

June 24, 2003

Dr. Nolan Hertel, Director  
Neely Nuclear Research Center  
Georgia Institute of Technology  
900 Atlantic Drive  
Atlanta, GA 30332-0425

SUBJECT: NRC INSPECTION REPORT NO. 50-160/2002-201

Dear Dr. Hertel:

The inspection effort involved the coordination of the confirmatory radiological survey activities performed by our contractor, Oak Ridge Institute for Science and Education, of your research reactor on October 21-23, 2002. In addition, various aspects of your reactor operations, decommissioning, and radiation protection programs were inspected, including selective examinations of procedures and representative records, interviews with personnel, and observations of the facility.

Based on the results of this inspection, it has been determined that: 1) the decommissioning of the 5 MWt Research Reactor has been performed in accordance with the approved Decommissioning Plan; 2) the terminal radiation survey and associated documentation from the licensee demonstrated that residual radioactive material at the facility and site is less than the NRC-approved guideline limits; and 3) since the licensee has met their NRC-approved guideline limits, the facility and site meet the criteria for license termination set forth in 10 CFR Part 20.1401(b)(2).

No safety concern or noncompliance with Nuclear Regulatory Commission (NRC) requirements was identified. No response to this letter is required.

Dr. N. Hertel

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In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at (the Public Electronic Reading Room) <http://www.nrc.gov/NRC/ADAMS/index.html>. Should you have any questions concerning this inspection, please contact Mr. Stephen Holmes at 301-415-8583.

Sincerely,

***/RA by Daniel E. Hughes, Acting for/***

Patrick M. Madden, Section Chief  
Research and Test Reactors Section  
Operating Reactor Improvements Program  
Division of Regulatory Improvement Programs  
Office of Nuclear Reactor Regulation

Docket No. 50-160  
License No. R-97

Enclosures: 1. NRC Inspection Report No. 50-160/2002-201  
2. Confirmatory Survey Plan for the Georgia Tech Research Reactor dated October 9, 2002  
3. Confirmatory Survey of the Georgia Tech Research Reactor, dated February 2003

cc w/enclosures: Please see next page

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-2-

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U. S. NUCLEAR REGULATORY COMMISSION

Docket No: 50-160

License No: R-97

Report No: 50-160/2002-201

Licensee: Georgia Institute of Technology

Facility: Georgia Institute of Technology Research Reactor (GTRR)

Location: 900 Atlantic Drive  
Atlanta, GA 30332

Dates: October 21-23, 2002

Inspector: Stephen W. Holmes

Approved by: Patrick M. Madden, Section Chief  
Research and Test Reactors Section  
Operating Reactor Improvements Program  
Division of Regulatory Improvement Programs  
Office of Nuclear Reactor Regulation

## EXECUTIVE SUMMARY

### Georgia Institute of Technology Research Reactor Report No: 50-160/2002-201

This routine, announced inspection involved the confirmatory radiological survey and the on-site review of selected activities being performed at the Georgia Institute of Technology Research Reactor. In addition, the activities audited during this inspection included: organization and staffing; review and audit functions; procedures; removal of materials; decommissioning activities; release criteria; confirmatory final survey; maintenance and surveillance; and radiation protection program. The inspector was assisted by the NRC's contractor, Oak Ridge Institute for Science and Education Environmental Survey and Site Assessment Program.

#### Organization and Staffing

- The organizational structure and their corresponding functions were consistent with Technical Specification Section 5.0, Amendment No. 14, dated July 22, 1999, and the Decommissioning Plan for the Georgia Institute of Technology Research Reactor facility dated June 1998.

#### Review and Audit Functions

- The audits conducted by the Technical Safety Review Committee and Georgia Institute of Technology Research Reactor staff were in accordance with the requirements specified in Technical Specification Section 5.2. and Decommissioning Plan Section 2.4.

#### Procedures

- The procedural control and implementation program was acceptably maintained and met Technical Specifications and Decommissioning Plan requirements.

#### Removal of Materials

- Fuel and radioactive and non-radioactive waste was removed from the site in accordance with the Georgia Institute of Technology Research Reactor Decommissioning Plan requirements, and Department of Transportation and Nuclear Regulatory Commission regulations.

#### Decommissioning Activities

- Decommissioning activities were performed as required by Decommissioning Plan Section 2.3 and licensee procedures.

#### Release Criteria

- Duratec used the appropriate guideline and screening values, as required by the NRC-approved Decommissioning Plan, in performing the final survey.

### Confirmatory Final Survey

- The elevated surface activity and exposure readings in the basement compressor room were due to naturally occurring radioactive material.
- Based on the results of the licensee's final status survey and Nuclear Regulatory Commission's confirmatory measurements, Georgia Institute of Technology has adequately demonstrated that the Georgia Institute of Technology Research Reactor facility satisfies the criteria for release for unrestricted use.

### Maintenance and Surveillance

- The maintenance program was implemented as required by Georgia Institute of Technology procedures.
- The licensee's program for surveillance and limiting conditions for operation confirmations satisfied Technical Specification and Decommissioning Plan requirements.
- The licensee's design change procedures were in place and were implemented as required by licensee procedures.

