UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

November 1, 2010

Ms. Charlotte Engstrom
Vice President and General Counsel
General Atomics
P.O. Box 85608
San Diego, CA 92186-9784

SUBJECT: NRC INSPECTION REPORT NO. 50-163/2010-01; 50-89/2010-01

Dear Ms. Engstrom,

The Nuclear Regulatory Commission's (NRC) Region IV office completed an inspection at your Torry Pines Mesa site near San Diego, CA involving four site visits conducted during the period from February 10, 2010 through August 30, 2010. These inspections were to observe activities related to the removal of all TRIGA spent fuel from your site, as licensed under your two NRC Part 50 licenses (R-38, Docket # 50-89 and R-67, Docket #50-163), and the shipment of the spent fuel to the Department of Energy's Idaho National Laboratory for storage. Additionally, one inspection was conducted at the Columbiana Hi Tech Fabrication facility in Greensboro, NC on June 16 - 17, 2010 to review the fabrication of failed fuel cans and transportation baskets being manufactured to support your fuel shipment activities. The results of the Columbiana Hi Tech inspection are included in this inspection report. Interim exit briefings were conducted with your staff on February 11, 2010, July 16, 2010, and July 30, 2010 with a final exit briefing conducted on September 21, 2010 after the third and final shipment was received at the Idaho National Laboratory on September 16, 2010. An exit briefing was also conducted at the Columbiana Hi Tech facility on June 17, 2010 at the conclusion of that portion of our inspection. In addition to the operational inspections conducted by NRC Region IV, security inspections were conducted by the NRC's Office of Nuclear Security and Incident Response (NSIR) and Region II. The results of the security inspections are issued as a separate inspection report, 070-734/2010-401, dated October 21, 2010 (Not Publically Available).

The operational inspection conducted by NRC Region IV involved a thorough review of your plans and preparation to remove the spent TRIGA fuel from your spent fuel pool in the Mark F reactor facility, the examination of the condition of the spent fuel to identify those fuel elements that would require special packaging in failed fuel cans prior to placement in the transportation cask, the actual placement and sealing of the damaged fuel into the failed fuel cans and the loading of the first NAC Legal Weight Truck (LWT) cask for shipment to Idaho.

The operations at your facility were safely performed and resulted in all three shipments being successfully completed with no significant issues identified. The personnel performing the work were very knowledgeable and qualified to perform their particular functions. GA did an excellent job of coordinating activities among the various groups needed to support the activities. This included personnel from Idaho National Laboratory, NAC International, the crane vendor, your

Non-Destructive Examination (NDE) contractor, security transport services and the San Diego police department. The GA personnel performed well in the areas of overall management of activities, fuel handling, radiation protection, and quality control.

During the activities observed by the NRC, two violations of NRC regulations were identified. The first involved the receipt inspection process for the baskets and the tagging of these items to show their operational status in accordance with your quality assurance program and 10 CFR Part 71, Subpart H “Quality Assurance.” The second violation related to a leak test conducted on the drain plug of one of the failed fuel cans to determine why the vacuum drying process was not being successful. A leak test was performed on the can without developing a field change and obtaining the necessary approvals as required by your site procedures. These two violations are discussed further in the attached “Inspector Notes.” See “Receipt Inspections” under the QA category and “Field Change” under the Procedures Category. Because of their low safety significance and because they were entered into your corrective action program, the NRC is treating these violations as Non-Cited Violations (NCV), consistent with Section 2.3.2 of the enforcement policy.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (if any) 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 http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Should you have any questions concerning this inspection, please contact the undersigned at 817-860-8191 or Senior Health Physicist Mr. Vincent Everett at 817-860-8198.

Sincerely,

/RA/

D. Blair Spitzberg, PhD, Chief
Repository & Spent Fuel Safety Branch

Docket: 50-163; 50-89
License: R-67; R-38

Enclosure:
NRC Inspection Report 50-163/10-01; 50-89/10-01

Attachments:
(1) Supplemental Inspection Information
(2) General Atomics Fuel Shipment – Inspector Notes

cc w/enclosure:

Dr. Keith E. Asmussen, Director
Licensing, Safety and Nuclear Compliance
General Atomics
P. O. Box 85608
San Diego, CA 92186-9784

Mr. John Greenwood
General Atomics
3550 General Atomics Court
P. O. Box 85608
San Diego, CA 92186-9784

Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Services
P. O. Box 942732
Sacramento, CA 94234-7320

Mr. Rubin Develasco, Manager
Decommissioning Project
General Atomics
P. O. Box 85608
San Diego, CA 92186-9784

Radiation Control Program Director
Gary W. Butner
Radiologic Health Branch
P. O. Box 997414, MS 7610
Sacramento, CA 95899-7414

Test, Research and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainsville, FL 32611

Mr. John Burchfield, Vice President
Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

Mr. Curtis Crissman
Senior Project Manager
Columbiana Hi Tech, LLC
1802 Fairfax Road
Greensboro, NC 27407

Mr. David Schroeder
Project Engineer
NAC International
3930 East Jones Bridge Road
Norcross, Georgia 30092

Mr. Mike Miles
Field Operations Manager
NAC International
3930 East Jones Bridge Road
Norcross, Georgia 30092

Internal distribution w/enclosure via e-mail:
E. Collins, Regional Administrator
Art Howell, D:DNMS
Chuck Cain, DD:DNMS
Blair Spitzberg, C:DNMS/RSFS
J. Weil, Congressional Affairs Officer
Craig Bassett, NRR/DRP/PROB
Earl Love, NMSS/DSFST/LID/R
Jim Shepherd, FSME/DWMEP/DURLD
Robert Temps, NMSS/DSFST/LID/R
James Pearson, NMSS/DSFST/LID/R
Kim Hardin, NMSS/DSFST/LID/L
Vincent Everett, RSFS
Lee Brookhart, RSFS
Marisa Herrera, Fee Coordinator, DRMA

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	■ Publicly Available	■ Non-Sensitive
Category A.	□ Non-publicly Available	□ Sensitive
KEYWORD:		

DNMS:RSFS	RSFS	NMSS	NMSS
V. Everett	L. Brookhart	R.Temps	E. Love
/RA/	***e-mail JVE***	***e-mail JVE***	***e-mail JVE***
10/ 26 /10	10/ 27 /10	10/ 19 /10	10/ 25 /10
NMSS	**C:RSFS**		
J.Pearson	DBSpitzberg		
e-mail JVE	***/RA/***		
10/ 14 /10	11/01/10		

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	050-163;50-089
License:	R-67;R-38
Report:	050-163/2010-01;050-089/2010-01
Licensee:	General Atomics International
Facility:	TRIGA Reactor Facility
Location:	3550 General Atomics Court San Diego, CA
Dates:	GA Fuel Examination Inspection – February 10-11, 2010 Columbiana Hi Tech Fabrication Inspection – June 16-17, 2010 GA Failed Fuel Can Loading Inspection – July 12-16, 2010 GA Program Inspection – July 26-30, 2010 NAC-LWT Cask Loading Inspection – August 26-31, 2010
Inspectors:	Vincent Everett, RIV/RSFSB Lee Brookhart, RIV/RSFSB Earl Love, DNMS/SFST Robert Temps, DNMS/SFST James Pearson, DNMS/SFST
Approved By:	D. Blair Spitzberg, PhD., Chief Repository & Spent Fuel Safety Branch
Attachment:	(1) Supplemental Inspection Information (2) General Atomics Fuel Shipment - Inspector Notes

ENCLOSURE

EXECUTIVE SUMMARY

General Atomics TRIGA Reactor Facility
NRC Inspection Report 050-163/10-01; 50-089/10-01

A series of routine, announced inspections reviewed the procedures, programs, training and operational aspects of the licensee's efforts to remove all spent TRIGA fuel from the General Atomics Torry Pines Mesa site held under NRC license. The spent fuel was packaged into three Nuclear Assurance Corporation (NAC) Legal Weight Truck (LWT) casks for shipment to the Irradiated Fuel Storage Facility (IFSF) at the Idaho National Laboratory (INL). The NAC-LWT cask is approved for use to transport the TRIGA spent fuel in accordance with Certificate of Compliance (CoC) 71-9225, Revision 53 as described in the NAC-LWT Safety Analysis Report, Docket No. 71-9225, Revision 41. The fuel had been used in the Mark F reactor (License # R-67, Docket 50-163), Mark I reactor (License # R-38, Docket 50-89) and the Mark III reactor (License # R-100, Docket 50-227). The Mark F and Mark I reactor facilities are in various stages of decommissioning. The Mark III reactor had been decommissioned and the license terminated in 1975.

