safety course
- Provide required exposure history information
- Read and sign an applicable RWP and comply with instructions
- Follow all special instructions given by HP.

5.1.3 Radioactive Materials Controls

Aerotest's radiation protection program establishes radioactive material controls that ensure the following:

- Prevention of inadvertent decommissioning radioactive waste (licensed) material release to uncontrolled areas.
- Assurance that personnel are not inadvertently exposed to radiation from licensed radioactive decommissioning waste materials.
- Minimization of the amount of radioactive waste material generated during decommissioning.

Decommissioning waste materials will not be released as clean waste. Such waste materials to be removed from the reactor facility will be shipped to an off-site licensed radioactive waste processing facility for survey, processing and disposal.

Pool water releases will be analyzed and filtered to ensure that discharges to sanitary sewerage will meet the requirements of 10 CFR 20.2003 disposal by release into sanitary sewerage and Aerotest liquid discharge procedures.

5.1.4 Dose Estimates

The total projected occupational exposure to complete the decommissioning of the ARRR is estimated to be 18.34 person-rem. This estimated was taken from NUREG/CR-1756 (Ref. 5-1). The estimate in this document was developed for a reference research reactor, a 1,000 kW TRIGA reactor.

This estimate is provided for planning purposes only. Detailed exposure estimates and exposure controls shall be developed during detailed planning of the decommissioning activities. Area dose rates used for this estimate are based on process knowledge and current survey maps (where available).

The dose estimate to members of the public as a result of decommissioning activities is estimated to be negligible. This is because site perimeter controls will restrict members of the public from the area where decommissioning activities are taking place. This is consistent with the estimate given for the "reference research reactor" in the "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" (NUREG-0586) (Ref. 5-2). The dose to the public during decommissioning (DECON) and truck transport transportation of radioactive waste from the reference research reactor referred to in the Final Generic Impact Statement is estimated to be "negligible (less than 0.1 man-rem)."

Activated pieces and any contaminated debris will be removed and shielded if required to meet U.S. DOT shipping requirements and disposal site Waste Acceptance Criteria.

5.2 RADIOACTIVE WASTE MANAGEMENT

5.2.1 Radioactive Waste Processing

The processes of decontamination, remediation and dismantlement of the ARRR will result in solid and liquid low-level radioactive waste, mixed waste and hazardous waste. Limited soil remediation is anticipated which will result in solid radioactive waste. This waste will be handled (processed and packaged), stored and disposed of in accordance with applicable sections of the Code of Federal Regulations (CFR), disposal site Waste Acceptance Criteria, California Department of Public Health requirements, ARRR Licenses and Permits, and the applicable implementing plans and procedures. Radioactive waste processing includes waste minimization or volume reduction, radioactive and hazardous waste segregation, waste characterization, neutralization, stabilization, solidification and packaging.

5.2.2 Radioactive Waste Disposal

Low-level radioactive waste will be processed and packaged for disposal at a licensed low-level waste site such as the Clive Utah site. The volume of low-level radioactive waste is estimated at 4,714 cu. ft. Mixed low-level waste will be prepared for shipment to off-site commercial processing and disposal facilities such as the Clive Utah site.

10 CFR 61, Licensing Requirements for Land Disposal of Radioactive Waste, Subpart D — Technical Requirements for Land Disposal Facilities, establishes minimum radioactive waste classification, characterization and labeling requirements. These requirements will be ensured through the implementation of project packaging and characterization procedures, Disposal Site Waste Acceptance Criteria for the contractor selected disposal site(s) and the Project-Specific Quality Assurance Plan. Training/ Qualifications will be provided for project waste management personnel to assure conformance to applicable 10 CFR 61 requirements as stated in the specific implementing procedures and plans. Audits and surveillances will be conducted per the Project-Specific Quality Assurance Plan based on ASME-NQA-1 and the requirements of 10 CFR 71.

10 CFR 71, Packaging and Transportation of Radioactive Material, establishes requirements for packaging, shipment preparation and transportation of licensed material. Aerotest is licensed by the USNRC to receive, possess, use and transfer licensed byproduct and source materials. 10 CFR 71 requirements will be met through the implementation of Aerotest approved packaging

and shipping procedures. Training will be provided for waste management personnel to assure conformance to applicable 10 CFR 71 requirements. Quality Assurance will confirm conformance to 10 CFR 71 Subpart H (Quality Assurance) requirements through the implementation of a Aerotest approved Project-Specific Quality Assurance Plan.

10 CFR 20.2006, *Transfer for Disposal and Manifests*, establishes requirements for controlling transfers of low-level radioactive waste intended for disposal at a land disposal facility; establishes a manifest tracking system; supplements requirements concerning transfers and record keeping; and requires generator certification that transported materials are properly classified, described, packaged, marked and labeled, and are in proper condition for transport. These requirements will be met through the implementation of project and Aerotest packaging and shipping procedures with the oversight of DOC and Aerotest Quality Assurance.

Radiological and mixed wastes will be disposed of at disposal sites per the applicable Disposal Site's Acceptance Criteria. Associated implementing plans and procedures will reflect the characterization, processing, removal of prohibited items, packaging and transportation requirements. Appropriate documentation will be submitted to designated disposal sites including, as required, certification plans, qualification statements, assessments, waste stream analysis, evaluations and profiles, transportation plans, and waste stream volume forecasts. Waste characterization, waste designation, waste traceability, waste segregation, waste packaging, waste minimization, and quality assurance and training requirements of the designated disposal sites will be incorporated in implementing procedures to assure conformance to disposal site requirements.

Generator State (California) and Treatment/Storage/Disposal Facility States (Utah, etc.) requirements for radioactive and mixed waste management will be incorporated into plans and procedures to assure conformance with applicable state regulations, licenses and permits. Applicable state regulations include California Department of Public Health requirements and Utah Department of Environmental Quality Rules (R313) for the control of ionizing radiation reflected in the Clive Utah Radioactive Material License.

Radioactive waste will be staged in designated controlled areas in accordance with USNRC 10 CFR 19 and 20 requirements. Mixed wastes will be staged in designated controlled areas per EPA 40 CFR requirements, 10 CFR 19 and 20, and per local and state permits. Measures will be implemented through plans and procedures to control the spread of contamination, limit radiation levels, prevent unauthorized access, prevent unauthorized material removal, prevent tampering, and prevent weather damage. The designated controlled areas will be approved by Radiological Work Permits (RWP) and/or Hazardous Work Permits (HWP). An HWP will be used when controls are imposed to protect against non-radiological hazards. The Aerotest Health and Safety manger is responsible for HRWP approval. The Aerotest Director has supervisory control over the Aerotest Health and Safety Manager.

Radioactive and mixed waste material will be packaged for shipment per 10 CFR, 40 CFR, 49 CFR, and the designated Disposal Site Criteria and placed in permitted interim storage (staged) until shipped. The quantity of waste packages staged for shipment will be a function of waste generation and packaging rate, shipment preparation rate, shipment rate, and disposal site acceptance rate. To meet this objective, shipments will be scheduled throughout the life of the Project to designated treatment, storage, and disposal facilities.

Radioactive material storage areas will be contained inside posted restricted areas according to existing Aerotest procedures and consistent with 10 CFR 20.

5.3 GENERAL INDUSTRIAL SAFETY PROGRAM

Industrial safety and industrial hygiene personnel, along with project management, shall be responsible to ensure that the project meets all occupational health and safety requirements. The primary functional responsibility is to ensure compliance with the OSHA of 1973. Specific responsibilities include conducting an industrial training program to instruct employees in general safe work practices; reviewing Decommissioning Project procedures to verify adequate coverage of industrial safety and industrial hygiene concerns and requirements; performing periodic inspections of work areas and activities to identify and correct any unsafe conditions and work practices; providing industrial hygiene services as required; and advising Project management on industrial safety matters and on the results of periodic safety inspections.

All personnel working on the ARRR Decommissioning Project will receive Health and Safety training in order to recognize and understand the potential risks involving personnel health and safety associated with the work at the ARRR. The Health and Safety training implemented at the ARRR is to ensure compliance with the requirements of the USNRC (10 CFR), the EPA (40 CFR), and OSHA (29 CFR). Workers and regular visitors will be familiarized with plans, procedures and operation of equipment to conduct their selves safely. In addition, each worker must be familiar with procedures that provide for good quality control. Section 4.5, *Training Program*, provides additional information.

5.4 RADIOLOGICAL ACCIDENT ANALYSES

Radiological Accident Analyses Potential radiological accidents during decommissioning the ARRR were evaluated by determining ARRR components and areas that contain the highest radioactive material inventory. The proposed decommissioning activities and methods in which radioactive material could be released to the work area or environment were considered. Since all special nuclear material will have been removed prior to decommissioning, the majority of the accidents discussed in the current license are not applicable. The accident identification process was supplemented by reviewing experiences at other non-power reactor decommissioning projects. The following radiological accidents were considered to present the highest potential consequences:

- Fire in Waste Storage Area
- Fire in activated graphite
- Dropped and damaged ion exchange column
- Dropped irradiated hardware liner
- Transportation accident

5.4.1 Fire in Waste Storage Area

The consequences, of a fire during decommissioning of the ARRR were considered and are not significantly different than the consequences of a fire during reactor operations. Most materials are metals, concrete, or similar non-combustible materials. Although some torch cutting operations may be performed during decommissioning, the likelihood is low that a fire would start or that a fire could become intense enough to release radioactive material.

Dry radioactive waste is normally collected and packaged in metal containers to limit the volume of dry radioactive waste available for consumption by fire. The accident scenario is for a fire to occur in the dry solid waste. It was assumed that the activity concentration in this material will be 10% of the concentration for the pool cleanup resin. This is very conservative as the resin column concentrates activity and most of the dry solid waste will have minimal contamination. A waste inventory of 360 cubic feet of dry solid waste was estimated to contain 0.11 millicuries assuming it would have the same activity distribution as the resin. It is estimated that combustion of this material would release approximately 25% of the contamination in a respirable form. The total exposure was estimated to be 30 mrem, to which the external dose is a negligible contributor.

5.4.2 Fire in Activated Graphite

As part of the decommissioning process there will be activated graphite removed. The accident scenario is for a fire in the graphite material. The graphite is approximately 2-feet by 2-feet by 4-feet and was assumed it would catch fire even though it is currently contained in aluminum. The graphite volume of 16 cubic feet was estimated to contain 1.41 Ci based upon an independent activation analysis of this graphite material. The calculated inventory is Eu-152 (93.3%), Eu-154 (6.5%) with other radionuclides less than 1%. It is estimated that combustion of this structural material would release approximately 25% of the contamination in a respirable form. The total exposure was estimated to be 182 mrem, to which the external dose is a negligible contributor.

5.4.3 Dropped Ion Exchange Column

An uncontrolled release of airborne radioactivity could occur during demolition activities involving contaminated or activated materials such as the pool water demineralizer. The pool water demineralizer was estimated to contain 3.3 cubic feet of resin and 79 millicuries based upon an independent laboratory analysis of the resin. Most of the inventory is Sr-90 (38%), Cs-137 (24%), Ce-144 (15%), Cd-109 (10%), Eu-154 (4%), Fe-55 (1.5%), Eu-155(1.4%), Nb-95(1.1%), Pu-241 (1.1%), Zr-95 (1.0%), with other radionuclides less than 1%. The worst-case accident scenario would be dropping the resin column liner as it is being lifted. It was conservatively assumed that 1 percent of the activity of this column was respirable and 10 percent of the respirable material escaped the column and became airborne during the accident. The TEDE is less than 1 mrem, to which the external dose is a negligible contributor.