### Radiation Protection Program

- The radiation protection program satisfied the requirements of 10 CFR 19.12 and 10 CFR Part 20.1101.
- Radiological postings satisfied regulatory requirements.
- Surveys were performed and documented as required by 10 CFR 20.1501(a), Technical Specifications, and licensee procedures.
- The personnel dosimetry program was acceptably implemented and doses were in conformance with licensee and 10 CFR Part 20 limits.
- Portable survey meters, radiation monitoring, and counting lab instruments were maintained according to Technical Specifications, industry/equipment manufacturer standards, and licensee and contractor procedures.
- The evaluation and administration of the respiratory program were adequately performed according to Decommissioning Plan and Nuclear Regulatory Commission requirements.
- The program for monitoring, storage, and release of effluents was acceptable.



## Report Details

### Summary of Plant Status

Georgia Institute of Technology (GIT), in Atlanta Georgia, has completed decommissioning its 5 MWt Research Reactor (GTRR) and associated systems. The reactor was located within the Neely Nuclear Research Center (NNRC) on GIT's main campus. The reactor was designed for several different research applications including experiments using high intensity neutron beams, gamma ray beams, and an uniform thermal neutron flux through a large sized beam. Although it was originally designed for 1 MWt output, it was upgraded to produce 5 MWt in 1974. The GTRR was built in the early 1960's as a research and training reactor. Operating under the Nuclear Regulatory Commission (NRC) License No. R-97, it went critical for the first time on December 31, 1964.

On November 17, 1995, all operations at the reactor ceased. GIT contracted NES, Inc. to perform the initial characterization survey and to provide a decommissioning plan for the GTRR. In October 1997, NES performed a characterization survey of the GTRR, based upon the GIT Decommissioning Project - Radiological Characterization Plan. Results of the characterization survey were provided in NES' "Georgia Institute of Technology Research Reactor Decommissioning Project Characterization Report" issued May 1998. GIT requested the NRC, by letters dated July 1, 1998, February 8, 1999, and May 28, 1999, to grant them the authorization to decommission the reactor according to their submitted decommissioning plan. On July 22, 1999, the NRC issued Amendment No. 14 to the reactor licence that approved GIT's Decommissioning Plan. GIT contracted with IT Corporation (IT) to decommission the GTRR facility. IT, through its subcontractor GTS Duratec (Duratec), started decommissioning operations December 1999. Final waste shipment was made August 2001.

The Final Status Survey Report for the GTRR facility was completed and issued June 2002. According to the report, all contaminated systems and components had been removed from the site. Potentially contaminated structural surfaces identified during characterization surveys had been removed and/or remediated such that the residual radioactivity is less than NRC Regulatory Guide 1.86 limits.

The NRC requested Oak Ridge Institute for Science and Education's (ORISE) Environmental Survey and Site Assessment Program (ESSAP) to perform a confirmatory survey of the GTRR facility. On October 21-23, 2002, the ESSAP team, accompanied by an NRC inspector, conducted this survey.

### **1. ORGANIZATIONAL STRUCTURE AND FUNCTIONS**

#### **a. Inspection Scope (Inspection Procedures (IP) 69001 and 40755)**

The inspector reviewed selected aspects of:

- organization and staffing
- qualifications
- management responsibilities
- administrative controls
- decommissioning activity records
- GTRR Decommissioning Plan (DP) dated June 1998

- Technical Specifications (TS), Amendment No. 14, dated July 22, 1999

b. Observations and Findings

The general organizational structure and staffing had not changed since the last inspection. The organizational structure and staffing at the facility were as reported in the Annual Report and as required by TS Section 5.1 and Figure 5.1. Review of records verified that management responsibilities were administered as required by TS Sections 5.2 thru 5.6 and applicable procedures.

The decommissioning of the reactor required GTRR management to assume additional project management responsibilities. Through record reviews and interviews with the reactor manager, radiation safety officer (RSO), and Duratec project manager, the inspector confirmed that both GTRR management and the decommissioning project organization structures were as required by DP Section 2.4 and Figure 2.2.

c. Conclusions

The organizational staff and their corresponding functions and responsibilities were consistent with TS Section 5.0, Amendment No. 14, dated July 22, 1999, and the DP for the GTRR facility dated June 1998

**2. REVIEW AND AUDIT FUNCTIONS**

a. Inspection Scope (IPs 69001 and 40755)

The inspector reviewed selected aspects of:

- Technical Safety Review Committee (TSRC) meeting minutes
- GTRR staff safety review records
- TSRC and GTRR staff audit records
- responses to safety reviews and audits
- personnel qualifications
- GTRR DP dated June 1998
- TS, Amendment No. 14, dated July 22, 1999

b. Observations and Findings

DP Section 2.4 states that the TSRC: 1) will review and approve all plans, policies and procedures to be performed under the GTRR Decommissioning Project, 2) will review and audit the decontamination and decommissioning project operations and activities, 3) members will be appointed by the President of Georgia Tech, and 4) will keep a written record of the meetings and will report directly to the President.

During inspections in 2000 and 2002, the inspector reviewed the qualifications of the TSRC members and confirmed that they met the requirements specified in TS Section 5.2 and DP Section 2.4. The results of the 2000 inspections were documented in NRC Inspection Report (IR) No. 50-160/2000-201 dated March 15, 2000, NRC IR No. 50-160/2000-202 dated August 31, 2000, and NRC IR

No. 50-160/2000-203 dated December 1, 2000. The inspector noted that the TSRC met more often than the required semiannual frequency and that a quorum was present each time. The inspector reviewed the minutes of the TSRC and determined that they provided guidance, direction, operations oversight, and 10 CFR 50.59 request reviews as required by the DP and TS.

TSRC meeting minutes and audit records and GTRR staff audit records showed that safety reviews and audits were conducted as required by TS Section 5.2(d). The content of the audits and safety reviews were consistent with the TS. These reviews provided appropriate guidance, direction, and oversight to ensure satisfactory decommissioning of the reactor.