A total of 263 TRIGA spent fuel elements were shipped to Idaho consisting of both aluminum clad and stainless steel clad assemblies. Spent fuel elements that were identified during the fuel examinations conducted in February 2010 as having gouges or holes penetrating the cladding were placed in sealed failed fuel cans inside the shipping casks. The fuel elements were enriched at three different levels of 19.5% U-235, 70% U-235 and 93% U-235. Maximum burn-up was 344,812 MWd/MTU. Decay time ranged from 13 years to 46 years.

During the period of 1980 to 1981, thirteen shipments of spent fuel from the Mark F and Mark I reactors, consisting of approximately 360 fuel assemblies, were shipped to DOE-Idaho. The TRIGA fuel packaged and shipped to Idaho during this inspection completes the removal of all NRC licensed TRIGA fuel from the site.

Details related to the 16 technical areas reviewed during this inspection are provided as Attachment 2 to this report. The following provides a summary of the findings for each category listed in Attachment 2.

Corrective Actions

- The licensee had established and implemented a corrective action program that met the requirements of 10 CFR Part 50, Appendix B to identify, track and correct conditions adverse to quality.

Crane Operations

- The 3 ton spent fuel pool crane was adequately sized to handle the approximately two ton loads required to remove the TRIGA fuel out of the pool. Maintenance, testing and inspections were current for the crane.

- The 240 ton mobile crane used to perform the lifts outside the fuel building was adequately sized, with a safety margin of more than 8 to 1 for the NAC-LWT cask loaded with TRIGA spent fuel. The crane had been recently inspected and a load test performed to renew the crane certification. The crane operators were well qualified to perform the heavy lifts.

Emergency Planning

- Adequate provisions had been included in the site emergency plan and procedures for classifying and responding to potential emergencies at the TRIGA facility. Provisions were included in the emergency plan for notifying outside response agencies if support was required.

- Prior to loading the first shipping cask, a thorough safety briefing was provided to all workers concerning the expected dose rates that would be encountered during the movement of the fuel and specific emergency actions to take if alarms sounded.

Fabrication Inspection

- The failed fuel cans and baskets were fabricated and tested at the Columbiana Hi Tech facility. The material was fabricated consistent with approved drawings and had been controlled under the procedural requirements of Columbiana's quality assurance program.

- Documentation was available to demonstrate that materials had been properly controlled during procurement and receipt and was traceable through the entire fabrication process. Good quality controls had been applied.

Failed Fuel Cans

- Failed fuel cans were used to package damaged and potentially damaged fuel elements. Seventeen fuel elements were identified for placement into sealed failed fuel cans during a fuel examination project conducted in February 2010.

- Ten failed fuel cans were used to seal the seventeen damaged fuel elements. The ten failed fuel cans were placed inside the three NAC-LWT shipping casks in accordance with the limitations established in the NAC-LWT Certificate of Compliance.

- The criteria used for determining whether a fuel element was damaged was consistent with the criteria established in the NAC-LWT SAR and the NRC's Interim Staff Guidance (ISG) – 1 "Damaged Fuel and Nuclear Energy Institute Protocol for Classification."

- One of the failed fuel cans had problems during the vacuum drying process. The can continued to fail the required vacuum test and was eventually found to have a damaged metal seal on the drain plug. The seal was replaced and the vacuum test successfully completed. No other failed fuel cans experienced any problems during the vacuum drying and helium backfill process.

Fuel Verification/Loading

- The characteristics of the TRIGA fuel elements met the NAC-LWT Certificate of Compliance for fuel type, enrichment, burn-up, decay time, uranium content and hydrogen to zirconium cladding ratio. A thorough fuel examination was conducted by representatives of the Department of Energy to confirm the condition of the fuel. The examination was observed by the NRC and identified those assemblies which were required to be sealed in failed fuel cans prior to shipment.

- Seventeen fuel elements required placement in sealed failed fuel cans. Ten failed fuel cans were used to hold the 17 fuel elements.

- Assignment of fuel to specific locations in the three shipping casks was completed and approved by GA, NAC international and DOE prior to the loading campaign. The assignments were consistent with the limitations for placement of fuel as specified in the NAC-LWT Certificate of Compliance.

- Two previously sectioned high enriched fuel elements had been stored in bags in a storage drum in the Mark F reactor room. These rubblized fuel elements were included in the fuel examination project and were placed in a sealed failed fuel can for shipment.

- A number of fuel elements required cropping in order to fit into the NAC-LWT baskets. This operations was conducted prior to the cask loading and involved cutting lead-out tubes or extension rods from 42 instrumented fuel elements (IFE) and fueled follower control rods (FFCR) plus 3 non-IFE fuel elements.

Helium Leak Testing

- Helium leak testing on the cask lid was performed after the cask had been vacuum dried and backfilled with helium. The leak rate limit was 2 x 10(-7) cc/sec in accordance with ANSI standard N14.6-1993. The leak rate on the lid was 4.9 x 10(-10) cc/sec.

- The annual helium leak testing on the port covers had been performed by the cask owner, NAC International, and was documented in the annual maintenance packages for all three casks used for the GA shipments.

Material Accountability

- Material accountability records for the fuel shipped from the GA site compared to the records of the amount of TRIGA fuel that had been stored onsite under the NRC license confirmed that all special nuclear material (SNM) under the NRC licenses had been transferred to DOE.

Procedures

- The NAC and GA procedures used for the fuel shipping project contained an adequate level of detail, were consistent with the procedural requirements in the NAC-LWT Safety Analysis Report, and had been approved by the appropriate levels of management.

- The GA procedures included provisions for making field changes during the loading process if conditions arose requiring a deviation from the procedural steps. However, during the vacuum drying of a failed fuel can, the investigation to determine the location of the leak was performed outside the steps of the approved procedure without complying with the field change provisions of the procedure. This was identified as a non-cited violation (NCV).

- Unloading procedures had been developed as a contingency should problems occur requiring the spent fuel to be removed from the shipping cask and returned to the spent fuel pool.

Quality Assurance

- The Quality Assurance Program Document and the Quality Division Instructions were being applied to the TRIGA fuel shipment project in accordance with the Mark F technical specifications and the Decommissioning Plan. However, the quality assurance requirements were not adequately applied to the receipt inspection process for the baskets, in that appropriate tagging of the items to indicate their status was not completed in accordance with the quality division instructions. This was identified as a non-cited violation (NCV).

- Quality assurance audits were adequately performed of the fuel shipping activities. No significant findings were identified by the licensee during the audits.

Radiological

- Good ALARA controls were applied to the work activities for the loading of the damaged fuel into the failed fuel cans and the loading of the fuel into the shipping cask. Work activities involved handling and transferring fuel elements with dose rates over 100 R/hr at 2 inches.

- The total personnel dose received by the workers during the loading of the failed fuel cans during July 12-16, 2010 was 0.365 manrem. The highest individual dose was 141 mrem, well below the 5,000 mrem annual limit. The highest extremity dose was 436 mrem to the hands. This was well below the 50,000 mrem annual limit.

- The total personnel dose received during the two day loading of the first NAC-LWT cask during August 26-30, 2010 was 0.138 manrem, based on the electronic dosimeter readings.

- Radiological practices including continuous air monitoring, periodic smear surveys of the work area, ongoing radiological dose rate measurements and good health physics coverage during the work activities contributed to the low personnel doses received during the fuel handling activities.

Records

- Requirements to retain records in accordance with NRC regulations related to special nuclear material inventories had been incorporated into the appropriate site procedures.

Safety Reviews

- The Criticality and Radiation Safety Working Group was providing oversight of facility activities related to criticality and radiation safety by conducting periodic audits in accordance with their charter.

- No safety evaluations had been performed by the licensee related to the fuel shipping project.

Special Topics

- A dry run practice was conducted the day prior to the actual fuel movement to verify all necessary equipment and supplies were readily available and to work out the fine details of the procedure implementation. The dry run verified that the crane had been appropriately located to facilitate the heavy lifts required to move the spent fuel baskets into the shipping cask.

- The receipt of the GA TRIGA fuel at the Irradiated Fuel Storage Facility at DOE's Idaho National Laboratory completed an effort that had begun 10 years earlier to remove the fuel from the GA Torry Pines Mesa site. The final shipment was received by DOE on September 16, 2010.