5.4.4 Dropped Irradiated Hardware Liner

Most of the activity in the reactor pool is contained in activated components that include control rods, instrument tubes, etc. Because cutting operations for components will be performed underwater, no cutting accident releases were postulated. However a liner filled with irradiated hardware could be dropped while it is lifted for placement into a shipping container. The activity in irradiated hardware is contained within the metal structure of the hardware item except for surface contamination. It would be highly unlikely for a component in the liner to break. If it did break, the diameters of any particles produced would be large enough that it is unlikely that the particles would remain airborne and be respirable. However, even though it is not plausible that an accident could result in measurable exposures at the site boundary, this scenario was evaluated because it includes the largest curie inventory and it demonstrates that potential exposures to the public are acceptable even when worst case assumptions are utilized.

A waste shipping liner containing 120 cubic feet of activated hardware was estimated to contain 1,460 curies (NUREG/CR-1756-v1, Tables E1-2, E.1-3, E.1-5 and E.1-6, for Reference Research Reactor). Most of the activity is Co-60 (84%), with Fe-55 (8.6%), Mn-54 (4.7%), and smaller inventories of other radionuclides. The worst-case accident scenario would be dropping the filled liner as it is being lifted. It was assumed that 1 percent of the activity of this liner was respirable and 1 percent of the respirable material escaped the liner and became airborne during the accident. The total exposure was estimated to be 143 mrem, to which the external dose is a negligible contributor.

5.4.5 Transportation Accidents

Various forms and quantities of radioactive waste will be shipped from the ARRR during the D&D project. The dose consequence from transportation accidents could be higher than the contamination accident scenarios described above because high-activity reactor components could be involved. As such, there is a potential for a moderate dose consequence of between 1 and 25 mrem for the public following a transportation accident. However, adherence to NRC and DOT radioactive material packaging and transportation requirements is considered a sufficient control measure for mitigating transportation-related incidents.

5.4.6 Accident Analysis Summary

The accident analysis shows that the postulated accident scenarios would result in TEDE's to the nearest member of the public that are less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem), USEPA 1992 (Ref.5-3) but for some scenarios, somewhat larger than the NRC normal operational annual dose limits for individual members of the public of 0.1 rem/yr (100 mrem/yr) 10 CFR 20.1301 (Ref. 5-4).

The results of the accident analysis show that off-site consequences from accidents are well below the U.S. EPA's PAGs; therefore, off-site emergency plans are not needed.

Table 5-1: Health Physics Equipment and Instrumentation

Instrument/ Detector Type	Radiation Detected	Scale/Range	Typical Background	Typical MDC – 95% Confidence Level	Usage
Scintillation (Ludlum 2350-1 or equivalent) ratemeter/scalar with Ludlum 43-89	Alpha Beta	0 to 500,000 cpm	<10 cpm ~300 cpm	100 dpm/100 cm^2 (direct alpha) 700 dpm/100 cm^2 (direct beta) 1,500 dpm/100 cm^2 (scan)	In process and FSS
Ludlum Model 2350-1/Ludlum 43-68 or equivalent Gas Flow Proportional	Alpha Beta	0 to 500,000 cpm	<10 cpm ~300 cpm	100 dpm/100 cm^2 (direct alpha) 400 dpm/100 cm^2 (direct alpha/beta) 1,100 dpm/100 cm^2 (scan)	In process and FSS
Ludlum Model 2350-1/Ludlum 43-37 or equivalent Gas Flow Proportional Floor Monitor	Alpha Beta	0 to 500,000 cpm	<30 cpm ~1,200 cpm	5,500 dpm/100 cm^2 (scan)	In process and FSS
Ludlum Model 2350-1/Ludlum HP-260 or equivalent, Geiger-Mueller (20 cm^2 Pancake)	Beta Gamma	0 to 500,000 cpm 720 cpm = 0.2 μR/h	100 cpm	2,100 dpm/100 cm^2 (direct) 8,000 dpm/100 cm^2 (scan)	General characterization and in process surveys
Ludlum Model 19 Micro-R meter or equivalent 1 in by 1 in NaI detector	Gamma	0 to 3,000 μR/h or 0 to 5,000 μR/h	5 to 8 μR/h	1 to 2 μR/h	General characterization and in process surveys
3" by ½" NaI scintillation detector digital scalar or equivalent	Gamma	0 to 500,000 cpm	2,500 cpm avg. shielded 7,000 cpm avg. unshielded	250 cpm 500 cpm	General characterization and in process surveys
Ludlum Model 2350-1/Ludlum 44-10 or equivalent 2" x 2" NaI scintillator	Gamma	0 to 500,000 cpm	5 to 8 μR/h 10,000 cpm	96 pCi/g (3% Enriched U) 107 pCi/g (20% Enriched U) 118 pCi/g (50% Enriched U) 132 pCi/g (75% Enriched U)	In process and FSS
Bicron AB-100 scintillation detector	Alpha Beta	0 to 500,000 cpm	<10 cpm ~750 cpm closed beta ~1,500 cpm open beta	70 dpm/100 cm^2 (direct) 850 dpm/100 cm^2 (direct) 3,900 dpm/100 cm^2 (scan)	General characterization and in process surveys
Canberra ISOCS Gamma Spectroscopy System, HPGe	Gamma	N/A	Varies with geometry and configuration	Varies with geometry and configuration	In process and FSS
Eberline Personnel Contamination Monitor PCM-1B	Gas Flow Proportional	N/A			Personnel contamination monitor
F&J Model HV-1 "Hi-Vol"	N/A	5-30 cfm	N/A	N/A	High volume air sampling
F&J Model LV-14M Gooseneck "Lo-Vol"	N/A	0.35-3.5 cfm	N/A	N/A	Low volume air sampling
Ludlum Model 333-2 air monitor	GM	10-105 cpm			Local airborne monitor

REFERENCES FOR SECTION 5

5-1 NUREG/CR-1756, *Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors*, March 1982

5-2 NUREG-0586, *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities*, August 1988

5-3 U.S. EPA, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, 400-R-92-001, 1992

5-4 USNRC, 10 CFR Part 20.1301, *Dose Limits for Individual Members of the Public*

6.0 PROPOSED FINAL RADIATION SURVEY PLAN

The intended course of action for ARRR decommissioning, based upon consideration of site and facility radiological characterization results, is to decontaminate structural materials to the extent practicable in balance with radioactive waste minimization considerations, and dismantle ARRR systems to the extent necessary for remediation, and packaging for burial those materials that cannot reasonably be decontaminated. As such, the Final Status Survey Plan (FSSP) (and subsequent Final Status Survey Report) discussed in this section deals with release of the ARRR building structures and grounds to unrestricted use. This section will also discuss the survey methods that will be utilized.

6.1 DESCRIPTION OF FINAL STATUS SURVEY PLAN

The purpose of the Final Status Survey is to demonstrate that the radiological condition of the ARRR structures is at or below established release criteria (see Section 4.7). It is anticipated that the U.S. NRC will then terminate the ARRR reactor licenses and release all areas of the ARRR.

Note that within the context of this section, the term DCGL refers to the release criteria specified in section 4.7, *Facility Release Criteria.*

The guidance as contained in the following regulatory documents was used in the development of this section of the decommissioning plan and should be used as guidance for the development of the FSSP.

- NUREG 1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (Ref. 6-1);
- NUREG–1757, Vol. 2, *Consolidated NMSS Decommissioning Guidance; Characterization, Survey, and Determination of Radiological Criteria* (Ref. 6-2); and
- NUREG 1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions* (Ref. 6-3).

When developed, the FSSP should also incorporate the following major elements:

- Radiological contaminants;
- DCGL summary;
- Area classification;
- Survey areas and units;
- Survey methodology;
- Survey instrumentation;
- Survey design;
- Data assessment; and

- Quality control.

The FSS will be designed to demonstrate that licensed radioactive materials have been removed from the Site to the extent that any remaining residual radioactivity is below the radiological criterion for unrestricted release. If the survey results pass the requirements of the FSSP, the survey unit will be suitable for unrestricted release. If survey results do not meet the criteria and testing requirements as specified, additional investigation and remediation as required will be performed. Additional investigations will include an evaluation of the survey design, instrumentation used and the statistical evaluations, as necessary.

6.1.1 Area Classification

Based upon information collected during the historical site assessment and measurements and sampling during the characterization, all soils and building structures were assigned a classification. The initial area classifications for the ARRR are provided Table 6-1.

6.1.2 Non-Impacted Areas

Non-impacted areas are defined as areas that have no reasonable potential for residual contamination. These include areas that have no impact from Site operations based upon the location(s) of licensed operations, Site use, topography, Site discharge locations, and other Site physical characteristics. These areas include the outlying land areas of the Site and would not require FSS surveys to satisfy regulatory requirements for unrestricted release.

6.1.3 Impacted Areas

Impacted areas may contain residual radioactivity from licensed activities. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2, or Class 3 designations as listed below.

- Class 1 areas are impacted areas that have or are expected to have concentrations of residual radioactivity that exceed the DCGL(s) or areas identified to have contamination in excess of the DCGL(s) prior to remediation will be considered Class 1;
- Class 2 areas are impacted areas that are not likely to have concentrations of residual radioactivity that exceed the DCGL(s); and
- Class 3 areas are impacted areas that are not expected to contain residual radioactivity or residual radioactivity levels are a small fraction of the DCGL(s).

Class 1 areas receive the highest degree of survey effort because they have the greatest potential for contamination, followed by Class 2 then Class 3 areas. When the available information was not sufficient to designate an area as a particular class, the area was classified as Class 1. Areas that are considered to be on the borderline between classes received the more restrictive classification.

Areas that have been classified based on contamination potential shall be further divided into survey units. An FSS will be performed in each survey unit and the data evaluated to demonstrate compliance with the release criterion. However, specific survey units are not defined in this document and shall be defined in the FSSP.

6.1.4 Background Reference Areas

Background reference area measurements are required when using statistical application of the Wilcoxon Rank Sum (WRS) test, and when background subtraction is required to correct gross activity measurements for natural activity present in materials prior to applying the Sign test.

Due to the age and location of the ARRR building structures, background reference areas may not be readily available for assessment and it may be appropriate to use typical material specific backgrounds per NUREG 1507 and/or other guidance documents.

Open land (soil) reference areas have a soil type similar to the soil type within the impacted survey units. If additional reference areas are required, consideration will be given to selecting reference areas that are most similar in terms of physical, chemical, geological, and biological characteristics.

6.1.5 Data Quality Objectives

The Data Quality Objectives (DQO) process will be used for designing and conducting all FSS. The appropriate design for a given survey will be developed using the DQO process. The seven steps of the DQO process are bulleted below.

- State the Problem
- Identify the Decision
- Identify Inputs to the Decision
- Define the Study Boundaries
- Develop a Decision Rule
- Specify Limits on Decision Errors
- Optimize the Design for Obtaining Data

6.1.6 Decision Errors

The probability of making decision errors is established as part of the decision process in establishing performance goals for the data collection design and can be controlled by adopting a scientific approach through hypothesis testing. In this approach, the survey results will be used to select between the null hypothesis or the alternate condition (the alternative hypothesis) as defined and shown below.

- Null Hypothesis (H_0) – the survey unit does not meet the release criterion; and
- Alternate Hypothesis (H_a) – The survey unit does meet the release criterion.

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criterion, or false negative. This occurs when the null hypothesis is rejected when in fact it is true. The probability of making this error is designated as "α".

A Type II decision error would result in the failure to release a survey unit when the residual radioactivity is below the release criterion, or false positive. This occurs when the Null Hypothesis is accepted when it is in fact not true. The probability of making this error is designated as "β".

Appendix E of NUREG 1757, Vol. 2, recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the guidance in NUREG 1757, α will be set at 0.05. A β value of 0.10 will initially be selected. The β value may be modified, as necessary, after weighing the resulting change in the number of required sampling and measurement locations against the risk of unnecessarily investigating and/or remediation of survey units that are truly below the release criterion.