By examining the TSRC's review of the DP and their audits of the operations and training programs, the inspector determined that the safety reviews, audits, and associated findings were satisfactory and that the licensee took the appropriate corrective actions in response to the findings.

The inspector reviewed selected decommissioning and facility change approvals. Records and observations showed that changes at the facility were acceptably reviewed in accordance with 10 CFR 50.59 and applicable licensee administrative controls. None of the changes constituted an unreviewed safety question or required a change to the TS. The inspector determined that TSRC 10 CFR 50.59 request reviews were adequately performed.

c. Conclusions

The audits conducted by the TSRC and GTRR staffs were in accordance with the requirements specified in TS Section 5.2 and DP Section 2.4. TSRC 10 CFR 50.59 request reviews were adequately performed.

**3. PROCEDURES**

a. Inspection Scope (IPs 69001 and 40755)

The inspector reviewed selected aspects of:

- administrative controls
- records for changes and temporary changes
- DP dated June 1998
- TS, Amendment No. 14, dated July 22, 1999
- decommissioning procedures
- logs and records

b. Observations and Findings

During decommissioning activities, the inspector confirmed that written health physics (HP) and decommissioning procedures were available for those tasks and items required by TS Section 5.3 and the DP Sections 2.3.1.1. and 3.1.2.2. The procedures

were routinely updated and then approved by the TSRC while minor modifications to the procedures were approved by the facility director.

Decommissioning procedures and operating plans reviewed and approved by the TSRC included those dealing with:

- Initial Radiological Survey Plan and Procedures
- Health and Safety Plan and Procedures
- Waste Management Plan and Procedures
- Management Plan
- Quality Assurance Plan and Procedures
- Radiation Protection Plan and Procedures
- Decommissioning Work Plan
- Final Radiological Survey Plan

Through review of the 2000 training records and interviews with staff, the inspector determined that the training of staff and contractor personnel concerning procedures was adequate. During the inspector's tours of the facility, it was observed that personnel performing radiation surveys, conducting instrument checks, issuing dosimetry, and performing the decommissioning work were doing so in accordance with applicable procedures.

c. Conclusions

Based on the procedures and records reviewed and observations of personnel during the inspections in 2000, it was determined that the procedural control and implementation program was acceptably maintained and met TS and DP requirements.

**4. REMOVAL OF MATERIALS**

a. Inspection Scope (IPs 69001, 86740, and 85102)

The inspector reviewed selected aspects of:

- transportation records
- disposal records
- NRC Forms 741 and 742
- DP dated June 1998

b. Observations and Findings

From 1964 through 1995, the licensee operated a heavy water moderated and cooled research reactor at the NNRC. The reactor was shut down on November 17, 1995, in preparation for the summer Olympic Games in Atlanta, GA, and was never restarted.

As noted in a previous NRC IR No. 50-160/1996-01, the irradiated fuel was shipped to the Savannah River Site on February 18, 1996. The licensee had previously shipped

the unirradiated fuel to the Oak Ridge National Laboratory site in Tennessee on January 31, 1996. The inspector confirmed that, as noted by DP Section 1.5, all fuel had been removed from NNRC prior to decommissioning.

Fifty-six (56) total radioactive waste shipments were made during the GTRR decommissioning. The final waste shipment occurred on August 3, 2001. Radioactive waste was sent to one of four consignees: 1 Duratek Inc.; 2 CNSI Barnwell; 3 Envirocare of Utah; and 4 Westinghouse Savannah River Site. During 2000, the inspector confirmed through records review, interviews with licensee staff, and actual observation, that radioactive waste was disposed of as required by DP Section 3.2 and in accordance with Department of Transportation and NRC regulations.

c. Conclusions

As a result of the records review and on-site observations made during decommissioning tours, it was confirmed that the fuel and radioactive waste were removed from the site in accordance with the GTRR DP requirements, and Department of Transportation and NRC regulations.

**5. DECOMMISSIONING ACTIVITIES**

a. Inspection Scope (IPs 69001 and 40755)

The inspector reviewed selected aspects of:

- operational logs and records
- decommissioning procedures
- decommissioning logs and records
- DP dated June 1998
- the facility during tours

b. Observations and Findings

As noted above, the reactor was permanently shut down on November 17, 1995. All irradiated reactor fuel was removed from the site on February 18, 1996. On July 22, 1999, following a request by the licensee and a review by the NRC, Amendment No. 14 to Facility License No. R-97 was issued which authorized decommissioning of the GTRR. The licensee's contractor started its decommissioning of the facility in January 2000. (Actual decommissioning of the facility was completed in May 2001, although the contractor's final survey of the facility continued for several months afterwards.)

Decommissioning activities focused on the dismantling and removal of the reactor proper, its support structures, auxiliary equipment and components, and the biological shield. The inspector examined the following selected tasks as directly described in DP Section 2.3, Decommissioning Activities and Tasks:

Reactor Complex

Vertical Beam Ports - The vertical beam ports will be removed - including the thimbles, thimble plugs, sample tubes, and liners. The lead will be removed from the plugs and sent to a mixed waste processor. The other items will be segmented as necessary, packaged, and disposed of as radioactive waste.

Shim Safety Rods and Drives - The four shim safety rods will be disconnected from the drives, removed through the top shield, cut in half, and disposed of as mixed waste. The shim safety rod drives will be disconnected, removed, segmented, and disposed of as radioactive waste.