Training & Qualifications

- All three personnel assigned as TRIGA reactor operators were currently trained and qualified in accordance with the NRC approved reactor operator requalification program.

Transportation

- Required NRC approval of the transportation route and notification of the states along the route was completed in accordance with NRC regulations.

- The required annual maintenance on the three shipping casks used for the fuel shipments had been completed within the past year and included the inspections required by the NAC-LWT SAR.

- The maximum weight of the loaded shipping cask and the center of gravity for the cask on the transport trailer was in compliance with the Certificate of Compliance and the SAR.

- The closure lid and test ports were properly inspected, bolted, torqued and tested.

- The shipping cask was vacuum dried and helium backfilled in accordance with the Certificate of Compliance.

- Radiation levels on the shipment were within regulatory limits. The shipment was properly placarded.

Exit Meetings

The inspectors presented the inspection results to members of the licensee management at the conclusion of the various key stages of the inspection on February 11, 2010, July 16, 2010, July 30, 2010 and a final exit on September 21, 2010 after the completion of the third and last shipment being received at Idaho on September 16, 2010. An exit had also been conducted at the Columbiana Hi Tech facility on June 17, 2010.

SUPPLEMENTAL INSPECTION INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

General Atomics

K. Asmussen, PhD., Director, Licensing, Safety and Nuclear Compliance
V. Barbat, Senior Reactor Operator
E. Drees, Program Manager
A. Fudel, Senior Engineer, Quality Assurance
J. Greenwood, Manager, TRIGA Reactor Facility
K. Moore, Radiation Protection Staff
V. Nicolayeff, P.E., Manager, Quality Assurance
P. Pater, CHP, Manager, Health Physics
K. Partain, Senior Quality Engineer
D. Pettycord, Manager, Quality Systems
W. Stout, Senior Reactor Operator
J. Sullivan, Radiation Protection Staff

Columbiana Hi Tech

J. Burchfield, Vice President of Quality
C. Crissman, Senior Project Manager
P. Morris, Level III NDE Inspector

CH2M-WG Idaho

A. Robb, Fuel Examiner
D. Luke, Fuel Examiner

NAC International

D. Schroeder, Project Engineer
M. Miles, Field Operations Manager
R. Bass, Quality Assurance Manager

RRL NDT Consulting, LLC

W. Yauch, Level III Leak Testing

INSPECTION PROCEDURES USED

IP 69002	Class III Research and Test Reactors
IP 69009	Class I Research and Test Reactor Fuel Movement
IP 86001	Design, Fabrication, Testing and Maintenance of Transportation Packages
IP 86740	Inspection of Transportation Activities

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

72-163/1001-01	NCV	Receipt Inspection of Items
72-163/1001-02	NCV	Leak Testing of Sealed Failed Fuel Can

Discussed

None

Closed

72-163/1001-01	NCV	Receipt Inspection of Items
72-163/1001-02	NCV	Leak Testing of Sealed Failed Fuel Can

List of Acronyms

CFR	Code of Federal Regulations
CoC	Certificate of Compliance
C&RS WG	Criticality and Radiation Safety Working Group
CRSC	Criticality and Radiation Safety Committee
DOE	Department of Energy
FE	Fuel Element
FFCR	Fueled Follower Control Rods
FHT	Fuel Handling Tool
FLIP	Fuel Life Improvement Program
FSR	Fuel Storage Rack
GA	General Atomics
IFE	Instrumented Fuel Elements
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
MTU	metric-ton uranium
MWd	megawatt days
NAC	Nuclear Assurance Corporation, Int.
NAC-LWT	Nuclear Assurance Corporation - Legal Weight Truck cask system
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
QA	Quality Assurance
QC	Quality Control
SAR	Safety Analysis Report
SNM	Special Nuclear Material
TRIGA	Training, Research, Isotopes, General Atomics

ATTACHMENT 2

GENERAL ATOMICS FUEL SHIPMENT

INSPECTOR NOTES

(TABLE OF CONTENTS)

GENERAL ATOMICS FUEL SHIPMENT

(Certificate of Compliance # 71-9225)
(INSPECTOR NOTES)

Category: Corrective Actions **Topic:** Corrective Action Program

Reference: 10 CFR 50 Appendix B.XVI

Requirement: Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Finding: The licensee had established a corrective action program that implemented the requirements of 10 CFR Part 50 Appendix B.XVI to identify, track and correct conditions adverse to quality. QAPD-09009, Section 16 "Control of Nonconforming Items" and Section 17 "Corrective Actions" established the requirement to identify, document and correct problems. Section 16 of QAPD-09009 referenced Procedure QP-15 for additional clarifications. Section 17 of QAPD-09009 specified that corrective action requests should be prepared, processed and closed out in accordance with Procedure QDI 16-5 and referenced Procedure QP-16 for guidance in identifying and correcting significant quality problems. Procedure QP-15, Procedure QP-16 and their associated implementing procedures contained in the quality division instructions (QDIs) provided procedures to identify, document and address conditions adverse to quality.

No corrective actions reports were issued from 2004 to 2009, and only one Nonconformance Report (NR-20155) had been issued in 2010 during the TRIGA fuel shipping project as of the July 30, 2010 NRC inspection. Two additional nonconformance reports (NR 21057 and NR 19625) were generated during the July NRC inspection. No trends related to the nonconformance reports could be determined due to the very low number of issues.

Documents Reviewed: (a) Quality Assurance Program Document QAPD-09009 "Site Decontamination and Decommissioning," Issue F (b) Quality Procedure QP-15 "Control of Nonconforming Items," Rev E (c) Quality Procedure QP-16 "Corrective Action," Rev E (d) Quality Division Instructions (QDI) 8-2 "Identification and Control of Items," Issue B (e) Quality Division Instructions (QDI) 10-1 "Receiving Inspection," Issue H (f) Quality Division Instructions (QDI) 14-2 "Control and Use of Inspection, Testing and Operating Status Tags," Issue C (g) Quality Division Instructions (QDI) 15-1 "Nonconformance Report Preparation," Issue J (h) Quality Division Instructions (QDI) 16-4 "Classification of Conditions Adverse to Quality," Issue E (i) Quality Division Instructions (QDI) 16-5 "Corrective Action Request Preparation," Issue D (j) Quality Division Instructions (QDI) 16-6 "Trend Report Preparation," Issue D (k) Quality Division Instructions (QDI) 16-7 "Root Cause Analysis," Issue B (l) Quality Division Instructions (QDI) 16-8 "Maintenance of Corrective Action Request Assignment Log," Issue B (m) Nonconformance Report (NR) 19625 "Acceptance Tags," dated July 29, 2010 (n) Nonconformance Report (NR) 20155 "Sealed Failed Fuel Can Leak," dated July 13, 2010 (o) Nonconformance Report (NR) 21057 "Receipt Inspection Hold Tags," dated July 26, 2010

Category: <u>Crane Operations</u> **Topic:** <u>Base Plate for Cask</u>

Reference: NAC Procedure 315-P-02

Requirement: Lower the cask onto the leveled base plate (seismic or standard, if used) and install the four attachment chains to the four attachment lugs on the cask external surface. At some facilities, the use of the seismic base plate may not be required. NAC will assist the user in determining if a seismic base plate or the standard base plate is appropriate or required for cask loading at a particular facility.

Finding: The standard base plate was used for the GA loading activities. The NAC-LWT cask was securely attached to the base plate with four chains connected to the external lugs of the NAC-LWT cask. The seismic loading plate is only used when the NAC-LWT cask is wet loaded in a spent fuel pool and the site where the loading is occurring is a high seismic site. The seismic base plate ensures the cask will not tip over during a seismic event and possibly damage adjacent fuel stored in the spent fuel pool racks.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14

Category: <u>Crane Operations</u> **Topic:** <u>Crane Specifications/Operations</u>

Reference: N/A

Requirement: The transportation cask will be moved onto the transport vehicle using a mobile crane. Sufficient safety margin shall be applied to the lift.