6.1.7 Statistical Tests

Appropriate tests will be used for the statistical evaluation of the survey data. Tests such as the Sign test and WRS test will be implemented using the unity rule, surrogate methodologies, or combinations thereof as described in MARSSIM.

If background is a significant fraction of the DCGL, the WRS test will be used. If the contaminant is not in the background or constitutes a small fraction of the DCGL, the Sign test will be used. This Sign test will be utilized for the building and structural surface surveys with the ability to subtract material specific and ambient gamma radiation background levels.

6.1.8 Integrated Survey Strategy

The integration of survey techniques and the systematic sampling and measurement is the final step in the survey design. This integration produces an overall strategy for performing the survey.

6.1.9 Scan Coverage

The amount of area to be covered by scan measurements is based upon the survey unit classification as described in Table 5.9 of MARSSIM and Table A.2 of NUREG 1757, Vol. 2, and is summarized in Table 6-2. The emphasis will be placed on a higher frequency of scans in areas of higher risk. This is referred to as a graded approach.

For Class 3 survey areas, biased surface scans will typically be performed on areas with the greatest potential of contamination. For open land areas, this may include surface drainage areas and collection points. For building and structural surfaces such as overhead surveys, this will include overhead horizontal surfaces and air collection systems.

6.1.10 Reference Grid

A reference grid will be used for reference purposes and to locate the sampling and measurement locations. The reference grid may be physically marked during the survey to aid in the collection of samples and measurements.

6.1.11 Systematic Sampling and Measurement Locations

Systematic sampling and measurement locations for Class 1 and Class 2 survey units will be located in a systematic pattern or grid. The grid spacing, L, will be determined using MARSSIM Equation 5-5 based upon the survey unit size and the minimum number of sampling or measurement locations determined. For Class 3 survey units, each sampling and measurement location will be randomly selected using a random number generator.

The systematic sampling and measurement locations within each survey unit will be clearly identified and documented for the purposes of reproducibility. Actual measurement locations will be marked and identified by tags, labels, flags, stakes, paint marks, photographic record, or equivalent.

6.1.12 Remediation and Reclassification

Based upon the survey data, it may be necessary to remediate the entire survey unit or only a portion of it. If an individual survey measurement (scan or direct) in a Class 2 survey unit exceeds the DCGL, the survey unit, or portion of the survey unit, will be evaluated, and if necessary, be reclassified to a Class 1 area and the survey re-designed and re-performed accordingly. If an individual survey measurement in a Class 3 survey unit exceeds 25% of the DCGL, the survey unit, or portion of a survey unit, will be evaluated, and if necessary, reclassified to a Class 2 survey unit and the survey re-designed and re-performed accordingly. After the elevated survey measurement is confirmed but cannot be thoroughly described as an isolated condition, i.e., it cannot be demonstrated with great certainty that this condition does not exist elsewhere in the survey unit; the survey unit will be reclassified. If the result cannot be duplicated, the individual and average measurement results with respect to the DCGL will be reviewed, and if the variability does not suggest the initial classification was inappropriate, the survey unit will not be reclassified.

6.1.13 Survey Instrumentation

Radiation detection and measurement instrumentation for the FSS will be selected to provide both reliable operation and with the best possible sensitivity to detect the Radionuclides Of Concern (ROCs). When possible, instrumentation selection will be made to identify the ROC at levels sufficiently below the DCGL. Detector selection will be based upon detection sensitivity, operating characteristics, and expected performance in the field. The instrumentation will, to the extent practicable, use data logging to automatically record measurements to minimize transcription errors.

Commercially available portable and laboratory instruments and detectors that will be used to perform survey measurements and sample quantification may include:

- Surface scanning;
- Direct surface contamination measurements;
- Gamma spectroscopy analysis of soil and other bulk materials;
- Alpha spectroscopy analysis of soil and other bulk materials; and
- Liquid scintillation counting of soil and other bulk materials.

Radiation detection and measurement instrumentation will be selected based on the type and quantity of radiation to be measured. The instruments used for direct measurements will be capable of detecting the radiation of concern to a MDC between 10% and 50% of the applicable DCGL values to the maximum extent practical. The use of 10% to 50% of the DCGL is an administrative limit only. Any value below the DCGL for a Class 1 area is acceptable.

Instruments and detectors will be calibrated for the radiation types and energies of interest or to a conservative energy source. Instrument calibrations will be documented with calibration certificates and/or forms and maintained with the instrumentation and project records. Calibration labels will also be attached to all portable survey instruments. Prior to using any survey instrument, the current calibration will be verified and all operational source and background checks will be performed.

Instrumentation used for FSS will be calibrated and maintained in accordance with approved calibration procedures. Radioactive sources used for calibration will be traceable to NIST and have been obtained in standard geometries to match the type of samples being counted. When a characterized high purity germanium (HPGe) detector is used, suitable NIST-traceable sources will be used for calibration, and the software set up appropriately for the desired geometry.

It will be necessary to determine the scan sensitivity for field instrumentation utilized during the FSS. This will determine the effectiveness of the surface scans in the ability to determine whether an area meets the criteria for release and will also be a factor in determining the number of samples and measurements that will be required to demonstrate compliance. Scan speeds will be established to the maximum extent practical to detect contamination at or below the release criteria for both open land soil and structural subsurface contamination surveys.

The scan MDC for open land areas may be reduced further by using the field instrumentation coupled with a GPS unit by enabling the scan data to be logged, downloaded, and mapped. By logging and mapping the data, it enables the scan data to be reviewed in its entirety as a data set in correlation with survey unit characteristics such as paved areas and surface soils vs. subsurface soils, etc. By being able to statistically review the data by color coding and adjusting ranges of data values, patterns and areas of concern can be identified more readily than during real time scanning by the survey technician. Additionally, by using the GPS system, it is more readily available to relocate specific areas for further investigation, survey, and sampling as necessary. This effectively maximizes the surveyor efficiency, thereby reducing the scan MDC.

Table 6-3 provides a list of typical laboratory analysis methods and the associated MDCs (sensitivities) expected for FSS for the SEFOR site. Methods listed are standard industry methods from the EPA and the Environmental Measurements Laboratory (EML).

Upon completion of the decontamination and remediation activities, a FSS will be performed per the guidance described in MARSSIM (Ref. 6-1). The results of the FSS will be summarized in a FSSR which will be submitted to the NRC and California Department of Public Health in support of the license termination requests.

6.1.14 Survey Design

Survey measurements and sample collection will be performed by personnel trained and qualified in accordance with applicable DOC procedures. The techniques for performing survey measurements and collecting samples, such as chain-of-custody, will also be specified in DOC procedures.

A gamma walkover survey (GWS) will be performed in outdoor soil areas with portable survey instruments sensitive to gamma radiation, to locate contamination in soil or other media. The survey instrument typically used will be a 2-in by 2-in NaI gamma scintillation detector. Scanning will generally be conducted by moving the detector in a serpentine pattern over the surface at a rate that does not exceed 1.5 feet per second (0.5 meters per second). A surface contamination monitor (sensitive to alpha or beta radiations, or both) will be used to monitor structural surfaces. The surface scans will be conducted by moving the detector at a rate of approximately one detector width per second with the detector held as close to the surface as possible without touching the surface. Alternate scan speeds and source to detector distances may be implemented to ensure the survey DQOs are met.

Both random and biased surveys will be performed. Biased surveys will be based on results of historical surveys, walk-downs, historical use of the area, areas remediated, characterization surveys, and professional judgment.

Soil samples should be collected from areas of elevated radiation identified during the GWS requiring investigation to evaluate if the soil activity concentration meets remediation criteria.

6.1.14.1 Survey Methods

Survey measurements and sample collection are performed by personnel trained and qualified in accordance with the applicable procedure. The techniques for performing survey measurements or collecting samples are specified in approved procedures.

The survey methods to be employed in the FSS will consist of combinations of gamma scans, scanning and static measurements of total surface contamination, and soil sampling. Any new technologies will meet the applicable DQOs, and the technical approach should be documented and/or outlined in the FSSP for review.

6.1.14.2 Scanning

Scanning is the process by which the survey technician passes a portable radiation detector within close proximity to the surface of a soil volume, or the surfaces of buildings/equipment with the intent of identifying residual radioactivity. Scan surveys that identify locations where the magnitude of the detector response exceeds an investigation level indicating that further investigation is warranted to determine the amount of residual radioactivity.

6.1.14.3 Total Surface Contamination Measurements

Static measurements of total surface contamination are obtained by stationing the detector in close proximity to the surface, counting for a pre-determined time interval, and recording the reading. Total surface contamination measurements may be collected at random locations within a survey unit, or may be collected at systematic locations. Total surface contamination measurements may also be collected at locations of elevated radioactivity identified by scan surveys as part of an investigation to determine the source of the elevated instrument response, or at locations likely to contain residual radioactivity based on knowledge of operational history and professional judgment.

6.1.14.4 Removable Surface Contamination (Smears)

Removable contamination or smear surveys will be performed to verify that the average level of H-3 within a survey unit meets the release limit per Section 4.7. A smear for H-3 will be performed at each direct surface radioactivity measurement location. A 100 cm^2 surface area will be wiped with a dampened circular cloth or paper filter using moderate pressure. Smear samples will normally only be obtained in building surfaces or in open land areas where hard standing structures are identified (concrete, asphalt, etc.).

6.1.14.5 Volumetric Sampling

Sampling is the process of collecting a portion of a medium as a representation of the locally remaining medium. The collected portion of the medium is then analyzed to determine the radionuclide concentration.

When and if necessary, bulk material samples may be analyzed via gamma spectroscopy, alpha spectroscopy or liquid scintillation counting as appropriate.

6.1.15 Soil Surveys

If the survey instrument scan MDC is less than the DCGL (using the unity rule), then scanning will be the primary method of surveying areas post-remediation. The average net count rate corresponding to the DCGL (or some fraction of) will be determined and used to guide the remediation. Once the area surface scans indicate levels below the DCGL, samples may be collected to confirm the scan results.

If the scan MDC is greater than the DCGL, scanning will still likely be initially used to guide remediation, but additional soil samples may be needed as the area approaches the level that can be released for unrestricted use. Suspect contaminated soil will be sampled and analyzed to determine if the levels are below DCGL.

6.1.16 Building Structural Surveys

For areas to be remediated or where there is a potential for residual surface activity, operational type surveys with surface contamination monitors will be performed. Scanning the surface at a rate of approximately one detector width per second will be performed to identify any areas of residual activity that exceed the gross activity DCGL. The count rate that corresponds to the gross activity DCGL will be determined for the instrument used and the surveyor will mark areas exceeding this value with paint, a marker, or other identifying means.

Following remediation, the area will be rescanned. When the area has been effectively remediated, a post-remediation survey will be documented. The results will be provided to the FSS engineer for evaluation and FSS.

Once the area has been determined to be ready for FSS, isolation and control measures will be established to ensure the area does not become further impacted by the surrounding remediation efforts. The isolation and control process may include posting or restricting access to the area.

6.1.17 Field Screening – Capability of Detection at DCGL

Table 5-1 form the previous section shows typical field instruments for performing in-process surveys. The same, or similar, instruments will be used during FSS. The typical MDCs are noted to be low enough to measure concentrations at the DCGL for field instruments used for scanning.

6.1.18 Investigation Levels

During the FSS, any areas of concern will be identified and investigated. This will include any areas as identified during the scan survey and any results identified during survey data post-processing and review that exceed the investigation levels. Based on this review, the suspect areas will be addressed by further biased surveys and sampling as necessary. In Class 1 and 2 areas, the investigation level will be set at the scan MDC for scan surveys and at the DCGL for direct measurement / sample results to ensure elevated areas are identified. In Class 3 areas, the investigation level will be set at the scan MDC for scan surveys and at 50% of DCGL based on MARSSIM guidance.