Horizontal Beam Gates - The ten horizontal beam gate drive motors will be disconnected and removed. The gates will be separated from the shafts and cut open. The lead inside will be removed and disposed of as mixed waste, and the remainder disposed of as radioactive waste.

Spent Fuel Storage Holes - The spent fuel storage hole plugs will be removed and disposed of as radioactive waste. The hole liners will be core drilled out and each liner will be cut in half, packaged, and disposed of as radioactive waste.

Piping and Instrumentation - This task involved the removal of miscellaneous piping and ventilation in and around the reactor complex. The materials will be disposed of as radioactive waste.

Lead Cover Plate - The lead cover plate will be removed in two distinct pieces - the inner plate and outer plate. The 24 lead and steel port plugs will be removed from the inner plate and cut open with an abrasive saw. The lead will be removed and disposed of as mixed waste, and the steel will be disposed of as radioactive waste.

Upper Top Shield - The upper top shield will also be removed in two distinct pieces - the inner shield plug and outer shield plug. The 24 concrete and steel inner port plugs and eight concrete and steel outer port plugs will be removed and disposed of as radioactive waste. The inner concrete and steel upper top shield will be removed and disposed of as radioactive waste. The outer concrete and steel upper shield plug will be removed and disposed of as radioactive waste.

Lower Shield Plug - The 31 lead, concrete, and steel port plugs will be removed from the lower top shield plug and cut open with an abrasive saw. The lead will be removed and disposed of as mixed waste. The remaining concrete and steel will be disposed of as radioactive waste.

Fuel Spray Manifold - The fuel spray manifold pipe will be cut free within the reactor, utilizing long-handled tools, and transferred to the contamination control envelope. The manifold will be further segmented and disposed of as radioactive waste.

Reactor Vessel - A remote operated robotic arm will be installed in the reactor vessel to facilitate segmentation. Using an abrasive saw connected to the robotic arm, the horizontal beam ports and through tubes will be cut free and lifted out.

The bottom pipes will be core bored and removed. The reactor vessel will be cut into sections using an abrasive saw mounted on the robotic arm. Lifting holes will first be drilled into each section with a drill attached to the robotic arm, and each section rigged. Each section will be lifted out with the overhead crane, transferred to the packaging area and disposed of as radioactive waste.

Graphite Retaining Sleeve - The graphite retaining sleeve will be removed in a similar fashion as the vessel. Each section will be disposed of as radioactive waste.

Graphite Removal - The 4-inch by 4-inch graphite stringers will be removed using long-handled tools from either the top of the biological shield or through the thermal column. The graphite will be packaged and disposed of as radioactive waste.

Horizontal Beam Ports - The beam port and through tube plugs will be removed and disposed as radioactive waste. Lead will first be removed from the through tube plugs by cutting the top off the plugs with an abrasive saw. The lead will be disposed of as mixed waste.

Boral Removal - The 1/4-inch boral sheet staked to the inside of the steel tank will be removed in a similar fashion as the vessel. Each section will be disposed of as radioactive waste.

Inner Steel Tank - The inner steel tank will follow a similar removal scenario to that described for the boral removal. The tank will be cut into sections using an abrasive saw mounted on the robotic arm. Lifting holes will first be drilled into each section, and each section will then be rigged. After cutting, the section will be transferred to the packaging area using the overhead crane. Each section will be disposed of as radioactive waste.

Lead Thermal Shield - The lead thermal shield was formed by pouring molten lead into the space between the inner and outer steel tanks. With the inner tank and cooling coils removed, the lead will be pried free of the outer tank in easily handled pieces with long-handled tools. The pieces will be lowered into a basket and transferred to a waste container. The lead will be disposed of as mixed waste.

Outer Steel Tank - The outer steel tank will be removed using the same methods as the removal of the inner steel tank. The tank may have to be pried free of the concrete prior to removal. Each section will be disposed of as radioactive waste.

Thermal Column Shutter and Shielding - In order to remove the thermal column shutter and shields, the two thermal column door plugs will be removed first, segmented with an abrasive saw and the lead removed. The steel cover plate will then be removed, segmented and packaged. The exposed lead shield will then be removed and packaged for processing. The concrete and steel blocks will also be removed and packaged. Segmenting of these blocks is not required. The concrete, steel and lead doors will be removed, segmented and packaged. Any



remaining lead will then be removed and packaged for disposal. The concrete and steel will be disposed of as radioactive waste and the lead as mixed waste.

Biomedical Irradiation Facility Shutter and Shielding - In order to remove the biomedical irradiation facility shutter, the aluminum cover plate will be removed first and segmented. The exposed lead bricks will then be removed and packaged. The movable shield plugs and doors will also be removed. The outer bismuth shield, the water tank, and the inner bismuth plug will be removed and packaged. Due to the package restrictions, segmenting of these items will have to be performed. The materials will be disposed as radioactive waste.

Fission Chambers - The fission chambers will be removed and packaged for disposal. The remaining U-235 will be packaged and shipped to an appropriate site.

#### Biological Shield

Activated Concrete - Due to the relatively small amount of activated concrete and the limited access, the concrete will be removed with a bobcat/jackhammer. The waste will be packaged and disposed of as radioactive waste.

Bottom Shield - As above, due to the relatively small amount of activated concrete and the limited access the concrete will be removed with a bobcat/jackhammer. The waste will be packaged and disposed of as radioactive waste.