Finding: The licensee had contracted for a crane from the Marco Crane and Rigging Company that was adequately sized to provide for an acceptable safety margin for lifting the NAC-LWT fully loaded cask. A Groove GMK 5240 truck crane was used with a capacity of 240 tons. Typically, NAC used an 80 ton crane with their NAC-LWT cask loading operations. The 240 ton crane was in very good physical condition and provided additional safety margin for the GA cask loading operations. Since the loaded NAC-LWT cask was conservatively estimated to weigh 52,000 lbs (26 tons) and the lifting yoke weighted 5161 lbs, the GWF 5240 crane rated at 240 tons provided a safety factor of over 8 to 1. The rope was a 22 mm wire rope with four reeves of rope down to the hook. The crane operators assigned to the work activities associated with all three cask loadings were qualified to operate the crane. The crane had been recently inspected and a load test performed as part of renewing the crane certification with the State of California on March 24, 2010.

Documents Reviewed: (a) Daily Inspection Reports for Marco Crane #2171 for August 25 thru August 29, 2010 (b) Owners Periodic (3 month) Inspection Report Required by Cal-OSHA dated, March 24, 2010 (c) Curtis Engineering Corp Annual Inspection Worksheet for the 240 ton Marco Crane #2171 including Load Test Report, dated March 24, 2010 (d) State of California Crane Certificate for Marco 240 ton Crane # 2171, dated March 24, 2010 (e) Transportation Workers Identification Credentials for Joseph C. Allen, David E. Morris, Marshall W. Sanders and Thomas K. Skippon

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Category: Crane Operations **Topic:** Hook Inspection

Reference: ASME B30.10, Sections 10-1.2.1

Requirement: The hook for the spent fuel pool crane shall be inspected on an annual basis under normal service and a record of condition made.

Finding: The spent fuel pool crane hook was inspected during the crane maintenance performed by D & R Crane Inc. and documented in their reports issued January 18, 2010 and July 23, 2010.

Documents Reviewed: (a) D & R Crane Inc. Service Report Job No S-14560, dated July 23, 2010 (b) D & R Crane Inc. Service Report Job No S-14231, dated January 18, 2010

Category: Crane Operations **Topic:** Shipping Cask Weight for Heavy Lift

Reference: NAC-LWT SAR Section 1.2.1.1/SAR Table 2.2.1-1

Requirement: Gross shipping weight of the NAC-LWT spent fuel shipping cask is approximately 52,000 pounds for the package. A summary of overall component weights is detailed in the SAR Table 2.2.1-1.

Finding: The gross shipping weight for each cask loaded at GA was calculated to be less than 52,000 lbs by approximately 900 lbs. The material weight inside each of the GA shipments was 3,095 lbs. The 52,000 lbs referenced in the SAR used a maximum material weight of 4,000 lbs for the material inside the shipping container.

Documents Reviewed: (a) Required Shipper Data (RSD) Form 434.28, "Fuel and Packaging Required Shipper’s Data Form," Rev 4

Category: Crane Operations **Topic:** Spent Fuel Pool Crane

Reference: NUREG 0612, Section 5.1.1 (6)

Requirement: The spent fuel pool crane should be inspected, tested and maintained.

Finding: Maintenance, testing and inspection of the spent fuel pool crane had been recently completed prior to the fuel movement activities. The licensee used the 3 ton crane located in the Mark F Reactor section of Building 21 to load the spent fuel into the fuel baskets. The last two maintenance records for the crane were reviewed, dated January 18, 2010 and July 23, 2010. The maintenance activities were performed by D & R Crane, Inc. The categories of the items tested included electrical power supply, control station, the hoist, the bridge/trolley, the runway and the brake system. A total of 63 items were check for each maintenance inspection. Both maintenance reports found zero discrepancies. The licensee’s crane exceeded the capacity needed to perform the required lifts. The maximum weight that the 3 ton crane would be subjected to was less than 4000 lbs. The inner shield of the intermediate transfer systems (ITS) cask weighed 3,150 lbs and the heaviest fuel basket (top basket) fully loaded with TRIGA fuel was 821 lbs.

Documents Reviewed: (a) D & R Crane Inc. Service Report Job No S-14560, dated July 23, 2010 (b) D & R Crane Inc. Service Report Job No S-14231, dated January 18, 2010 (c) Required Shipper Data (RSD) Form 434.28 "Fuel and Packaging Required Shipper’s Data Form," Rev 4

Category: Emergency Planning **Topic:** Emergency Response Plans

Reference: License R-67 Tech Spec 8.3.b

Requirement: Written instructions shall be in effect for emergency and abnormal conditions including provisions for evacuation, reentry, recovery and medical support.

Finding: The licensee maintained an emergency procedure for the TRIGA facility consistent with the site Radiological Contingency Plan. Procedure TRF-EP-01 included provisions for classifying an emergency; responding to and reporting an emergency; alerting and evacuating affected personnel from the accident area; and reentry and recovery actions. Emergency conditions that were covered in Procedure TRF-EP-01 included fires, medical emergencies, security violations, earthquakes, radiation releases and loss of water from the Mark F pool. An emergency organization with alternates was established that included an emergency response & recovery director, radiological assessment team, emergency response team, security, media coordinator and other supporting specialized personnel. Offsite support had been arranged for security, fire and medical emergencies. Emergency equipment had been pre-staged including radiological survey equipment. Notifications to offsite organizations, including the NRC and State of California were provided for in the phone notification list.

Emergency classifications for an Unusual Event and Alert were described in the emergency procedure. Examples of each level of emergency were provided. For the TRIGA reactor facility, an example of an unusual event that could be related to fuel movement activities was a suspected failure of fuel cladding integrity which could result in the incidental release of radionuclides beyond the TRIGA reactor facility exclusion area. An example of an alert was radiation levels in the TRIGA reactor facility building at levels requiring the emergency evacuation of all personnel from the area. Another example of an alert was the loss of radioactive material control that causes radiation dose rates or airborne radionuclides to increase in ambient radiological exposure levels by a factor of 1000 throughout the TRIGA reactor facility building.

The licensee provided emergency instructions to all personnel associated with the loading of the casks at the pre-job briefing conducted August 27, 2010 for the loading of the first shipping cask. The health physics manager discussed the possible radiological conditions that could occur during the loading activities including a discussion of the highest expected doses from one of the failed fuel cans. The senior reactor operator then discussed the klaxon alarm and showed a diagram of the evacuation routes and the assembly area where everyone was to congregate.

Documents Reviewed: (a) Procedure TRF-EP-01 "Emergency Procedures - TRIGA Reactor Facility," Rev 22

Category: Fabrication Inspection **Topic:** Corrective Action Program During Fabrication

Reference: 10 CFR 71.133

Requirement: Measures shall be established to assure that conditions adverse to quality, such as deficiencies, deviations, defective material and equipment and nonconformances are promptly identified and corrected.

Finding: Columbiana Hi Tech was implementing a corrective action program for the fabrication and testing of the baskets and the failed fuel cans consistent with the requirements in 10 CFR 71.133. Procedure O-11 provided for the identification of concerns that could have a serious effect on safety or operability as a "Significant Condition Adverse to Quality." Less serious nonconformances were identified as "Deviations." Any employee could initiate a "Request for Corrective Action." An example form was provided in the procedure. The request for corrective action was signed by the Department Manager with copies to the Vice President of Quality and the company President. A log was maintained and copies of the corrective action requests (CAR) kept in a binder which was reviewed by the NRC during this inspection. During 2010, no CARs had been issued. In 2009, ten CARs had been issued, none identified as significant. The Vice President of Quality felt that the lack of recent condition reports was due to Columbiana Hi Tech previously performing fabrication activities for General Atomics and NAC, International and having been audited numerous times by them. Corrective actions taken in the past to deal with problems had been successful in preventing repeat issues.

An issue in 2008 had been identified with the fabrication of the TRIGA baskets during an earlier fabrication job. On August 23, 2008, CAR 08-04 had been issued identifying problems with the TRIGA basket guide bar fillet welds and the failed fuel can lid fillet welds. The welds could not be completed as shown on the drawings. Columbiana Hi Tech had contacted NAC and based on verbal directions from NAC had completed the welding instead of waiting for revision and approval of the welding instructions. The traveler indicated welding was done by the drawings instead of by the verbal instructions from NAC. Columbiana Hi Tech took corrective actions, including additional training of the staff, to make employees aware that this was not an acceptable practice.