6.2 FINAL STATUS SURVEY REPORT

The FSS planning, data, and assessment information will be compiled for each survey unit. The documentation shall provide a complete and unambiguous record of the radiological status of each survey unit relative to the established DCGLs. The information provided will also allow for an independent evaluation of the FSS results at a later time, including a repeat survey, commonly referred to as a confirmatory survey.

The following list provides a summary of the information that will be provided in the FSS report as a minimum.

- Overview of the results of the FSS;

- Discussion of changes that were made in the FSS from what was proposed in this document;
- Description of the method by which the number of samples was determined for each survey unit;
- Number of measurements/samples performed/collected in the survey unit;
- Description of the survey unit, including maps of measurement and sampling locations showing random start systematic locations for Class 1 and 2 survey units and random locations for Class 3 survey units;
- Discussion of remedial actions and unique features;
- Measured sample concentrations in units that are comparable to the DCGL;
- Statistical evaluation of the measured concentrations;
- Judgmental and miscellaneous sample data sets reported separately from systematic data;
- Discussion of anomalous data, including areas of elevated direct radiation detected during scanning that exceeded investigation levels or measurement locations in excess of the DCGL;
- A statement that the survey unit satisfied the DCGL and the maximum, as necessary;
- Description of any changes in the initial survey unit assumptions relative to the extent of residual radioactivity;
- Description of how ALARA practices were employed to achieve final activity levels; and
- If a survey unit fails, a description of the investigation process and a discussion of the impact of the failure on other survey units and the site in general.

Table 6-1: ARRR Initial Classifications

Building / Structure / Site Area	Section	Subsection	Classification
Reactor High Bay Building	Reactor Area	Reactor Enclosure	1
		N-Ray Exposure	2
		South End Radiography	2
		Rad Material Storage Room	1
		Office Supply Room	2
		Machine Shop	2
		Employee Lockers	2
	Mezzanine	N-Ray Gauge Office	2
		Preparation lab	2
		Chemical lab	2
		Sheet Metal Fabrication Area	2
		Instrument Calibration Area	2
		Storage	1
		Electronics Lab	2
	Office Area	General Managers Office	2
		Accounting Office	3
		Business Office	3
		Ladies' Room	3
		Men's Room	3
		Control Room	2
		Lunch/Conference Room	2
	Outside	Walls & Roof	3
Building Addition 1	Office Area	Office Space	3
		Customer Viewing Area	3
		Quality Control Room	3
		Dark Room	3
		Hallway	3
	High Ceiling Area	Explosive Storage Safe	3
		Film Storage Room	3
		Shipping & Receiving	3
		N-Ray Setup Area	3
		Counting Room	1
	Outside	Walls & Roof	3
Tagging Building	High Roof Area	Entry Vestibule	3
		Tagging Room	3
		Back Room	3
		Safe	1

Building / Structure / Site Area	Section	Subsection	Classification
	Outside	Walls & Roof	3
Storage Building	Inside	All Surfaces	3
	Outside	Walls & Roof	3
Demineralizer Building	Inside	All Surfaces	1
	Outside	Walls & Roof	3
Heat Exchanger Building	Inside	All Surfaces	1
	Outside	Walls & Roof	3
Compressor Building	Inside	All Surfaces	3
	Outside	Walls & Roof	3
Maintenance Office Bldg	Inside	All Surfaces	2
	Outside	Walls & Roof	3
Chemical Storage Shed	Inside	All Surfaces	3
	Outside	Walls & Roof	3
Waste Storage Tank Area	Above Grade	Tank Pads	2
	Sump Area	Sump	1
Soil Areas	Inside Fence	All Areas	3
Paved Areas	Inside Fence	North parking Area	3
		East of Buildings	3
	Outside Fence	Front Parking lot	3

Table 6-2: Scan Coverage

Area Classification	Scan Coverage	Surface Activity Measurements or Soil Samples
Class 1	100%	As determined by statistical tests; additional measurements/samples to account for small areas of elevated activity as necessary
Class 2	10 to 100%	As determined by statistical tests
Class 3	1 to 10% (Judgmental)	

Table 6-3: Laboratory Analysis Methods and Sensitivities

Analyte	Medium	Method	Sensitivity Soil (pCi/g)	Sensitivity Smears (dpm/100 cm^2)	Description
H-3	Soil and Smears	EML-LV-539-17 or equivalent	10	100	Liquid Scintillation
Co-60	Soil	EML GA-01-R MOD, EPA 901.1 or equivalent	0.5	N/A	Gamma Spectrometry
Cs-137	Soil	EML GA-01-R MOD. EPA 901.1 or equivalent	1.0		Gamma Spectrometry
Pu-238/239	Soil	EML A-01-R MOD, STM D-3972 or equivalent	0.2		Alpha Spectrometry

REFERENCES FOR SECTION 6

6-1 NUREG-1575, Revision 1, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, August 2000

6-2 NUREG-1757, Volume 2, Revision 1, *Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria*, September 2006

6-3 NUREG 1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*

7.0 TECHNICAL SPECIFICATIONS

After the nuclear fuel is removed from the reactor and shipped off site, most of the technical specifications for the operating license will not apply after the license is amended to possession – only or modified by an order to decommission as discussed in NUREG-1537 (Ref. 7-1). The applicable Technical Specifications for the ARRR TRIGA Reactor decommissioning will be set forth in an amendment request to Facility License No. R-98, Docket No. 50-228.

As decommissioning progresses, further requests for changes to the Technical Specifications may be submitted in an application for amendment to the license pursuant to 10 CFR 50.59.

REFERENCES FOR SECTION 7

7-1 NUREG- 1537 Rev. 0, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*

8.0 PHYSICAL SECURITY PLAN

All radiation restricted areas are secured from unauthorized entry. During non-working hours, all nuclear facility sensitive areas are locked. Aerotest maintains routine surveillance of the reactor site through a private alarm company which is authorized to contact the local police department.

Existing physical security and material control and accounting plans approved by the Nuclear Regulatory Commission as may be amended will continue to be implemented.

These existing plans meet the requirements in NUREG-1537, Chapter 17 (Ref. 8-1)

REFERENCES FOR SECTION 8

8-1 NUREG- 1537, Rev. 0, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*.

9.0 EMERGENCY PLAN

As required by the USNRC, Aerotest has a Reactor Facility Emergency Plan for responding to emergencies at the Reactor Facility. The purpose of this plan is to minimize any emergency's effect on the public, personnel, reactor facility and the environment surrounding the facility. Removal of spent fuel from the site would significantly reduce the potential for significant release of radioactive material off site. Any airborne or liquid releases due to decommissioning activities would have negligible impact off site. The most likely accident scenario is a contaminated and/or injured individual. This scenario is adequately addressed by the existing emergency plan. Training will be provided to key personnel to ensure their familiarity with the emergency plan and their expected responses.

10.0 ENVIRONMENTAL REPORT

The Environmental Report (Ref. 10-1) is provided as Appendix B.

REFERENCES FOR SECTION 10

10-1 *Environmental Report Aerotest Radiography and Research Reactor San Ramon, California*, July 2011.

11.0 CHANGES TO THE DECOMMISSIONING PLAN

As the decommissioning progresses, and up until the termination of the license, changes to the Technical Specifications will be via a Request for License Amendment pursuant to 10 CFR 50.90 (Ref. 11-1).

Aerotest requests that changes to the Decommissioning Plan be allowed with local approval by the Aerotest Director and the Aerotest Health and Safety Manager, and without prior USNRC approval, unless an unreviewed safety question is involved. An unreviewed safety question involves:

1. The increase of probability of occurrence or the increase of consequences of an accident or malfunction of equipment important to safety compared to that situation previously evaluated in the SAR, or
2. The possibility for an accident or malfunction of a different type than previously analyzed in the SAR, or
3. The reduction in margin of safety as defined in the SAR.

Reports and records of changes to the Decommissioning Plan, and retention of documents, will be in accordance with the applicable portions of 10 CFR 50.59 (Ref. 11-2).

REFERENCES FOR SECTION 11

11-1 10 CFR 50.90, Application for amendment of license or construction permit.

11-2 10 CFR 50.59, Changes, tests and experiments.

APPENDIX A
SUMMARY OF CHARACTERIZATION RESULTS

SUMMARY OF CHARACTERIZATION RESULTS

The Characterization Report provides the results of characterization and survey activities performed at the Aerotest Radiography and Research Reactor (ARRR) facility in San Ramon, California. The primary purpose of the characterization report was to provide information for development of a Decommissioning Plan for the facility. The reactor facility is owned and licensed by Aerotest Operations, Inc., which awarded a contract to EnergySolutions, LLC to perform characterization activities and develop a Decommissioning Plan and associated cost estimate. The characterization activities were designed to define the nature, extent and location of residual radioactive material and other hazardous materials that remain in the facility. For reactor facilities such as the ARRR, characterization also includes determining the amount of neutron-activated materials in structural components and in the biological shield. The Characterization Report includes survey results for alpha, beta and gamma removable activity and fixed activity on surfaces such as floors, walls, equipment and in some areas, on ceilings. It also includes exposure rate measurements taken throughout the facility to determine general area gamma radiation levels and surveys of specific radioactive sources and radioactive waste currently present at the facility. In addition to radiological surveys, samples of soil, wood, spent resin, water and other known or potentially contaminated materials were collected and sent to an off-site laboratory for analysis.

Before performing characterization activities, a Historical Site Assessment (HSA) Report was generated using information gathered by EnergySolutions during an on-site visit that took place in March 2011. The HSA process included reviewing facility historical operations, obtaining reactor and facility design data, reviewing operational data and reports, conducting interviews with in-house personnel, and obtaining facility radiological survey data. This information was used to ensure that radiological surveys, sampling, and assessments performed during characterization activities were properly designed.

The HSA Report identifies certain systems, structures, and components that might contain asbestos-containing materials (ACM), mercury-containing equipment (switches, thermometers, light ballasts, etc.), polychlorinated biphenyls (PCBs) in electrical components and other hazardous materials such as lead. Historical facility experiments, past operations, spills and leaks or material storage could have possibly caused cross-contamination of hazardous materials with radioactive material, which results in mixed waste. As a result, characterization included analysis of materials and systems to determine the potential presence of mixed low-level radioactive material.

The areas included in the characterization survey were divided into 17 survey packages. Each survey package contained instructions, drawings, and location codes to facilitate the collection of measurements and/or samples. Table 1.1 provides a brief overview of the results of the characterization surveys.