During the inspections in 2000, the inspector observed various of these activities as they were being conducted including: piping and instrumentation, upper top shield, graphite removal, lead thermal shield, fission chambers, and activated concrete. In order to verify that all the above tasks had been performed in accordance with the DP, the inspector also reviewed the related licensee and contractor records and surveys, and toured the facility. The inspector determined that the above tasks had been completed in accordance with final approved DP.

#### c. Conclusions

Based on the observations made during the inspection, decommissioning activities have been performed as required by DP Section 2.3 and licensee procedures.

### **6. RELEASE CRITERIA**

#### a. Inspection Scope (IPs 69001 and 40755)

The inspector reviewed selected aspects of:

- DP dated June 1998
- Georgia Institute of Technology Research Reactor Decommissioning Project Characterization Report, issued May 1998
- Final Status Survey Report for the GTRR facility issued June 2002

b. Observations and Findings

The primary contaminants of concern for the GTRR are beta-gamma emitters—fission and activation products—resulting from reactor operation. The NRC-approved guidelines for release for unrestricted use for building surfaces were based on those for beta-gamma emitters contained in NRC Regulatory Guide 1.86 (NRC 1974). These guidelines are:

- 5,000  $\beta$ - $\gamma$ - dpm/100 cm<sup>2</sup>, averaged over a 1 m<sup>2</sup> area
- 15,000  $\beta$ - $\gamma$ - dpm/100 cm<sup>2</sup>, maximum in a 100 cm<sup>2</sup> area
- 1,000  $\beta$ - $\gamma$ - dpm/100 cm<sup>2</sup>, removable.

However, due to the presence of the hard-to-detect-radionuclides H-3 and Fe-55, the above guidelines were modified to account for the contributing activity of these radionuclides. The modified guidelines are (Shaw 2002):

- 2,400  $\beta$ - $\gamma$ - dpm/100 cm<sup>2</sup> average activity in a 1 m<sup>2</sup> area
- 7,200  $\beta$ - $\gamma$ - dpm/100 cm<sup>2</sup> maximum activity in a 100 cm<sup>2</sup> area
- 313  $\beta$ - $\gamma$ - dpm/100 cm<sup>2</sup> removable activity

GIT's final survey plan (GTS 2000) stated that radionuclide concentrations in soil for the contaminants of concern would meet the NRC published (Federal Register Vol. 64 page 68396, December 7, 1999) screening values for selected radionuclides in surface soils. The screening values for the GTRR radionuclides of interest are summarized below.

<b>Radionuclide</b>	<b>Guideline Value (pCi/g)</b>
H-3	110
Fe-55	10,000
Pu-239/240	2.3
U-233/234	13.0
U-238	14.0
Ni-59	5,500
Cs-134	5.7
Cs-137	11.0
Co-60	3.8
Eu-152	8.7
Eu-154	8.0
Mn-54	15.0
Ag-110m	3.9
Zn-65	6.2
Sr-90	1.7
C-14	12.0
Ni-63	2,100
Tc-99	19.0

The inspector observed and interviewed Duratec, IT's representative.

The inspector determined that Duratec used the appropriate guideline and screening values as calculated in the Characterization Report and specified in the approved DP.

c. Conclusions

Duratec used the appropriate guideline and screening values as required by the DP, in performing the final survey.

**7. CONFIRMATORY FINAL SURVEY**

a. Inspection Scope (IPs 69001 and 40755)

The inspector reviewed selected aspects of:

- DP dated June 1998
- Georgia Institute of Technology Research Reactor Decommissioning Project Characterization Report, issued May 1998
- Final Status Survey Report for the GTRR facility issued June 2002

b. Observations and Findings

(1) Overview

DP Section 4.0, "Proposed Final Radiation Survey Plan," describes the final radiation survey to be conducted of the facility prior to license termination. This survey is required "in order to ensure that the area satisfies the unrestricted release criteria for radioactive material according to NUREG/CR- 5849." (DP Section 4.1) Additionally, DP Section 4.2.3 specifies, "As stated in NUREG/CR-5849, proper documentation of every aspect of the final survey is necessary for future reference to the decommissioning survey. An accurate mapping of the reactor containment building and surrounding areas within this decommissioning project will be maintained for future review and verification by a regulatory inspector."

Although the licensee is responsible for performing and documentation the decommissioning and final status survey (Final Status Survey Report for the GTRR facility issued June 2002), the NRC verifies the licensee's performance through inspections during decommissioning and a confirmatory final survey at the end.

As part of this confirmatory process ESSAP reviewed and evaluated GIT's final survey plan and report (GTS 2000 and Shaw 2002). The documents were reviewed for general thoroughness, accuracy, and consistency. Data were evaluated to assure that areas exceeding guidelines were identified and had undergone remediation. Final status survey results were compared with guidelines to ensure that the data had been interpreted correctly. Comments were provided

to the NRC, documenting the review of the final survey plan and the final survey report.

The procedures, methods, and data submitted by GIT were considered to be appropriate and adequately documented the radiological status of the GTRR. ESSAP confirmed that the licensee modified the gross activity guidelines to account for hard-to-detect radionuclides. This data was reviewed by ESSAP to evaluate its appropriateness of use and determined it to be satisfactory.

ESSAP performed confirmatory surveys of the GTRR during the period October 21 to 23, 2002. The surveys were performed in accordance with the site-specific survey plan submitted to and approved by the NRC and the ORISE/ESSAP Survey Procedures and Quality Assurance Manuals (ORISE 2002a, 2000a, and 2002b). ESSAP surveys, their individual findings, and overall results are described in the sections following.