A review of the 2009 CARs found a variety of issues related to Columbiana Hi Tech work for other clients. These were reviewed to determine the type of issues that were being reported and entered into the corrective action process and how they were being dispositioned. CAR 09-10 related to out-of-dimension tolerances on support rails being built for horizontal storage modules. The tolerances exceeded the dimension requirements by as much as 5/32" on a required 1/4" +/- 1/8" dimension. Disposition was to use-as-is. CAR 09-09 related to a supplier that provided material of width 170 mm instead of 160 mm. This was a container that was undergoing refurbishing. The material was replaced with material of the correct dimension. CAR 09-08 identified that required plating of alternate port cover bolts had not been done. This was identified during the receipt inspection process of the bolts. CAR 09-07 identified that welders performing quality affecting work for a particular job were not qualified to AWS D1.1 code requirements as specified in the contract. The welders were qualified to ASME Section IX instead. The parts that were welded were discarded. Welders are now qualified on both standards.

Overall, reviewing the log of corrective action requests indicated a variety of issues that were being identified and documented in the corrective action system. The Vice President of Quality was very knowledgeable in the corrective action process and the need to identify conditions such that effective corrective actions could be implemented to prevent repeat problems.

Documents Reviewed: (a) Columbiana Hi Tech (CHT) Procedure O-11 "Corrective Action Procedure," dated May 24, 2002 (b) Columbiana Hi Tech Nonconformance Log for the period March 24, 2009 to June 10, 2010 (c) Corrective Action Report (CAR) 09-10 "Dimensions Out-of-Tolerance on DSC Support Rail Assembly (HSM)" dated November 5, 2009 (d) CAR 09-09 "Dimension Discrepancy Not Caught by Vendor," dated October 22, 2009 (e) CAR 09-08 "Inspection of Plating of Alternate Bolt Covers Was Not Done," dated July 28, 2009 (f) CAR 09-07 "Welder Qualifications," dated July 9, 2009 (g) CAR 08-04 "TRIGA Basket Guide Bar Fillet Welds," dated August 23, 2008

Category: Fabrication Inspection **Topic:** Fabrication of Failed Fuel Cans

Reference: NAC-LWT SAR Sect 1.2.3.1.2/Drawing 315-40-86,87,88

Requirement: The failed fuel can is a 3.25-inch outside diameter tube with a 0.065-inch thick wall. The bottom of the failed fuel can included a check valve and drain plug for draining the can. The top of the failed fuel can has a bolted lid which is sealed with a metallic O-ring once the damaged fuel has been loaded. A diaphragm valve on the lid provides for draining, drying, and helium backfilling of the can. The failed fuel can is constructed of austenitic stainless steels as shown on Drawing Nos 315-40-086 (Sealed Failed Fuel Can), -087 (Lid), and -088 (Body).

Finding: The failed fuel cans were fabricated consistent with the description in the NAC-LWT SAR and in accordance with NAC Drawing Nos. 315-40-086, 315-40-087, and 315-40-088. Fabrication was performed at the Columbiana Hi Tech facility in Greensboro, North Carolina. The NRC conducted an inspection of Columbiana Hi Tech on June 16-17, 2010. Columbiana Hi Tech was fabricating failed fuel cans and the baskets for NAC, International for use with the GA TRIGA fuel shipments. Eleven failed fuel cans had been fabricated consisting of: three (3) short cans, serial Nos. 512, 513 and 514, fabrication subassembly No. 315-088-99; and eight (8) long cans, serial Nos. 622 thru 629, fabrication subassembly No. 315-088-98. The "long can" was 47.6" in length and weighted 17 lbs. The "short can" was 32.5" in length and weighted 14 lbs.

The fabrication activities were nearing completion. Testing of a long failed fuel can (s/n #629) was performed during the NRC inspection. Testing included a load test, hydro test, visual examination of the welds and a liquid penetrant examination of the welds. All the observed tests and examinations related to can #629 were satisfactory, though it took two tries for the hydro test because the O-ring did not seal properly the first time. The Columbiana Hi Tech Level 3 NDE Inspector was knowledgeable and experienced in conducting the required tests and the examinations.

The failed fuel cans had been engraved with unique identifiers. A verification that the engraving had not resulted in the minimum wall thickness being exceeded was performed. SAR Drawing 315-40-88 specified a wall thickness of 0.065 inches and referenced ASME Code SA-249. ASME Code SA-249 specified a 10% allowance for the dimension. Purchase Order 20100079 was reviewed and included the specifications for the pipe that had been purchased for constructing the failed fuel cans. The purchase order specified 0.065 inches +/- 10%. An actual piece of the pipe was retrieved from storage and a calibrated micrometer used to measure the thickness. It was measured at several locations on both ends showing a consistent 0.062 inch value. The depth of the engravings were measured and found to be 0.0009 inches resulting in a wall thickness of 0.0611 inches. This falls within the 0.0585 to 0.0715 range required to meet the specifications of the drawing.

Documents Reviewed: (a) NAC-LWT SAR Drawing 315-40-086 "Assembly, Sealed Failed Fuel Can, TRIGA Fuel," Rev 1 (b) NAC-LWT SAR Drawing 315-40-087 "Canister Lid Assembly, Sealed Failed Fuel Can, TRIGA Fuel," Rev 6 (c) NAC-LWT SAR Drawing 315-40-088 "Canister Body Assembly, Sealed Failed Fuel Can, TRIGA Fuel," Rev 2 (d) Columbiana Hi Tech (CHT) Procedure 10010-LT "Load Test Procedure (for TRIGA and Fuel Cans)," Rev 0 (e) CHT Procedure 10010-HYDRO "Pressure Test Procedure," Rev 0 (f) CHT Procedure QIP-VT "Visual Examination Procedure," Rev 2 (g) CHT Procedure 10010-VT "Supplement to QIP-VT Acceptance Criteria Per ASME B&PV Code, Section III, Subsection NG," Rev 0 (h) CHT Procedure QIP-PT "Nondestructive Examination Procedure - Penetrant Inspection," Rev 3 (i) CHT Procedure 10010-PT "Supplement to QIP-PT Acceptance Criteria Per ASME B&PV Code, Section III, Subsection NG," Rev 0

Category: Fabrication Inspection **Topic:** Fabrication of GA Baskets

Reference: NAC-LWT SAR Sect 1.2.3.1.2/Drawing 315-40-70,71,72

Requirement: The TRIGA fuel basket assemblies being fabricated for General Atomics consisted of fifteen modules: three fuel base modules (drawing 315-40-070), nine intermediate modules (drawing 315-40-071), and three top module (drawing 315-40-072). The three intermediate modules were interchangeable, but the base and top modules were required to be in their proper position. Each module has up to seven cells (fuel positions) for loading TRIGA fuel elements or cluster rods. The center cell of each module of the nonpoisoned basket configuration was blocked by a welded stainless steel baffle that prevents loading of that cell. The nonpoisoned configuration was also referred to as the 24-element basket or the 120-element loading, based on the maximum of 120 intact TRIGA fuel elements that may be loaded into the basket in this configuration. The nonpoisoned basket modules are shown in Drawings 315-40-070, -071, and -072.

Finding: The baskets were fabricated at the Columbiana Hi Tech facility in Greensboro, NC consistent with the NAC drawings. Each NAC-LWT shipping cask can hold one top module (basket), three interchangeable intermediate modules and one bottom module. Columbiana had been contracted to produce enough baskets for three NAC-LWT shipping casks. The top module was 48.3" long, the bottom module was 34.7" long and the intermediate modules were 31.5" long. The baskets were non-poison and did not include borated steel plates for criticality control. Fabrication activities related to the fuel basket assemblies were nearing completion at the time of the NRC inspection. The Columbiana personnel involved with the GA basket fabrication were knowledgeable and skilled in fabricating the baskets to the required specifications and the ASME code standards.

Material inspection reports were on file including associated certified material test reports (CMTRs) and other laboratory analyses. The material specifications were traceable back to the vendor design drawing specifications and were in accordance with the design specifications. Final documentation data module packages of base (s/n’s: 19, 20 and 21), intermediate (s/n’s: 158 thru 166) and top (s/n’s: 419, 420 and 421) baskets were reviewed including documentation of the various non-destructive examinations of the baskets. Selected inspection reports and certain welding operations were reviewed. Material controls of each basket assembly were observed during the inspection. Traceability of items was being maintained throughout processing operations. Assemblies were adequately identified as to inspection status and storage areas complied with specified requirements. Item markings were clear and not detrimental.