Table 1.1
Overview of Characterization Survey

Survey Package	Location Description	Survey Results
C-001	Reactor Building Inside Bioshield	Residual Activity Detected
C-002	Reactor Building Rad Material Storage Area	Residual Activity Detected
C-003	Demineralizer Building	Residual Activity Detected
	Maintenance Office	Residual Activity Not Detected
	Heat Exchanger Building	Residual Activity Not Detected
C-004	Waste Storage Tank Area	Residual Activity Detected
C-005	Building Addition 1 Counting Room	Residual Activity Detected
	Reactor Building Conference/Lunch Room	Residual Activity Not Detected
	Reactor Building Control Room	Residual Activity Not Detected
	Reactor Building Employee's Lockers	Residual Activity Not Detected
	Reactor Building General Manager's Office	Residual Activity Not Detected
	Reactor Building Machine Shop	Residual Activity Not Detected
	Reactor Building Office Supply Room	Residual Activity Not Detected
	Reactor Building South End Radiography	Residual Activity Not Detected
C-006	Tagging Building Safe	Residual Activity Not Detected
C-007	Reactor Mezzanine Preparation lab	Residual Activity Not Detected
	Reactor Mezzanine Chemical lab	Residual Activity Not Detected
	Reactor Mezzanine Instrument Calibration Area	Residual Activity Not Detected
	Reactor Mezzanine Electronics Lab	Residual Activity Not Detected
	Reactor Mezzanine Stairway	Residual Activity Not Detected
	Reactor Mezzanine N-Ray Gauge office	Residual Activity Not Detected
	Reactor Mezzanine Sheet Metal Fab Area	Residual Activity Not Detected
	Reactor Mezzanine Storage area	Residual Activity Detected
C-008	Building Addition 1 Office Space	Residual Activity Not Detected
	Building Addition 1 Customer Viewing Area	Residual Activity Not Detected
	Building Addition 1 Quality Control Room	Residual Activity Not Detected
	Building Addition 1 Dark Room	Residual Activity Not Detected
	Building Addition 1 Hallway	Residual Activity Not Detected
	Building Addition 1 Explosive Storage Safe	Residual Activity Not Detected
	Building Addition 1 Film Storage Room	Residual Activity Not Detected
	Building Addition 1 Shipping & Receiving	Residual Activity Not Detected
	Building Addition 1 N-Ray Setup Area	Residual Activity Not Detected
C-009	Reactor Building Men's Room	Residual Activity Not Detected
	Reactor Building Ladies' Room	Residual Activity Not Detected
	Reactor Building Business Office	Residual Activity Not Detected
	Reactor Building Accounting Office	Residual Activity Not Detected
C-010	Tagging Building Entry Vestibule	Residual Activity Not Detected
	Tagging Area	Residual Activity Not Detected

Survey Package	Location Description	Survey Results
	Tagging Area Back Room	Residual Activity Not Detected
	Storage Building	Residual Activity Not Detected
	Compressor Building	Residual Activity Not Detected
	Chemical Shed	Residual Activity Not Detected
C-011	Reactor Building Exterior Walls	Residual Activity Not Detected
	Building Addition 1 Exterior Walls	Residual Activity Not Detected
	Tagging Building Exterior Walls	Residual Activity Not Detected
	Storage Building Exterior Walls	Residual Activity Not Detected
C-012	All Other Buildings Exterior Walls	Residual Activity Not Detected
C-013	Parking Area Outside Fence	Residual Activity Not Detected
C-014	Paved Areas Inside Fence	Residual Activity Not Detected
C-015	Soil Areas Inside Fence	Residual Activity Not Detected
C-016	Main Cooling Tower	Residual Activity Not Detected
	Backup Cooling Tower	Residual Activity Not Detected
C-017	Waste Storage Tanks	Residual Activity Detected
C018	Soil Samples	Residual Activity Not Detected

Table 1.2 provides the bases used for identifying a survey unit in Table 1.1 as having residual activity.

Table 1.2
Basis for Residual Activity Designation

Survey Package Number	Location Description	Basis
C-001	Reactor Building Inside Bioshield	Elevated removable activity measurements on interior of shield walls, floor inside shielded area, wood roof top shield and top of a portable shield located on top of wood roof (Attachment D of characterization report). Decontamination was performed later.
C-002	Reactor Building Rad Material Storage Area	Elevated removable activity measurements on east wall and floor (Attachment D of characterization report). Floors was decontaminated later.
C-003	Demineralizer Building	Elevated removable activity measurements on outside of demin and on floor (Attachment D of characterization report). Demin and floors were decontaminated later.
C-004	Waste Storage Tank Area	Activity was detected in a sample of water and sediment obtained from the waste tank sump. Appendix A, results for Sample 03 Sump Pit.
C-005	Building Addition 1 Counting Room	Elevated removable activity measurements on top of the shielded "cave" used for counting samples with the gamma spectroscopy system (Attachment D of characterization report). The cave top was decontaminated later.
C-007	Mezzanine Storage Area	Elevated removable activity measurements on top of radioactive item storage table and on irradiated N-Ray Tube Cavity, (Attachment D of characterization report).
C-017	Waste Storage Tanks	Activity was detected in a sample of water and sediment obtained from the waste tank sump. Appendix A, results for Sample 03 Sump Pit.

While Tables 1.1 and 1.2 can be used to identify areas with residual activity, more detailed information is required to determine the magnitude and extent of the residual activity.

This detailed information is provided in the characterization report and in the report attachments and appendices. The areas identified as containing residual activity may not require remediation. Remediation requirements will be dependent on the criteria for release for unrestricted use, the magnitude of contamination, the extent of the contamination, the radionuclides present and their relative fractions. This characterization survey report should be reviewed in detail prior to determining if remediation may be required.

The results of the characterization survey can be used to support numerous activities associated with the preparation of the decommissioning plan. The results:

- Identify areas containing, or likely to contain, residual contamination.
- Identify the radionuclides of interest and their relative fractions.

- Provide information necessary to justify proposing the screening values in Tables B.1 and B.2 of Appendix B in NUREG 1757 as the criteria for release for unrestricted use.
- Provide information necessary to ensure the proper interpretation of survey results during remedial action and final status surveys.
- Provide information necessary to estimate the scope of the decommissioning.
- Provide information necessary to estimate waste volumes and allowable disposal options.
- Provide information necessary to assist in the classification of survey units in support of the final status survey.

The results of the characterization survey demonstrated that practices employed by Aerotest to minimize the spread of contamination were effective. In general contamination was confined to those areas and systems expected to be contaminated. Although the Nuclear Regulatory Commission has not approved the proposed criteria for release for unrestricted use, several of the areas surveyed appear to meet the proposed criteria for release for unrestricted use in their current state. Care should be taken to minimize the potential spread of contamination to these areas during future decommissioning activities.

REFERENCES FOR APPENDIX A {tc "REFERENCES FOR APPENDIX A " \l 2}

A-1 *Characterization Survey Report for the Aerotest Radiography & Research Reactor, San Ramon, California*, CS-HP-PR-008, Rev. 0, October 2011, Energy*Solutions*, LLC

APPENDIX B
ENVIRONMENTAL REPORT for DECOMMISSIONING ARRR

RADIOLOGICAL ACCIDENT ANALYSIS
for
DECOMMISSIONING
AEROTEST RADIOGRAPHY and RESEARCH REACTOR

Approvals Page

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1.0 PURPOSE AND OBJECTIVES

This calculation provides an analysis of the potential radiological accidents that could occur during decommissioning of the Aerotest Radiography and Research Reactor (ARRR) and affect the public or occupational health and safety. Prior to the initiation of decommissioning activities all reactor fuel will be removed from the ARRR. Therefore potential accidents involving reactor fuel were not considered. The accident analyses show that the doses to the public from potential accidents are below the U.S. EPA Protective Action Guides (PAGs) that have been developed to protect members of the public from the consequences of accidents (EPA 400-R-92-001, 1992). Therefore, no new protective measures are required to protect public or occupational health and safety. Bounding analyses of potential accidents at a level of detail consistent with existing information about the radiological hazards at ARRR were performed.

2.0 CONCLUSIONS

The accident analysis shows that the postulated accident scenarios would result in TEDEs to a member of the public at the site boundary that are much less than the U.S. EPA's lower PAG of 1 rem (1000 mrem) (USEPA 1992) and the NRC dose limits for individual members of the public of 0.1 rem/yr (100 mrem/yr) (10 CFR 20.1301).

The results of the accident analysis show that off-site consequences from accidents are well below the U.S. EPA's PAGs and the NRC's dose limits for individual members of the public; therefore, off-site emergency plans are not needed.

3.0 ASSUMPTIONS/INPUTS

3.1 ASSUMPTIONS

The following assumptions were used in the accident analyses:

- The radionuclide inventories were based on the data provided in the NUREG/CR-1756 (Ref. 4.4) and Characterization Report for the Aerotest Radiography & Research Reactor San Ramon, California October 2011, CS-HP-PR-008.
- To be conservative, unfavorable weather conditions for atmospheric dispersion were assumed. For the purposes of this analysis, atmospheric stability class F with a wind speed of 2 m/s (6.6 ft/s) was assumed, which represents a situation with minimal dispersion of a potential radioactive plume. In addition, the radioactive material was assumed to be released at ground level and to remain airborne as it travels downwind.
- A screening analysis approach was used for ARRR accident analysis because the radioactive inventories are very small compared to those

in operating reactors and in fuel cycle facilities subject to NRC regulation.

- The screening analysis for ARRR consists of identifying and analyzing plausible accident scenarios that could occur during decommissioning activities.

3.2 POTENTIAL RADIOLOGICAL ACCIDENTS

Identifying potential accident scenarios included evaluating ARRR areas that contain the highest inventories of radioactive material, describing energy sources and external events, reviewing proposed activities, and considering combinations of these elements that could lead to a release of radioactive material. Because of the limited inventory, the evaluation of accident scenarios conservatively assumed that no design or procedural controls would be available to prevent or mitigate accidental releases, even though such controls will be implemented during decommissioning activities. This assumption allows for a worst-case accident analysis to be performed.

3.3 HIGHEST RADIONUCLIDE INVENTORIES AT ARRR

One area with a high radiological material inventory at ARRR is the reactor pool. Most of the activity in the pool is contained in activated components that include control rods, instrument thimbles, etc. During decommissioning, these components will be cut mechanically underwater and lifted and placed into transport liners underwater. A waste shipping liner containing activated hardware was estimated to contain approximately 1,460 curies (NUREG/CR-1756-v1, Tables E1-2, E.1-3, E.1-5 and E.1-6, for Reference Research Reactor). Most of the activity is Co-60 (84%), with Fe-55 (8.6%), Mn-54 (4.7%), and smaller inventories of other radionuclides. The removal and shipping of activated components are simple operations, and the worst-case accident scenario would be dropping and breaching one of these filled transport liners as it is being lifted.

The demineralizer column for the reactor pool water has a concentrated radionuclide inventory. During decommissioning the resin column will be removed and placed into a transport container. The worst case accident scenario is the demineralizer column being dropped and broken open during handling. The estimated column activity is 78.7 millicuries. Most of the inventory is Sr-90 (38%), Cs-137 (24%), Ce-144 (15%), Cd-109 (10%), Eu-154 (4%), Fe-55 (1.5%), Eu-155(1.4%), Nb-95(1.1%), Pu-241 (1.1%), Zr-95 (1.0%), with other radionuclides less than 1%.

As part of the decommissioning process radioactive waste will be staged prior to loading it into shipping containers for transportation off-site. The accident scenario is a fire in the dry solid waste (DAW) for shipping (i.e., rags, wipes and anticontamination clothing). It was assumed that the activity levels in this material were 10% of activity in the resin column. This is very conservative as the resin column concentrated activity and most of the dry solid waste will have minimal contamination. The estimated DAW activity is 108 millicuries. The isotopic mixture is Sr-90 (38%), Cs-137 (24%), Ce-144 (15%), Cd-109 (10%), Eu-154 (4%), Fe-55 (1.5%), Eu-155(1.4%), Nb-95(1.1%), Pu-241 (1.1%), Zr-95 (1.0%), with other radionuclides less than 1%.

Another accident scenario considered is a fire in the graphite material related to the thermal column during its removal from the reactor. The estimated graphite activity is 1.41 curies. The calculated isotopic mixture is Eu-152 (93.3%), Eu-154 (6.5%), with other radionuclides less than 1%.

All of the other rooms at ARRR have smaller radioactive inventories than the reactor pool, waste staging area and the demineralizer system. Therefore, the results of accident analyses conducted for decommissioning the reactor pool and the Reactor Complex (including the demineralizer system and waste staging area) bound the potential impacts of other postulated indoor accidents during decommissioning of the ARRR.

Therefore, accident analyses were performed for the scenarios listed above because they bound the impacts of potential radiological accidents during the decommissioning of the ARRR.