## (2) Surface Scans

Surface scans for beta and gamma radiation were performed over approximately 100 percent of the floor surfaces in the basement and on the first floor and 50 percent of the floor surfaces on the second floor. Surface scans for beta radiation were performed over approximately 50 percent of the lower walls in the basement, excluding the Stairwell General Area, 10 percent on the first floor, and 5 percent on the second floor. Surface scans for beta radiation were also performed in the vessel tunnel over approximately 50 percent of the surface.

Particular attention was given to remediated and adjacent surfaces, cracks and joints in the floors and walls, and other locations where residual radioactive material may have accumulated. Surface scans were not performed on any upper wall or ceiling surfaces, in the Helium Rupture Disk Chamber, or in the Reactor Building Ventilation Hold-Up Duct areas. Scans were performed using gas proportional and NaI scintillation detectors coupled to ratemeters or ratemeter-scalers with audible indicators. Locations of elevated direct radiation were noted for further investigation.

ESSAP identified two areas of elevated beta surface radiation. One area was found on a scabbled portion of the wall in the Bismuth Leak area. Another area was found on the floor of the processor equipment room. The concrete block walls in the air compressor room were also noted as being uniformly elevated. Scans of the remaining surfaces did not identify any additional locations of elevated beta or gamma radiation.

Surface scans of outdoor locations including soil areas, paved areas, and gravel surfaces were performed over approximately 50 to 100 percent of the accessible areas using a sodium iodide scintillation detector coupled to a ratemeter.

Gamma surface scans were within the range of ambient background levels except for an area adjacent to the NNRC that was determined to be caused by radiation “shine” from the hot cell facility and storage vault.

### (3) Surface Activity Measurements

Construction material-specific backgrounds were determined in areas of similar construction, but without a history of radioactive material use. Ambient gamma backgrounds were determined in areas where direct beta measurements were performed; these background measurements were used to correct gross beta surface activity measurements.

Direct measurements for total beta activity were performed at 35 locations, chosen randomly and based on surface scan results. Additional measurements to determine the average activity level in one area were also performed. Dry smears were collected at each direct measurement location for determining removable gross alpha and gross beta activity. Wet smears were collected from areas adjacent to direct measurement locations to determine the H-3 and C-14 activity. Direct measurements were performed using gas proportional detectors coupled to ratemeter-scalers.

ESSAP identified an activity of 9,700 dpm/100 cm<sup>2</sup> over approximately 0.5 m<sup>2</sup> in the elevated area identified in the Bismuth Leak area, with an average activity of 1700 dpm/100 cm<sup>2</sup> over the contiguous one square meter area. The elevated area identified in the process equipment room was limited to approximately 100 cm<sup>2</sup> with an activity of 4,100 dpm/100 cm<sup>2</sup>. An activity range of 2,700 to 5,100 dpm/100 cm<sup>2</sup> was determined for the concrete block in the air compressor room, which GIT claimed resulted from naturally occurring radioactive material in the blocks. Confirmatory scans on the interior and exterior of the room found the radiation levels to be evenly distributed throughout the blocks, confirming the activity was from the material used to make them. Removable activity levels ranged from 0 to 3 dpm/100 cm<sup>2</sup> for gross alpha and from -5 to 45 dpm/100 cm<sup>2</sup> for gross beta. H-3 removable activity levels ranged from 3 to 466 dpm/100 cm<sup>2</sup>. C-14 removable activity levels ranged from -2 to 86 dpm/100 cm<sup>2</sup>.

### (4) Exposure Rate Measurements

ESSAP obtained background exposure rate measurements from various locations within the NNRC, having similar construction as the GTRR. The NNRC has a site history of radiological material usage; however, there are no other buildings similar in construction to the GTRR and NNRC on the GIT campus. Exposure rate measurements, using a microrem meter at one meter above the floor, were performed in the center of selected areas or rooms within the GTRR.

Average interior building exposure rates ranged from 9 to 25  $\mu$ R/h. Background exposure rates performed in the NNRC ranged from 18 to 20  $\mu$ R/h.

Exterior exposure rate measurements, using a microrem meter at one meter above the surface, were performed at five random locations from the reactor yard area surrounding the GTRR.

Average exterior exposure rates ranged from 14 to 18  $\mu\text{R/h}$ . Background exposure rates performed at various intersections on the GIT campus ranged from 12 to 20  $\mu\text{R/h}$ .

(5) Sampling

ESSAP collected surface soil (0-15 cm) samples at each exposure rate measurement location.

Analysis of the soil samples by gamma spectroscopy for gamma-emitting mixed fission and activation products identified Cs-137 at typical fall out concentrations. Radionuclide concentrations for Co-60 and Cs-137, which are the predominant radionuclides of concern at research reactor facilities ranged from -0.02 to 0.03 pCi/g for Co-60 and -0.02 to 0.21 pCi/g for Cs-137. All other radionuclides of concern were reported as less than the respective minimum detectable concentration of the procedure, which ranged from 0.03 to 0.11 pCi/g.

(6) ESSAP Results

Compliance for residual surface activity was shown using the GIT calibration methodology approved by the NRC. Since ESSAP's calibration method differs, this required adjusting the ESSAP-calculated surface activity by the ratio of the efficiencies for the GIT and ESSAP methods. The correction factor was approximately 2.3. All corrected ESSAP confirmatory surface activity measurements, including the identified elevated areas, met guidelines and did not require further remediation. Additional investigation by the inspector verified that the concrete block in the air compressor room was made from material with a high composition of naturally occurring radioactive material.