Documents Reviewed: (a) NAC-LWT SAR Drawing 315-40-070 "Weldment 7 Cell Basket, TRIGA Fuel Base Module," Rev 6 (b) NAC-LWT SAR Drawing 315-40-071 "Weldment 7 Cell Basket, TRIGA Fuel Intermediate Module," Rev 6 (c) NAC-LWT SAR Drawing 315-40-072 "Weldment 7 Cell Basket, TRIGA Fuel Top Module," Rev 6

Category: Fabrication Inspection **Topic:** Quality Assurance During Fabrication

Reference: 10 CFR 71 Subpart H Quality Assurance

Requirement: Quality assurance requirements apply to the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operations, maintenance, repair and modification of components of packaging that are important to safety.

Finding: Quality assurance requirements were adequately applied to important to safety components at the Columbiana Hi Tech facility for the fabrication of the failed fuel cans and the GA baskets. The NRC inspection included review of numerous quality related documents associated with the fabrication process such as shop travelers, weld control records, non-destructive examination (NDE) reports, pressure test records along with activities associated with commercial grade dedication of materials classified as Important to Safety, Category A. The applicable work sequence number, work documents, description of work, and inspector qualifications of the personnel that performed the NDE work was verified as acceptable. Travelers were properly sequenced and included adequately described test documents. Clear instructions were provided that defined the examination to be performed, characteristics to be inspected, acceptance criteria, measurement and test equipment (M&TE) required, test prerequisites and results reporting.

Traceability of all the basket and failed fuel can weldments was maintained as required throughout the inspection process. NDE specifications were adequate. Welding procedure specifications (WPS's) and procedure qualification records (PQR's) met the ASME Section IX requirements. Approved basket weldment filler wire and flux was used as required by and defined in the applicable welding procedure specifications. Procedures and records were reviewed to determine if individuals performing quality-related activities were trained and certified. A sample of personnel training, qualification and certification records of quality control inspectors, NDE personnel and welders was determined adequate. Welder qualification records were appropriately tracked for proficiency as part of the welder qualification tracking system.

Components and materials used on the shop floor for fabrication were traced back to their associated purchase orders and applicable design drawings. In each case, documentation was adequate to verify that the material conformed to the requirements of the associated design drawings. Adequate processes were established to translate design information into procurement documents and to trace the use of the material in the fabrication process. Important to Safety, Category A, purchase orders specific to plate, pipe, weld wire and gasket materials were reviewed. Appropriate quality and technical requirements had been applied. Documentation was available to effectively trace each of the materials selected back to the applicable purchase order and the associated heat/lot numbers. Commercial grade dedication procedures, plans, and reports were reviewed. Adequate documents were established that defined the dedication process including the identification of critical characteristics for acceptance. Verification of the critical characteristics of the material specifications included requirements for inspection, test and source verification. Measurement and test equipment (M&TE) used to support fabrication activities contained calibration stickers and was current for their calibration dates.

A sample of vendor audit reports for companies maintained on the Qualified Vendor List was reviewed. Audits of the companies were conducted in accordance with QA administrative procedures and audit checklist formats. The audit checklists were detailed with respect to items and documents reviewed during the audits.

NACs implementation of its quality assurance plan for fabrication activities at Columbian Hi Tech was adequate. The fabrication processes were assessed to be good, especially with regard to the quality of workmanship and facility housekeeping practices. Overall Columbiana Hi Tech complied with NACs purchase order (#783918, Revision 1) procurement specification in the areas of purchased material, certified material test reports, NDE services and auditor, welder, and inspector qualifications. No discrepancies were identified between the NAC fabrication specification, the SAR and the Certificate of Compliance No. 9225 for the NAC-LWT package as it related to the failed fuel cans and basket assemblies.

Documents Reviewed: (a) Columbiana Hi Tech (CHT) Procedure 10010-LT "Load Test Procedure (for TRIGA and Fuel Cans)," Rev 0 (b) CHT Procedure 10010-HYDRO "Pressure Test Procedure," Rev 0 (c) CHT Procedure QIP-VT "Visual Examination Procedure," Rev 2 (d) CHT Procedure 10010-VT "Supplement to QIP-VT Acceptance Criteria Per ASME B&PV Code, Section III, Subsection NG," Rev 0 (e) CHT Procedure QIP-PT "Nondestructive Examination Procedure - Penetrant Inspection," Rev 3 (f) CHT Procedure 10010-PT "Supplement to QIP-PT Acceptance Criteria Per ASME B&PV Code, Section III, Subsection NG," Rev 0 (g) NAC Task Plan No. 65005300-TP-03 "Fabrication of TRIGA Baskets, Sealed Canisters, etc," Rev 1 (h) NAC Purchase Order to CHT, No. 783918, Revision 1, dated February 8, 2010 (i) NAC Project Plan No. 65005300-PM-01 "Shipment of GA TRIGA Spent Fuel to the Idaho National Laboratory," dated February 8, 2010 (j) NAC Procurement/ Fabrication Specification, No. 315-S-17 "TRIGA Fuel Basket Assemblies, Damaged Fuel Cans and Associated Fuel Handling and Transport Components," Rev 2 (k) Weld Procedure Specification (WPS) No. 08081-101, Rev 6 (l) GTAW/Manual; WPS No. 08081-103, Rev 1 (m) GTAW/Manual, WPS 08082-106, Rev 0 (n) Semi Automatic or Machine GMAW (o) Welder/Operator Qualification Matrix dated June 1, 2010 (p) Liquid Penetrant (expendable materials), Penetrant Lot No. 08D128, expiration date April 2013 (q) Developer Lot No. 8M22K, expiration date December 2010 (r) Cleaner lot No. 8M13K, expiration date December 2010 (s) Document No. 315-Q-02, Attachment A "Quality List for the NAC Legal Weight Truck Cask," Rev 17 (t) Purchase Orders: 20100079 (pipe), 20090372 (weld wire), 20100071 (o-ring), and 20100068 (top and bottom plates) - Category A (u) CHT Certificate of Conformance "1 inch Thick Plate/ASME SA-240 T304, Part No. 315-391-072, Item 8, Heat No. 80K1," dated March 17, 2009 (v) CHT Commercial Grade Dedication Summary, dated February 27, 2008 and Worksheet, dated March 18,2008 for 1 inch thick Plate supplied by Energy & Process (P.O. 20050273) (w) CHT Qualified Vendors List dated June 7, 2010 (x) CHT Commercial Grade Dedication Summary, dated February 26, 2010 and Worksheet dated March 24, 2010 for pipe supplied by Consolidate Pipe & Supply Co. [P.O. 20100079, dated February 26, 2010] (y) CHT Procedure Nos. P-02, "Commercial Grade Materials/Items Including Services" (z) Procedure Q-21 "Qualification and Certification of NDE Personnel" (aa) Data Packages for Basket Weldments: Qty. (3) BU-GSF-911-35-019 thru 021; Qty. (8) BU-GSF-911-32-158 thru 166; and Qty. (3) BU-GSF-911-48-419 thru 421 (bb) Calibration Records for: 5-2000 psi Deadweight tester, ID No. CHT-168; 20,000-lbs Floor Scale, ID No. CHT-541; NAC Failed Fuel Can-External Go gage, ID No. CHT-261; 30-250 in-lb Torque Wrench, ID No. CHT-532; 0-1000PSI Pressure Gage, ID No. CHT-600; 2000-lbs Dynamometer, ID No. CHT-041; 0-250 in/lb Torque Tester, ID No. CHT-176

Category: Failed Fuel Can **Topic:** Failed Fuel Can Drain Plug Torque Values

Reference: NAC-LWT SAR Section 7.1.7

Requirement: After removing the water from the failed fuel can, torque the drain plug to 50 +/- 10 foot-pounds.

Finding: Since the failed fuel was dry loaded, no water removal was required after the fuel element was loaded. As such, the drain plugs were not removed from the failed fuel cans as part of the loading process. The drain plugs were installed at the fabricator shop at Columbiana Hi Tech and torqued to 50 ft-lbs prior to shipment to GA. The drain plugs were installed with a metal furon seal. At GA, the receipt inspection included verification of the 50 ft-lb torque value using a calibrated torque wrench. Procedure 315-P-15, Section 4.1 stated "Verify that the drain plug seal is installed and torqued to 50 ft-lbs." Two torque wrenches were available at GA to perform the torque verification. The licensee provided current certificate of calibrations #12058 and #12042 for the two torque wrenches.