4.0 REFERENCES

4.1 U.S. EPA, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, 400-R-92-001, 1992

4.2 USNRC, 10 CFR Part 20.1301, *Dose Limits for Individual Members of the Public*

4.3 10 CFR Part 20, *Standards for Protection Against Radiation*, Federal Register, Vol. 56, No. 88, NRC May 21, 1991

4.4 *Characterization Report for the Aerotest Radiography & Research Reactor San Ramon, California,* CS-HP-PR-008, October 2011

4.5 *Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors*, NUREG/CR-1756, March 1982

4.6 International Commission on Radiological Protection (ICRP), *Limits for Intakes of Radionuclides by Workers*, 1979, ICRP Publication No. 30, Annals of the ICRP Vol. 2, No. 3/4

4.7 *Federal Guidance Report No. 11*, EPA-520/1-88-020, September 1988

4.8 *Federal Guidance Report No. 12*, EPA-402-R-93-081, September 1993

4.9 NRC Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants

5.0 POTENTIAL ACCIDENT SCENARIOS

Considering the planned decommissioning activities, accident scenarios that could result in releasing radioactive material as airborne particles small enough to be respirable were evaluated. Such releases could occur during a fire in a waste storage area, as a result of dropping a resin column or as a result of dropping of a container of activated hardware. Because the ARRR is outside of the 100-year floodplain, extreme precipitation events are not expected to cause off-site radiological impacts.

Based on the decommissioning activities outlined in the ARRR Decommissioning Plan and the radiological inventories identified in Section 3.3, the following accident scenarios were evaluated:

- A waste shipping liner containing activated hardware could be dropped while moving it from the pool to a transportation cask and the liner could be breached.
- A demineralizer resin column could also be dropped, then burst open and release airborne particles.
- The potential for fires was also considered. A fire could occur in an area used to stage dry solid waste for shipping (i.e., rags, wipes and anticontamination clothing).
- A fire could occur in a block of activated graphite material.

6.0 METHODOLOGY FOR CALCULATING TOTAL EFFECTIVE DOSE EQUIVALENT

The consequences of accidents were quantified by calculating the TEDE to a member of the public at the site boundary. Then the calculated TEDE was compared to the U.S. EPA's lower PAG of 1 rem (1,000 mrem) and the NRC 0.1 rem/yr (100 mrem) dose limit for a member of the public, to determine whether or not the calculated exposure is acceptable without mitigation actions. Equation 6 - 1 was used to calculate the TEDE[1]:

$$TEDE_i = CEDE_i + Ext_i \qquad (6\text{-}1)$$

Where:

TEDE = total effective dose equivalent

CEDE = committed effective dose equivalent

Ext = contribution from external irradiation

i = radionuclide.

[1] This estimate of the TEDE neglects any contribution from gamma rays emitted by radionuclides deposited on the ground. Such doses build up relatively slowly and, if necessary, can be controlled by various countermeasures.

The committed effective dose equivalent (CEDE) is the dose contribution from inhalation as the cloud containing particulate radioactivity passes by the receptor. Consistent with the lung model developed by the International Commission on Radiological Protection (Ref. 4.6), the CEDE is found by:

$$CEDE_i = Q_i (x/Q) \times B \times D_i \times t \quad (6\text{-}2)$$

Where:

Q_i = the release rate of nuclide *i* / per unit time, in Ci/sec

T = the duration of the release, in sec. The product of Q_i x t is equal to the activity of nuclide *i* released. For the accident scenarios in this document there is a rapid release of activity, which is assumed to occur in one second.

x/Q = the atmospheric dispersion factor (concentration integrated over the duration of cloud passage) per unit activity released, in s/m^3. The derivation of x/Q presented in Appendix A shows that for a distance of 100 meters (0.06 mi) in atmospheric stability class F with a wind speed of 2 m/s, $x/Q = 4.52 \times 10^{-3}\ s/m^3$.

B = the breathing rate, typically $3.3 \times 10^{-4}\ m^3/s$. (This is the breathing rate for adults during light activity [Ref. 4.6]).

D_i = the dose conversion factor that converts the amount of activity inhaled into the CEDE. Values of D_i are given in Federal Guidance Report No. 11 (Ref. 4.7, EPA-520/1-88-020, September 1988).

The dose contribution from external irradiation is found by:

$$Ext_i = Q_i \times (x/Q) \times F_i \times t \quad (6\text{-}3)$$

Where:

F_i = the external dose coefficient for air submersion. Values of F_i are given in Federal Guidance Report No. 12 (Ref. 4.8, EPA-402-R-93-081, September 1993). Units are ([mrem/s]/[Ci/m^3]).

7.0 ANALYSIS OF POTENTIAL ACCIDENT SCENARIOS

7.1 SCENARIO 1: RELEASE WHILE MOVING IRRADIATED HARDWARE LINER

Most of the activity in the reactor pool is contained in activated components that include control rods, instrument tubes, etc. Because cutting operations for components will be performed underwater, no cutting accident releases were postulated. However a liner filled with irradiated hardware could be dropped while it is lifted for placement into a shipping container and the liner could be breached. The activity in irradiated hardware is contained within the metal structure of the hardware item except for surface contamination. It would be highly unlikely for a component in the liner to break. If it did break, the diameters of any particles produced would be large enough that it is unlikely that the particles would remain airborne and be respirable. However, even though it is not plausible that an accident could result in measurable exposures at the site boundary, this scenario was evaluated because it includes the largest curie inventory and it demonstrates that potential exposures to the public are acceptable even when worst case assumptions are utilized.

A waste shipping liner containing 120 cubic feet of activated hardware was estimated to contain 1,460 curies (Ref. 4.5, NUREG/CR-1756-v1, Tables E1-2, E.1-3, E.1-5 and E.1-6, for Reference Research Reactor). Most of the activity is Co-60 (84%), with Fe-55 (8.6%), Mn-54 (4.7%), and smaller inventories of other radionuclides. The worst-case accident scenario would be dropping the filled liner as it is being lifted. If 1 percent of the activity of this liner was respirable and 1 percent of the respirable material escaped the liner and became airborne during the accident (i.e., approximately 5.4×10^{2} g, assuming a waste density of 1.6 g/cm^3), the airborne quantities of radionuclides would be 1.22×10^{-01} Ci of Co-60, 1.25×10^{-02} Ci of Fe-55, 6.83×10^{-03} Ci of Mn-54, and lesser quantities of other radionuclides. Using the values of x/Q and B given in Section 6.0, assuming a 1 second release, and using Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-1.

Table 7-1. TEDE Calculation Table for Scenario 1: Hardware Liner Drop

Nuclide i	*Lung Clearance Class**	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s]/ [Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
C-14	Special	3.90E-06	2.09E+06	8.30E-04	4.33E-05	5.21E-11	4.33E-05
Mn-54	W	6.83E-03	6.70E+06	1.51E+02	2.43E-01	1.67E-02	2.60E-01
Fe-55	W	1.25E-02	1.34E+06	0.00E+00	8.84E-02	0.00E+00	8.84E-02
Co-60	Y	1.22E-01	2.19E+08	4.67E+02	1.42E+02	9.16E-01	1.42E+02
Ni-59	W	2.36E-05	9.18E+05	0.00E+00	1.15E-04	0.00E+00	1.15E-04
Ni-63	W	2.71E-03	2.30E+06	0.00E+00	3.31E-02	0.00E+00	3.31E-02
Zn-65	Y	1.82E-03	2.04E+07	1.03E+02	1.98E-01	3.04E-03	2.01E-01
Nb-93m	W	4.31E-09	2.95E+07	1.64E-02	6.77E-07	1.14E-12	6.77E-07
Nb-94	W	5.58E-08	4.41E+08	2.85E+02	1.23E-04	2.56E-07	1.23E-04
Total		**1.46E-01**	--	--	**1.42E+02**	**9.35E-01**	**1.43E+02**

* C-14 Special Model, Mn-54 oxides correspond to lung clearance Class W, Fe-55 oxides Class W, Nickel oxides Class W, and Zn-65 all compounds Class Y, Niobium oxides class W, and Co-60 oxides Class Y, from Federal Guidance Report # 11 (USEPA 1988), Table 3.

As shown in Table 7-1, the TEDE is 143 mrem, to which the external dose is a negligible contributor. The 143 mrem TEDE derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (Ref. 4.1, USEPA 1992). The TEDE was greater than the NRC normal operational annual dose limit for individual members of the public of 0.1 rem/yr (Ref. 4.2, 10 CFR 20.1301).

7.2 SCENARIO 2: RELEASE FROM A DROPPED DEMINERALIZER COLUMN

The reactor pool demineralizer has a concentrated radionuclide inventory. During the decommissioning, the resin column will be removed and placed into a transport container. The accident scenario is for the resin column to be dropped and broken open during handling. The activity is contained within the resin beads inside the demineralizer column except for surface contamination. If the column was dropped and broken the diameters of any particles produced would be large enough that it is unlikely that the particles would remain airborne and be respirable. However, even though it is not plausible that an accident could result in measurable exposures at the site boundary, this scenario was evaluated because it is one of the few areas with a significant curie inventory and it demonstrates that potential exposures to the public are acceptable even when worst case assumptions are utilized.

A demineralizer resin column containing 3.3 cubic feet of resin was estimated to contain 79 millicuries of activity based upon an independent laboratory analysis of the resin. Most of the inventory is Sr-90 (38%), Cs-137 (24%), Ce-144 (15%), Cd-109 (10%), Eu-154 (4%), Fe-55 (1.5%), Eu-155(1.4%), Nb-95(1.1%), Pu-241 (1.1%), Zr-95 (1.0%), with other radionuclides less than 1%. The worst-case accident scenario would be dropping the resin column liner as it is being lifted. If 1 percent of the activity of this column was respirable and 10 percent of the respirable material escaped the column and became airborne during the accident (i.e., approximately 74 g, assuming a waste density of 0.8 g/cm^3), the airborne quantities of radionuclides would be 2.97×10^{-05} Ci of Sr-90, 1.85×10^{-05} Ci of Cs-137, 1.21×10^{-05} Ci of Ce-144, 7.73×10^{-06} Ci of Cd-109, 3.24×10^{-06} Ci of Eu-154, 1.20×10^{-06} Ci of Fe-55, 1.08×10^{-06} Ci of Eu-155, 8.80×10^{-07} Ci of Nb-95, 8.65×10^{-07} Ci of Pu-241, 7.87×10^{-07} Ci of Zr-95, and lesser quantities of other radionuclides. Using the values of x/Q and B given in Section 6.0 and Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-2.