Except for the air compressor room in the basement, all exposure rate measurements were less than 5  $\mu\text{R/h}$  above background for each survey unit.

Confirmatory surface soil samples were less than the screening values listed in the GIT final survey plan (GTS 2000).

c. Conclusions

Based on the above observations, surveys, evaluations, and analyses, the inspector concluded that: 1) the elevated surface activity and exposure readings in the basement compressor room were due to naturally occurring radioactive material; and 2) based on the results of the licensee's final status survey and ESSAP's confirmatory measurements, GIT has adequately demonstrated that the GTRR facility satisfies the criteria for release for unrestricted use.

**8. MAINTENANCE AND SURVEILLANCE**

a. Inspection Scope (IP 40755)

The inspector reviewed selected aspects of:

- maintenance procedures
- equipment maintenance records
- surveillance and calibration procedures
- surveillance, calibration, and test data sheets and records
- reactor periodic checks, tests, verification, and decommissioning activities
- facility design and DP changes and records
- NNRC Procedure 4200, "10 CFR 50.59 Review Program for Changes and Tests During Decommissioning," Revision 01, dated November 1, 1999
- TS, Amendment No. 14, dated July 22, 1999

b. Observations and Findings

(1) General Maintenance

During decommissioning general maintenance was focused on the support services and equipment and not on any reactor systems. The inspector reviewed maintenance records, interviewed staff and observed minor maintenance performed on the various systems in operation. Based on the inspector's interviews and observations, general maintenance was acceptable for an industrial site.

(2) Surveillance

The inspector reviewed records of the TS Section 3 required surveillance verifications performed during 2000. The results of the surveillances for the radiation monitoring system and the ventilation system were within prescribed TS limits and procedure parameters, and in close agreement with the previous surveillance results.

(3) Change Control

TS or DP related 10 CFR 50.59 changes required review by the TSRC in accordance with TS Section 5.2.

The inspector reviewed various TSRC approved change packages for changing the method of accomplishing certain decommissioning activities. The inspector determined that the changes had been evaluated, reviewed, and approved as required by NNRC Procedure 4200, "10 CFR 50.59 Review Program for Changes and Tests During Decommissioning," Revision 01, dated November 1, 1999. The reviews were technically complete and adequately documented. Additionally, the inspector concluded that TSRC 10 CFR 50.59 reviews and approvals were focused on safety, and met licensee program requirements.

c. Conclusions

The licensee's program for surveillance and limiting conditions for operation verification satisfied TS and DP requirements. The licensee's maintenance and design change programs were in place and were being implemented as required by licensee procedures.

**9. RADIATION PROTECTION**

a. Inspection Scope (IPs 69001 and 40755)

The inspector reviewed selected aspects of the radiation protection program (RPP):

- Radiation Protection Training
- radiological signs and posting



- facility and equipment during tours
- routine surveys and monitoring
- survey and monitoring procedures
- dosimetry records
- maintenance and calibration of radiation monitoring equipment
- periodic checks, quality control, and test source certification records
- NNRC Radiation Protection Program (RPP)
- event/incident records

b. Observations and Findings

(1) Radiation Protection Program

Although individual procedures had been revised and some added, the RPP had not functionally changed since the last inspection. The licensee reviewed the RPP at least annually in accordance with 10 CFR 20.1101(c). This review and oversight was provided by the TSRC as required by TS Section 5.2.d(9) and DP Section 2.4.3.

The inspector's review of procedure change records, revisions, and radiation work permits (RWP), confirmed that the RSO, individually and as a TSRC member, reviewed and approved RWPs, and advised the Director and TSRC on matters regarding radiological safety as required by TS Section 5.1.b, DP Section 2.4.1, and the RPP.

Through record reviews and interviews with GTRR and Duratec staffs, the inspector confirmed that the RPP was applied to all activities during the decommissioning project, as required by DP Section 3.1 and GTRR procedures.

(2) Radiation Protection Postings

The inspector observed that caution signs, postings and controls to radiation and contaminated areas at the NNRC were acceptable for the hazards involved and were implemented as required by 10 CFR Part 20, Subpart J. The inspector observed licensee and contractor personnel and verified that they complied with the indicated precautions for access to such areas. The inspector confirmed that current copies of NRC Form-3 and notices to workers were posted in appropriate areas in the facility as required by 10 CFR Part 19.11.

(3) Radiation Protection Surveys

The inspector audited the GTRR daily, monthly, quarterly, and other periodic contamination and radiation surveys, including airborne activity sampling, performed from 2000 to 2003. The surveys were performed and documented as required by DP Section 3.0, and GTRR survey procedures. HP surveys required for special decommissioning activities, such as RWPs, were also performed and documented as required. Results were evaluated and corrective actions taken and documented when readings/results exceeded set action levels.



(4) Dosimetry

The inspector confirmed that dosimetry was issued to staff, contractors, and visitors as outlined in licensee procedures. The licensee's dosimetry issuing criteria specified that dosimetry should be issued to individuals who might receive a dose equivalent exceeding 10 percent of the annual limits specified in 10 CFR Part 20.1201(a). This criteria meet the requirements of 10 CFR 20.1502 for individual monitoring. Training records showed that personnel were acceptably trained in radiation protection practices. During the inspection the inspector observed that workers and staff wore their dosimetry as required.

The licensee used a National Voluntary Laboratory Accreditation Program-accredited vendor to process personnel thermoluminescent dosimetry. Dosimetry results were reviewed by the RSO and doses above the facility's ALARA limits were investigated as required. The inspector's review of the licensee's radiological exposure records from 2000 to 2003 verified that occupational doses were within 10 CFR Part 20 limitations.