A problem occurred with the drain plug for can # 625 during the first attempt to complete vacuum drying. Can #625 contained fuel element 6415 in a sealed FLIP fuel tube. After approximately 5 hours of vacuum drying the can and unsuccessfully trying to pass the 10 minute test to maintain the pressure increase below 5 torr, the licensee stopped the drying process. The can was slightly pressurized with helium and a soapy bubble test of the sealed areas on the top of the can and on the manifold was performed. No bubbles were detected. The can was then raised out of the shielded work station and the drain plug was tested. The drain plug was found to be leaking. Fuel element 6415 was removed from can #625 and returned to the shielded transfer cask. Dose rates on fuel element 6415 were 100 R/hr at 2". Fuel element 6415 was eventually placed in can #626 and successfully sealed.

An In-House Inspection Plan was developed to examine the drain plug on can #625 to determine the reason for the leak. The torque value of 50 ft-lb was confirmed. The drain plug was removed and the furon metal seal was found to be deformed due to not being properly seated when the drain plug was originally tightened. Threads on the drain plug and the failed fuel can housing were not damaged. A new seal was installed and the drain plug reinstalled and properly torqued. Can #625 was later used and successfully met the vacuum drying criteria. Nonconformance Report 20155 was issued July 13, 2010 documenting the leaking drain plug. All remaining failed fuel cans that had not yet been loaded were vacuum tested to verify no additional cans had the same problem. All remaining cans passed.

Documents Reviewed: (a) NAC Procedure 315-P-15 "NAC-LWT TRIGA Basket Loading Procedure," Rev 2 (b) Instrument Calibration Solutions Certificate of Calibration #12042 for SNAP ON torque wrench Model QC4R400 (serial # 1197500099), dated February 4, 2010 (c) Instrument Calibration Solutions Certificate of Calibration #12058 for SNAP ON torque wrench Model QC2FR75 (serial # 0997600935), dated February 4, 2010 (d) Nonconformance Report (NR) 20155 "Sealed Failed Fuel Can," dated July 13, 2010

Category: Failed Fuel Can **Topic:** Failed Fuel Can Helium Backfill

Reference: NAC-LWT SAR Section 7.1.7

Requirement: After the failed fuel can meets the dryness pressure test of < 5 torr (6.7 mbar) rise in 10 minutes, backfill the can with helium to a pressure of 1 atmosphere (0 psig), +1, -0 psi. Shut and lock the lid diaphragm valve. Disconnect the testing hose from the lid test connection.

Finding: All failed fuel cans were backfilled with helium in compliance with the SAR requirement. The requirement for helium backfill of the failed fuel cans was provided in Procedure 315-P-15, Section 4.14. This section required the can to be backfilled with helium to 1 atmosphere (0 psig, +1, -0 psig). The licensee had purchased four helium bottles with a purity of 99.995%. The certificate of conformance dated April 13, 2010 documented the purity of the four helium bottles. A pressure gauge with a plus 30 psig to minus 30 psig scale was used on the manifold for measuring the helium pressure placed into the cans. NAC had two manifolds, but only one of the two manifolds had a helium pressure gauge. This manifold was always used when filling the helium into the cans. The gauge was labeled as NIST traceable calibrated and referenced to Certificate of Calibration #12804. The calibration on the gauge had been successfully completed on September 28, 2009 (due date of September 28, 2010). Helium backfilling was observed by the NRC on selected cans during the week and compliance with the backfill pressures confirmed.

Documents Reviewed: (a) NAC Procedure 315-P-15 "NAC-LWT TRIGA Basket Loading Procedure," Rev 2 (b) Praxair Certificate of Conformance for Helium Cylinders T367215, 5620858, W215866, and 35060, dated April 13, 2010 (c) Instrument Calibration Solutions Certificate of Compliance #12804 (PO# 744318 Rev 2) for Instrument ID NAC-PG-02 Compound Pressure Gauge, dated April 26, 2010

Category: Failed Fuel Can **Topic:** Failed Fuel Can Inspection

Reference: NAC-LWT SAR Section 7.1.7

Requirement: Examine the failed fuel can body and inspect for damage. Verify that the lid sealing surface is clean and free of defects. Visually verify that the drain plug seal is installed and the drain plug is partially threaded into the drain plug adapter to allow for draining.

Finding: The failed fuel cans were examined by the NAC personnel prior to loading the failed fuel and found to be free of defects and clean with the drain plugs installed. Procedure 315-P-15, Section 4.1 stated "Examine the sealed failed fuel canister body and inspect for damage. Verify the lid sealing surface is clean and free from defects. Verify the drain plug seal is installed and torqued to 50 ft-lbs." The eleven failed fuel cans (ten were used) were examined in accordance with the procedural requirements and temporarily stored in a room near the spent fuel pool ready for use. Immediately prior to use, each failed fuel can was again visually examined and wiped with a moist clean cloth. Since the fuel loading was performed as a dry process as opposed to loading underwater, the drain plugs had already been installed at the fabricator shop and torqued to 50 ft-lbs prior to shipment to the GA site. The receipt inspection at GA by the NAC personnel included a verification that the torquing had been properly done by using a calibrated torque wrench to verify the 50 ft-lb value. Certificate of Calibration #12042 and #12058 for the two calibrated torque wrenches was provided by NAC with calibration dates of February 4, 2010 (calibration due February 4, 2011).

Documents Reviewed: (a) NAC Procedure 315-P-15 "NAC-LWT TRIGA Basket Loading Procedure," Rev 2 (b) Instrument Calibration Solutions Certificate of Calibration #12042 for SNAP ON torque wrench Model QC4R400 (serial # 1197500099), dated February 4, 2010 (c) Instrument Calibration Solutions Certificate of Calibration #12058 for SNAP ON torque wrench Model QC2FR75 (serial # 0997600935), dated February 4, 2010

Category: Failed Fuel Can **Topic:** Failed Fuel Can Lid Examination

Reference: NAC-LWT SAR Section 7.1.7

Requirement: Examine the failed fuel can lid and inspect for damages. Visually verify that the sealing surface is clean and free of defects. Lubricate the lid bolts, install the lid seal and verify that the lid valve is in the open position and the valve lock set screw is retracted.

Finding: The failed fuel can lids were inspected prior to placing onto the failed fuel can and the lid bolts were lubricated with NEVER-SEEZ, which is an acceptable lubricant per Table 7.3-1 of the NAC SAR. The lid valve was open and the valve lock screw was retracted, except in one case. Procedure 315-P-15, Section 4.7 stated "Examine the failed fuel can lid and inspect for damages. Verify the sealing surface is clean and free from debris. Lubricate the lid bolts and install the lid seal. Verify that the lid valve is in the open position and the valve lock set screw is retracted."

During the closure of the lid valve on Can #627 containing fuel element 6372, after vacuum drying and backfilling with helium, the lid valve handle became jammed against the valve lock set screw before the valve was fully closed. The set screw had not been retracted all of the way such that when the lid valve handle was closed, it encountered the set screw and became jammed. Trying to back the set screw out with the tool used for loosening and tightening the set screw did not work. A very good effort by the licensee's staff and the use of a different tool finally relieved the pressure on the screw and the licensee was able to properly position the lid valve and engage the valve lock set screw.

Documents Reviewed: (a) NAC Procedure 315-P-15 "NAC-LWT TRIGA Basket Loading Procedure," Rev 2
(b) NAC-LWT SAR Table 7.3-1 "Bolt and Torque Table," Footnote #1, Rev 41

Category: Failed Fuel Can **Topic:** Failed Fuel Can Lid Seal Clean

Reference: CoC Condition 5(b)(2)(vii)(c)/SAR Sect 7.1.7

Requirement: After loading the damaged fuel into the failed fuel can, visually verify that there is no debris in the lid sealing surface and thread areas.

Finding: The NAC personnel verified that the lid seal area and thread area were clean. Procedure 315-P-15, Section 4.7 stated "Examine the failed fuel can lid and inspect for damage. Verify the sealing surface is clean and free from defect." This section followed the step where the damaged fuel element had been placed inside the failed fuel can.

Documents Reviewed: (a) NAC Procedure 315-P-15 "NAC-LWT TRIGA Basket Loading Procedure," Rev 2

Category: Failed Fuel Can **Topic:** Failed Fuel Can Lid Torque Values

Reference: NAC-LWT SAR Section 7.1.7

Requirement: Install the failed fuel can lid and torque the bolts to 150 +/- 10 inch pounds. Torque any two diametrically opposes bolts first, then torque the remaining two bolts. Complete the torque sequence by verifying the torque of all four bolts in a clockwise direction.