Table 7-2. TEDE Calculation Table for Scenario 2: Dropped Demineralizer Column

Nuclide *i*	*Lung Clearance Class**	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
C-14	Special	2.03E-08	2.09E+06	8.30E-04	2.25E-07	2.71E-13	2.25E-07
Na-22	D	1.09E-06	7.66E+06	4.00E+02	4.44E-05	7.03E-06	5.15E-05
Fe-55	W	1.20E-06	1.34E+06	0.00E+00	8.49E-06	0.00E+00	8.49E-06
Co-60	Y	1.69E-07	2.19E+08	4.67E+02	1.97E-04	1.27E-06	1.98E-04
Ni-63	W	1.77E-07	2.30E+06	0.00E+00	2.17E-06	0.00E+00	2.17E-06
Sr-90	D	2.97E-05	2.39E+08	2.79E-02	3.78E-02	1.33E-08	3.78E-02
Nb-95	Y	8.80E-07	5.81E+06	1.39E+02	2.72E-05	1.96E-06	2.91E-05
Zr-95	W	7.87E-07	1.59E+07	1.33E+02	6.64E-05	1.69E-06	6.81E-05
Cd-109	Y	7.73E-06	4.51E+07	1.09E+00	1.86E-03	1.36E-07	1.86E-03
Cs-134	D	3.24E-07	4.63E+07	2.80E+02	7.96E-05	1.46E-06	8.10E-05
Cs-137	D	1.85E-05	3.19E+07	2.87E-02	3.14E-03	8.53E-09	3.14E-03
Ce-144	Y	1.21E-05	4.07E+08	3.16E+00	2.62E-02	6.16E-07	2.62E-02
Eu-152	W	5.39E-07	2.21E+08	2.09E+02	6.32E-04	1.82E-06	6.34E-04
Eu-154	W	3.24E-06	2.86E+08	2.27E+02	4.93E-03	1.19E-05	4.94E-03
Eu-155	W	1.08E-06	4.14E+07	9.22E+00	2.37E-04	1.60E-07	2.37E-04
Pu-238	Y	2.85E-08	2.88E+11	1.81E-02	4.36E-02	8.29E-12	4.36E-02
Pu-239	Y	4.55E-08	3.27E+11	1.57E-02	7.90E-02	1.15E-11	7.90E-02
Pu-241	Y	8.65E-07	4.96E+09	2.69E-04	2.28E-02	3.74E-12	2.28E-02
Am-241	W	2.35E-07	4.44E+11	3.03E+00	5.54E-01	1.15E-08	5.54E-01
Cm-242	W	3.67E-08	1.73E+10	2.11E-02	3.37E-03	1.25E-11	3.37E-03
Total		**7.87E-05**	--	--	**7.78E-01**	**2.80E-05**	**7.78E-01**

* C-14 Special Model, Na-22 all forms correspond to lung clearance Class D, Iron oxides Class W, Cobalt oxides Class Y, Nickel oxides Class W, Strontium (all but $SrTiO_3$) Class D, Niobium oxides and hydroxides, Zirconium oxides, hydroxides, halides and nitrates Class Y, Cadmium oxides and hydroxides Class Y, Cesium all forms Class D, Cerium oxides, hydroxides and fluorides Class Y, Europium all forms Class W, Plutonium oxides Class Y, Americium all forms Class W, and Curium all forms Class W, from Federal Guidance Report # 11 (USEPA 1988), Table 3.

As shown in Table 7-2, the TEDE is 0.78 mrem, to which the external dose is a negligible contributor. The 0.78 mrem TEDE derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (Ref. 4.1, USEPA 1992). The TEDE was less than the NRC normal operational annual dose limit for individual members of the public of 0.1 rem/yr (Ref. 4.2, 10 CFR 20.1301).

7.3 SCENARIO 3: RELEASE FROM A WASTE STAGING AREA FIRE

As part of the decommissioning process there will be radioactive waste staged prior to loading it into shipping containers for transportation off-site. The accident scenario is a fire in the dry solid waste for shipping (i.e., rags, wipes and anticontamination clothing). It was assumed that the activity concentration in this material is equal to 10% of the concentration for the resin column. This is very conservative as the resin column is concentrated activity and most of the dry solid waste will have minimal contamination. A waste inventory of 360 cubic feet of dry solid waste was estimated to contain 0.11 millicuries, assuming it would have the same activity distribution as the resin in Scenario 2. The calculated inventory contains Sr-90 (38%), Cs-137 (24%), Ce-144 (15%), Cd-109 (10%), Eu-154 (4%), Fe-55 (1.5%), Eu-155(1.4%), Nb-95(1.1%), Pu-241 (1.1%), Zr-95 (1.0%), with other radionuclides less than 1%.

It is estimated that combustion of this material would release approximately 25% of the contamination in a respirable form. The combustion of this waste under this scenario would release 5.10×10^{-03} Ci of Sr-90, 3.18×10^{-03} Ci of Cs-137, 2.08×10^{-03} Ci of Ce-144, 1.33×10^{-03} Ci of Cd-109, 5.57×10^{-04} Ci of Eu-154, 2.05×10^{-04} Ci of Fe-55, 1.85×10^{-04} Ci of Eu-155, 1.51×10^{-04} Ci of Nb-95, 1.49×10^{-04} Ci of Pu-241, 1.35×10^{-04} Ci of Zr-95, and lesser quantities of other radionuclides. Using the values of x/Q and B given in Section 6.0 and Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-3.

Table 7-3. TEDE Calculation Table for Scenario 3: Waste Staging Area Fire

Nuclide i	*Lung Clearance Class*	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
C-14	Special	3.48E-06	2.09E+06	8.30E-04	3.33E-05	4.02E-11	3.33E-05
Na-22	D	1.87E-04	7.66E+06	4.00E+02	6.59E-03	1.04E-03	7.63E-03
Fe-55	W	2.05E-04	1.34E+06	0.00E+00	1.26E-03	0.00E+00	1.26E-03
Co-60	Y	2.91E-05	2.19E+08	4.67E+02	2.92E-02	1.89E-04	2.94E-02
Ni-63	W	3.04E-05	2.30E+06	0.00E+00	3.21E-04	0.00E+00	3.21E-04
Sr-90	D	5.10E-03	2.39E+08	2.79E-02	5.60E+00	1.98E-06	5.60E+00
Nb-95	Y	1.51E-04	5.81E+06	1.39E+02	4.03E-03	2.91E-04	4.32E-03
Zr-95	W	1.35E-04	1.59E+07	1.33E+02	9.85E-03	2.51E-04	1.01E-02
Cd-109	Y	1.33E-03	4.51E+07	1.09E+00	2.75E-01	2.01E-05	2.75E-01
Cs-134	D	5.56E-05	4.63E+07	2.80E+02	1.18E-02	2.17E-04	1.20E-02
Cs-137	D	3.18E-03	3.19E+07	2.87E-02	4.65E-01	1.27E-06	4.65E-01
Ce-144	Y	2.08E-03	4.07E+08	3.16E+00	3.88E+00	9.13E-05	3.88E+00
Eu-152	W	9.25E-05	2.21E+08	2.09E+02	9.38E-02	2.69E-04	9.41E-02
Eu-154	W	5.57E-04	2.86E+08	2.27E+02	7.31E-01	1.76E-03	7.32E-01
Eu-155	W	1.85E-04	4.14E+07	9.22E+00	3.52E-02	2.37E-05	3.52E-02
Pu-238	Y	4.89E-06	2.88E+11	1.81E-02	6.47E+00	1.23E-09	6.47E+00
Pu-239	Y	7.81E-06	3.27E+11	1.57E-02	1.17E+01	1.71E-09	1.17E+01
Pu-241	Y	1.49E-04	4.96E+09	2.69E-04	3.38E+00	5.55E-10	3.38E+00
Am-241	W	4.04E-05	4.44E+11	3.03E+00	8.22E+01	1.70E-06	8.22E+01
Cm-242	W	6.31E-06	1.73E+10	2.11E-02	5.01E-01	1.85E-09	5.01E-01
Total		**1.15E-02**	--	--	**1.15E+02**	**4.16E-03**	**1.15E+02**

* C-14 Special Model, Na-22 all forms correspond to lung clearance Class D, Iron oxides Class W, Cobalt oxides Class Y, Nickel oxides Class W, Strontium (all but $SrTiO_3$) Class D, Niobium oxides and hydroxides, Zirconium oxides, hydroxides, halides and nitrates Class Y, Cadmium oxides and hydroxides Class Y, Cesium all forms Class D, Cerium oxides, hydroxides and fluorides Class Y, Europium all forms Class W, Plutonium oxides Class Y, Americium all forms Class W, and Curium all forms Class W, from Federal Guidance Report # 11 (USEPA 1988), Table 3.

As shown in Table 7-3, the TEDE is 115 mrem, to which the external dose is a negligible contributor. The 115 mrem TEDE derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (Ref. 4.1, USEPA 1992). The TEDE was greater than the NRC normal operational annual dose limit for individual members of the public of 0.1 rem/yr (Ref. 4.2, 10 CFR 20.1301).

7.4 SCENARIO 4: RELEASE FROM A GRAPHITE BLOCK FIRE

As part of the decommissioning process there will be activated graphite removed. The accident scenario is a fire in the graphite material. The graphite is approximately 2-feet by 2-feet by 4-feet and it was assumed to ignite and produce a fire even though it is currently contained in aluminum. The graphite volume of 16 cubic feet was estimated to contain 1.41 Ci based upon an independent activation analysis of this graphite material (Ref. 4.4). The calculated inventory is Eu-152 (93.3%), Eu-154 (6.5%) with other radionuclides less than 1%.

It is estimated that combustion of this structural material would release approximately 25% of the contamination in a respirable form. The combustion of this waste under this scenario would release 1.64×10^{-01} Ci of Eu-152, 1.14×10^{-02} Ci of Eu-154 and lesser quantities of other radionuclides. Using the values of x/Q and B given in Section 6.0 in Equations 6-1 through 6-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 7-4.

Table 7-4. TEDE Calculation Table for Scenario 4: Graphite Block Fire

Nuclide *i*	*Lung Clearance Class*	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s] /[Ci/m^3])	CEDE$_i$ (mrem)	Ext$_i$ (mrem)	TEDE$_i$ (mrem)
C-14	Special	9.33E	2.09E+06	8.30E-04	1.03E-03	1.25E-09	1.03E-03
S-35	D	8.73E	3.02E+05	9.00E-04	1.40E-10	1.27E-15	1.40E-10
Cl-36	D	1.23E	2.24E+06	8.26E-02	1.46E-05	1.63E-09	1.46E-05
Ca-41	W	5.63E	1.35E+06	0.00E+00	4.03E-05	0.00E+00	4.03E-05
Ca-45	W	6.23E	6.62E+06	3.20E-03	2.19E-05	3.21E-11	2.19E-05
Mn-54	W	1.05E	6.70E+06	1.51E+02	3.75E-07	2.57E-08	4.01E-07
Fe-55	W	8.56E	1.34E+06	0.00E+00	6.08E-04	0.00E+00	6.08E-04
Co-60	Y	1.60E	2.19E+08	4.67E+02	1.86E-04	1.20E-06	1.87E-04
Ni-63	W	1.04E	2.30E+06	0.00E+00	1.27E-06	0.00E+00	1.27E-06
Eu-152	W	1.42E	2.21E+08	2.09E+02	1.66E+02	4.77E-01	1.67E+02
Eu-154	W	9.88E	2.86E+08	2.27E+02	1.50E+01	3.62E-02	1.51E+01
Eu-155	W	6.86E	4.14E+07	9.22E+00	1.51E-02	1.02E-05	1.51E-02
Gd-153	W	7.76E	9.77E+05	1.37E+01	4.03E-05	1.72E-06	4.20E-05
Total		**1.52E**	--	--	**1.81E+02**	**5.13E-01**	**1.82E+02**

* C-14 Special Model, Sulfur Sulfates & sulfides Class D, Chlorine (see assignment of associated element) Class D, Calcium all forms Class W, Manganese Oxides, hydroxides, halides & nitrates Class W, Iron oxides Class W, Cobalt oxides Class Y, Nickel oxides Class W, Europium all forms Class W, and Gadolinium oxides, hydroxides & fluorides Class W, from Federal Guidance Report # 11 (USEPA 1988), Table 3.

As shown in Table 7-4, the TEDE is 182 mrem, to which the external dose is a negligible contributor. The 182 mrem TEDE derived using conservative conditions is less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (Ref. 4.1, USEPA 1992). The TEDE was greater than the NRC normal operational annual dose limit for individual members of the public of 0.1 rem/yr (Ref. 4.2, 10 CFR 20.1301).

8.0 CONCLUSIONS

The accident analysis shows that the postulated accident scenarios would result in TEDE's to the nearest member of the public that are less than the U.S. EPA's lower PAG of 1 rem (1,000 mrem) (Ref. 4.1, USEPA 1992) but for some scenarios, somewhat larger than the NRC normal operational annual dose limits for individual members of the public of 0.1 rem/yr (100 mrem/yr) (Ref. 4.2, 10 CFR 20.1301).