(5) Radiation Monitoring Equipment

The calibration and periodic checks of the portable survey meters, radiation monitoring, air sampling, and counting lab instruments were performed by facility staff or by certified contractors. The inspector confirmed that the licensee's calibration procedures and annual, quarterly, semiannual and monthly calibration, test, and check frequencies satisfied TS Section 4.3.3, DP Section 3.1, and 10 CFR 20.1501(b) requirements, and the American National Standards Institute N323 "Radiation Protection Instrumentation Test and Calibration" or the instruments' manufacturers' recommendations. The inspector verified that the calibration and check sources used were traceable to the National Institute of Standards and Technology and that the sources' geometry and energies matched those used in actual detection/analyses.

The inspector also reviewed Duratec instrument calibrations. Their calibration and periodic checks of the portable survey meters, radiation monitoring, air sampling, and counting lab instruments were performed by their staffs or by certified contractors. The inspector confirmed that calibration procedures and annual, semiannual quarterly, monthly, and daily calibrations, tests, and check frequencies satisfied Duratec HPS procedures. Calibrations also met 10 CFR Part 20.1501(b) requirements, and the American National Standards Institute N323 "Radiation Protection Instrumentation Test and Calibration" or the instrument's manufacturers' recommendations. The inspector verified that the calibration and check sources used were traceable to the National Institute of Standards and Technology and that the sources' geometry and energies matched those used in actual detection/analyses.

The inspector reviewed the calibration lists and confirmed that calibrations for the radiation monitoring and counting lab equipment in use had been performed and that all portable instruments in use were calibrated.

All instruments checked by the inspector had current calibrations appropriate for the types and energies of radiation they were used to detect and/or measure.

(6) Respiratory Protection

DP Section 3.1.6 states that the Respiratory Protection Program will be implemented by the decommissioning contractor in compliance with ANSI Z-88.2, US NRC Regulatory Guide 8.15, 10 CFR 20.1701 through 20.1704, and OSHA requirements.

While conducting inspections during decommissioning activities at the facility, the inspector reviewed the respiratory protection program in use by contractor personnel. The inspector noted that the licensee and contractor had established a respiratory protection program as required by DP Section 3.1.6 and were using tested and certified NIOSH/MSHA equipment as required. Records and observation showed that air sampling was being conducted, surveys and bioassays were completed as required, testing of respirators was being done, fit testing of individuals was performed, and individuals were required to pass a physical in order to qualify to use a respirator. The respiratory protection program was in compliance with 10 CFR 20.1703 and the DP.

(4) Effluents

The program for the monitoring and storage of radioactive liquid, gases, and solids was acceptable. Radioactive effluents were monitored and released when within established limits as outlined in licensee procedures and the regulations. The principles of As Low As Reasonably Achievable (ALARA) were acceptably implemented to minimize radioactive releases. Monitoring equipment was maintained and calibrated as required. Records were current and acceptably maintained.

c. Conclusions

Based on the observations made and records audited, it was determined that, because: 1) surveys were completed and documented as required by 10 CFR 20.1501(a) and licensee procedures, 2) postings met regulatory requirements, 3) the personnel dosimetry program was acceptably implemented and doses were in conformance with licensee and 10 CFR Part 20 limits, 4) portable survey meters, radiation monitoring, and counting lab instruments were maintained and calibrated as required, 5) the evaluation and administration of the respiratory program were adequately performed, and 6) the program for monitoring, storage, and release of effluents was acceptable, the RPP implemented by the licensee satisfied NRC and DP requirements.

**5. EXIT MEETING SUMMARY**

The inspector presented the inspection results to members of licensee management at the conclusion of the inspection on October 23, 2002. The licensee acknowledged the

findings presented and did not identify as proprietary any of the material provided to or reviewed by the inspector during the inspection.

### **PARTIAL LIST OF PERSONS CONTACTED**

*T. Bauer	Project Leader, ESSAP
*T. Brown	Field Staff, ESSAP
*R. Eby	Executive Engineer, (Vice President Energy, Environment, and Systems) CH2M HILL
*N. Hertel	Director, Neely Nuclear Research Center
*R. Ice	Manager, Office of Radiation Safety
P. Jones	Project Manager, GTS Duratec Field Services
G. Kalinauskas	Senior Project Engineer, IT Corporation
R. Morton	Field Staff, ESSAP

\* Attended exit meeting.

The inspector also contacted other supervisory, technical and administrative staff personnel as well.

### **INSPECTION PROCEDURE (IP) USED**

IP 69001	Class II Non-Power Reactors
IP 40755	Class III Non-power Reactors
IP 85102	Material Control and Accounting - Reactors
IP 86740	Inspection of Transportation Activities

### **ITEMS OPENED AND CLOSED**

#### **Open**

None

#### **Closed**

None

### **PARTIAL LIST OF ACRONYMS USED**

Duratec	GTS Duratec
DP	Georgia Institute of Technology Research Reactor Decommissioning Plan dated June 1998
ESSAP	Environmental Survey and Site Assessment Program
GIT	Georgia Institute of Technology
GTRR	Georgia Institute of Technology Research Reactor
HP	Health Physics
IT	IT Corporation
NNRC	Neely Nuclear Research Center
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
RWP	Radiation Work Permits
RPP	Radiation Protection Program
RSO	Radiation Safety Officer
TS	Technical Specifications

TSRC      Technical Safety Review Committee