The results of the accident analysis show that off-site consequences from accidents are well below the U.S. EPA's PAGs; therefore, off-site emergency plans are not needed.

9.0 APPENDICES

Appendix A DERIVATION OF χ/Q

APPENDIX A

DERIVATION OF χ/Q

In the accident analysis presented in Section 6.0, the quantity χ /Q is used to express the dilution of the released effluent as it travels 50 meters (0.03 mi) to the nearest neighbor. χ/Q is calculated using the well-established formula for Gaussian Dispersion, which is applicable when the effluent is released at such a rate that it does not perturb the existing pattern of turbulent eddies in the atmosphere. This is the expected case for small releases such as are evaluated in Section 7.0. χ/Q was calculated using the formula:

$$\frac{\chi}{Q} = \frac{1}{U_{10}\pi\Sigma_y\sigma_z} \qquad (9\text{-}1)$$

where $\Sigma_y = M\Phi_y$, for distances of 800 meters or less and the value of M is determined from Figure 3 of NRC Regulatory Guide 1.145 (M=1 for all cases where the wind speed is 6 meters per second or more). For a wind speed of 2 m/s and atmospheric stability class F, M=4.

Then for a wind speed of 2 m/s: $\frac{\chi}{Q} = \frac{1}{U_{10}\pi M\sigma_y\sigma_z}$ (9-2)

where:

σ_y = the lateral plume spread in meters (m)

σ_z = the vertical plume spread in meters (m)

U_{10} = the wind speed (meters/second) measured at a height of 10 meters.

The value of Φ_y is determined from Figure 1 of NRC Regulatory Guide 1.145 and Φ_z from Figure 2 of NRC Regulatory Guide 1.145. The values are a function of the distance, d, from the source and Pasquill's turbulence types. For ARRR a distance of 50 meters (164 feet) and category F (Moderately Stable) meteorological conditions were utilized. Using d = 50 meters for category F meteorological conditions yielded Φ_y = 1.9 meters and Φ_z = 1.3 meters. Using these values of Φ_z and Φ_y and a wind speed, u, of 2 m/s, Equation 9-2 yields

$$\frac{\chi}{Q} = \frac{1}{(2)\pi(4)(1.9m)(1.3m)} = 1.61\times10^{-2}\ \sec/m^3$$

ATTACHMENT (9)

DECOMMISSIONING FINANCIAL ASSURANCE PROJECTION

Aerotest Operations, Inc.
May 30, 2012

ATTACHMENT (9)

DECOMMISSIONING FINANCIAL ASSURANCE PROJECTION

	1 year	Qtr Payments 3-year	5-Year	7-Year	9-Year
Autoliv Cash Contribution 3,285,800	$3,350,343	$3,483,257	$3,621,445	$3,765,115	$3,914,485
Aerotest Contribution (Qtr payments until end of 5th year)	$105,658	$323,240	$549,454	$571,251	$593,914
	$3,456,001	$3,806,497	$4,170,899	$4,336,366	$4,508,399
Decommissioning costs escalated by 3%			($3,809,143)		($4,287,224)
		Difference	$361,756		$221,175

ATTACHMENT (10)

CONFORMING AMENDMENTS

Aerotest Operations, Inc.
May 30, 2012

AEROTEST OPERATIONS, INC.

DOCKET NO. 50- 228

AEROTEST RADIOGRAPHY AND RESEACH REACTOR (ARRR)

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
License No. R-98

1. The Nuclear Regulatory Commission (NRC or the Commission), having previously made the findings set forth in License No. R-98 issued on January 28, 1981, has now found that:

A. The application for indirect transfer of license filed by Aerotest Operations, Inc., ~~X-Ray Industries, Inc., and Autoliv ASP, Inc.~~ Nuclear Labyrinth LLC, for Aerotest Operations, Inc. (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

B. Construction of the facility has been substantially completed in conformity with Construction Permit No. CPRR-86, and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

E. Aerotest Operations, Inc. is technically and financially qualified to possess, use, and operate the facility in accordance with the rules and regulations of the Commission;

F. The issuance of this operating license will not be inimical to the common defense and security or to the health and safety of the public, and does not involve a significant hazards consideration;

G. The receipt, possession, and use of byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including Sections 30.33, 70.23, and 70.31;

H. The licensee is qualified to be the holder of the license; and

I. The transfer of the license is otherwise consistent with applicable provisions of law, regulations, and orders issued by the Commission pursuant thereto.

2. Facility Operating License No. R-98, Aerotest Operations, Inc., previously owned by OEA, Inc., is hereby being indirectly transferred to Aerotest Operations, Inc., owned by ~~Aerotest Holdings, LLC, a wholly owned subsidiary of X-Ray Industries, Inc.~~ Nuclear Labyrinth LLC, and the license reads as follows:

A. This license applies to the Aerotest Radiography and Research Reactor (ARRR), held by Aerotest Operations, Inc., ~~a pool-type nuclear reactor~~ owned by ~~Aerotest Holdings, LLC, a wholly owned subsidiary of X-Ray Industries, Inc.~~ Nuclear Labyrinth LLC The facility is located at the Aerotest Operations site near San Ramon, California, and is described in the application dated September 14, 1964 (the application), and in supplements thereto, including the applications for transfer of license dated April 24, 1974, and application for indirect transfer dated ~~January 19, 2010.~~ May 30, 2012.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Aerotest Operations, Inc.:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess, use, and operate the reactor at the designated location in San Ramon, California, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material", to receive, possess, and use up to 5.0 kilograms of contained uranium 235 in connection with operation of the reactor; and

(3) Pursuant to the Act and 10 CFR Part 30, "Licensing of Byproduct Material", (1) to receive, possess, and use a 2 curie americium-beryllium neutron startup source, and (2) to possess, but not to separate, such byproduct material as may be produced by operation of the reactor.

Amendment No. 2

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state power levels not in excess of 250 kilowatts (thermal).

(2) Technical Specifications

The Technical Specifications hereby contained in Appendix A, as revised through Amendment No. 1 , are ~~herby~~ incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Security Plan

The licensee shall maintain in effect and fully implement all provisions of the NRC-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR Section 50.54(p). The approved security plan consists of the document withheld from public disclosure pursuant to 10 CFR 2.790(d), entitled "Aerotest Operations, Inc. Security Plan" dated August 10, 1976, submitted by letter dated October 4, 1976, as revised ~~January 16, 1979.~~ February 28, 2005.

D. Reports

In addition to reports otherwise required under the license and applicable regulations:

(1) The licensee shall report in writing to the Commission within 10 days of its observed occurrence any incident or condition relating to the operation of the facility which prevented or could have prevented a nuclear system from performing its safety function as described in the Technical Specifications or in the Hazards Summary Report.

(2) The licensee shall report to the Commission in writing within 30 days of its observed occurrence any substantial variance disclosed by operation of the facility from performance specifications contained in the Hazards Summary Report or the Technical Specifications.

Amendment No. 2

11.0 Fuel Storage and Transfer

11.1 The fuel storage pits located in the floor of the reactor room shall accommodate a maximum of 19 fuel elements (700 gm U-235) in storage racks dry or flooded with water. The fuel storage pits shall be secured with a lock and chain except during fuel transfer operations.

11.2 Additional fuel storage racks may be located in the reactor tank. Each of these storage facilities shall be so designed that for all conditions of moderation k eff shall not exceed a value of 0.8.

11.3 A fuel handling tool shall be used in transferring fuel elements of low radioactivity between the storage pits and the reactor; a shielded fuel transfer of highly radio-active fuel elements. The fuel handling tool shall remain in a locked cabinet under the cognizance of the Reactor Supervisor when not authorized for use.

11.4 All fuel transfers in the reactor tank shall be conducted by a minimum staff of three men, and shall include a licensed Senior Operator and a licensed Operator. The staff members shall monitor the operation using appropriate radiation monitoring instrumentation. Fuel transfers outside the reactor tank but within the facility shall be supervised by a licensed Operator.

11.5 Not more than one fuel element shall be allowed in the facility which is not in storage or in the core lattice.

12.0 Administrative Requirements

12.1 Organization

12.1.1 The Reactor Supervisor shall have responsibility of the reactor facility. In all matters pertaining to reactor operations and to these Technical Specifications, the Reactor Supervisor shall be responsible to the President of the Licensee. The President of the Licensee shall report to the Board of Directors of ^the Licensee.

12.1.2 The Radiological Safety Officer shall review and approve all procedures and experiments involving radiological safety. He shall enforce rules, regulations and procedures relating to radiological safety, conduct routine radiation surveys and is responsible to the Manager, Aerotest Operations.

12.1.3 The Reactor Safeguards Committee shall be composed of not less than five members, of whom no more than three are members of the operating organization. The committee

ATTACHMENT (11)

REQUEST FOR WITHHOLDING PROPRIETARY INFORMATION

Aerotest Operations, Inc.
May 30, 2012

License No. R-98
May 30, 2012

Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Aerotest Radiography and Research Reactor
Docket No. 50-228
Request for Withholding of Proprietary Information related to Aerotest Radiography and Research Reactor

Ladies and Gentlemen:

Pursuant to 10 CFR 2.390. Public Exemptions, Request for Withholding, Aerotest Operations Inc. (Aerotest) and Nuclear Labyrinth, LLC (Nuclear Labyrinth) hereby request that information contained in their Application for Approval of Indirect Transfer of Control of License Pursuant to 10 C.F.R. § 50.80 dated May 30, 2012 (the "Application") be withheld from public disclosure as such application contains trade secrets, commercial information, and financial information which is Business Proprietary.

In support of our request we state the following:

1. The information sought to be withheld from public disclosure are sections of the Application for Approval of Indirect Transfer of Control of License Pursuant to 10 C.F.R. § 50.80 dated May 30, 2012 submitted by Aerotest Operations Inc. and Nuclear Labyrinth marked as Business Confidential including Attachments (6) and (7).
2. The persons making this request are Dario Brisighella of Aerotest and Dr. David M. Slaughter of Nuclear Labyrinth.
3. The basis for proposing that the information be withheld from public disclosure is that information contained in the Application includes confidential business and financial information of Aerotest and Nuclear Labyrinth, which information has been held in confidence by Aerotest and Nuclear Labyrinth, is a type customarily held in confidence, is not available in public sources, and if publicly disclosed would be likely to cause substantial harm to the competitive position of Aerotest and Nuclear Labyrinth taking into account the value of the information, the

amount of effort and money expended by Aerotest and Nuclear Labyrinth in developing the information and the difficult with which the information could be acquired or duplicated by others;

4. If such information was disclosed to the public, Aerotest and Nuclear Labyrinth believe that competitive harm would result;
5. The Application has been marked to show locations of the information sought to be withheld.

Dario Brisighella, being dully sworn, states that he is President of Aerotest, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this request for the Aerotest Radiography and Research Reactor, and that all the matter and facts set forth herein are true and correct to the best of his knowledge.

Dario Brisighella
President

Subscribed and sworn to before me, a Notary Public, in and for the county and state above named, this 25 day of May 2012.

Notary Public in and for the State of Utah

My Commission Expires: 10/20/2012

TERRI SCOVILL
Notary Public, State of Utah
Commission # 575822
My Commission Expires
October 20, 2012

David M. Slaughter, PhD, being dully sworn, states that he is Chief Executive Officer, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this request for Nuclear Labyrinth, and that all the matter and facts set forth herein are true and correct to the best of his knowledge.

David M. Slaughter, PhD

Subscribed and sworn to before me, a Notary Public, in and for the county and state above named, this 25 day of May 2012.

Notary Public in and for the State of Utah

My Commission Expires: 10/20/2012

TERRI SCOVILL
Notary Public, State of Utah
Commission # 575822
My Commission Expires
October 20, 2012

Aerotest Operations, Inc.
May 30, 2012