Finding: Lid torquing was performed according to the torque requirements. Alternating bolts were torqued to 150 in-lbs using a calibrated torque wrench. The torquing of all ten failed fuel cans was observed by the NRC inspector. Two calibrated torque wrenches were available from NAC for use in closing the failed fuel can lids. Certificates of Calibration #12802 and #12050 for the two calibrated torque wrenches was provided by NAC with calibration dates of April 13, 2010 (calibration due April 13, 2011) and February 4, 2010 (calibration due February 4, 2011), respectively.

Documents Reviewed: (a) Instrument Calibration Solutions Certificate of Calibration #12802 for SNAP ON torque wrench Model QC2R200 (serial # 1097302502), dated April 13, 2010 (b) Instrument Calibration Solutions Certificate of Calibration #12050 for Proto torque wrench Model 6066CX (serial # DHA55734), dated February 4, 2010

Category: Failed Fuel Can **Topic:** Failed Fuel Can Vacuum Drying

Reference: NAC-LWT SAR Section 7.1.7

Requirement: Prior to backfilling the failed fuel can with helium and sealing it, evacuate the failed fuel can to a pressure below 10 torr (13 mbar) and continue vacuum pumping for 10 minutes. Stop and isolate the vacuum pump and monitor the vacuum pressure for a minimum of 10 minutes. If the pressure rise is > 5 torr (6.7 mbar) in 10 minutes or less, the can is not considered free of water. Repeat the vacuum drying and pressure rise testing until the pressure rise test results in < 5 torr rise in 10 minutes.

Finding: All ten failed fuel cans were vacuum dried in compliance with the SAR requirement. The requirement for vacuum drying the failed fuel cans was provided in Procedure 315-P-15, Sections 4.12 and 4.13. The procedure required the can to be vacuum dried to a pressure below 10 torr and to continue vacuum drying for a minimum of 15 minutes.

Then stop and isolate the vacuum pump and monitor the can pressure for a minimum of 10 minutes. If the pressure rise was less than 5 torr in the 10 minutes, the can was verified as dry of free water. If the pressure exceeded 5 torr in 10 minutes, the vacuum drying process was repeated. The licensee's vendor (NAC) provided a high quality vacuum drying pump and manifold which was capable of evacuating the can to pressures of 0.1 torr during the tests. The pressure readings were displayed on an electronic monitor connected to a pressure sensor on the manifold. The pressure sensor (serial # 96110018W) was calibrated against a NIST traceable standard on November 16, 2010 (due date of November 16, 2011). A second vacuum manifold was also available and used on several failed fuel cans. This pressure sensor (serial # 15797-444/2009W) was calibrated against a NIST traceable standard on April 28, 2010 (due date of April 28, 2011). Drying time from the start of the vacuum pumping until the can passed the 10 minute criteria was typically less than an hour. For the 10 minute test, typically the pressure on the can was around 0.1 - 0.3 torr at the start of the test. Most cans rose less than 0.4 torr during the 10 minute test. However, can #624 rose 4.0 torr and can #622 rose 4.8 torr during the 10 minute test interval.

One can was not successfully dried during the first attempt. Can #625 had a problem vacuum drying and after approximately 5 hours the drying process was stopped. The fuel element was removed and placed back into the shielded transfer cask, then transferred to a new failed fuel can. Examination of Can #625 determined that the seal on the bottom drain plug had not seated properly during installation at Columbiana Hi Tech, the manufacturer for the can. The licensee issued Nonconformance Report (NR) 20155 for the seal problem. The seal on can #625 was replaced and the can was used for the loading of the last fuel element. The vacuum drying test showed a 10 minute rise of 0.4 torr.

Documents Reviewed: (a) NAC Procedure 315-P-15 "NAC-LWT TRIGA Basket Loading Procedure," Rev 2 (b) Pre-Cal Services, Inc Calibration Certificate #11090045 for Leybold Model DI200/DM11 (serial # D961100181 W/20600021119), dated November 16, 2009 (c) Pre-Cal Services, Inc Calibration Certificate #04100105 for Leybold Model PV20/DI 200-S2 (serial # 15797-444/2009 W/15812/A800083010F3), dated April 28, 2010 (d) Nonconformance Report (NR) 20155 "Sealed Failed Fuel Can," date July 13, 2010

Category: Failed Fuel Can **Topic:** Failed Fuel Can Water Removal

Reference: NAC-LWT SAR Section 7.1.7

Requirement: Pressurize the failed fuel can with air or helium to 5-15 psig to remove the water. Continue the purge for at least 5 minutes after bubbles appear from the base of the failed fuel can.

Finding: The ten failed fuel cans were dry loaded resulting in no need to pressurize the failed fuel can to remove water.

Documents Reviewed: None

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Category: Failed Fuel Can **Topic:** Failed Fuel/Fuel Debris Definition

Reference: NAC-LWT SAR Table 1.1-1

Requirement: Damaged TRIGA fuel is fuel with known or suspected clad breach (i.e. cladding defects that permit the release of gases from the interior of the rod and/or allow water intrusion into the clad to fuel gap while submerged.) Fuel debris is damaged fuel that does not maintain its structural integrity, including fuel particles, fuel debris and broken fuel rods.

Finding: Criteria for damaged fuel was included in Procedure PLN-218, Appendix D "Packaging Criteria for Storage at Irradiated Fuel Storage Facility (IFSF)." The criteria was consistent with NRC guidance in Interim Staff Guidance (ISG)-1 and with the criteria established in the NAC-LWT SAR, Table 1.1-1 for damaged TRIGA fuel and TRIGA fuel debris. Procedure PLN-218, Appendix D provided a table of the criteria for when damage to an element required the use of a failed fuel can. Shipment in a failed fuel can was required for elements with (1) greater than pinholes or hairline cracks in the cladding and/or (2) the possibility that further degradation of the fuel is likely under the anticipated normal conditions of handling, packaging, transportation and long term storage.

The fuel examination project conducted in February 2010 resulted in 17 fuel elements classified as damaged. These were (1) #6368 stainless steel FLIP fuel element 70% enriched classified as a leaker, (2) #6326 stainless steel FLIP fuel element 70% enriched classified as de-clad and sealed in a FLIP fuel tube [sometimes referred to as a TRIGA can], (3) #6372 stainless steel FLIP fuel element 70% enriched classified as de-clad and sealed in a FLIP fuel tube, (4) #6415 stainless steel FLIP fuel element 70% enriched classified as de-clad and sealed in a FLIP fuel tube, (5) #1590 aluminum clad 20% enriched with a dent on the bottom thru the cladding, (6) #1676 aluminum clad 20% enriched with numerous dents appearing to go thru the cladding, (7) #LEU-4 stainless steel clad 20% enriched requiring the pressure sensing line to be cut and removed to fit into the cask, (8) #FDT-1E stainless steel clad 93% enriched requiring the pressure sensing line to be cut and removed to fit into the cask, (9) #FDT-2E stainless steel clad 93% enriched classified as de-clad and rubble [this is one of the two bags that had been stored in Drum 53], (10) #FDT-4E stainless steel clad 93% enriched classified as de-clad and rubble [this is the second of the two bags that had been stored in Drum 53], (11) #FDT-7E stainless steel clad 93% enriched requiring the pressure sensing line to be cut and removed to fit into the cask

Six stainless steel clad 1/2" diameter test element were identified as requiring placement in a sealed failed fuel can (FPT-1, FPT-2, FPT-3, RPT-1, LEU-1, LEU-2). These fuel elements were slightly bent which could cause a loading problem and required cropping of the upper portion (tri-flute) to fit into the cask. Four were 93% enriched (the FPT and RPT elements) and two were 20% enriched (LEU-2 and LEU-4).

Documents Reviewed: (a) CH2M-WG Idaho, LLC Procedure PLN-218 "Examination of TRIGA and Other Fuels," Idaho Cleanup Project, CH2M-WG Idaho, LLC," dated May 27, 2008 (b) NAC-LWT Safety Analysis Report (SAR), Rev 41 (c) NRC Interim Staff Guidance (ISG) - 1 "Damaged Fuel and Nuclear Energy Institute (NEI) Protocol for Classification," Rev 2

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Category: Failed Fuel Can **Topic:** Limit of Two Damaged Elements

Reference: CoC 9225 Cond. 5(b)(2)(vii)(c)/SAR Section 7.1.7

Requirement: No more than the equivalent of two (2) fuel elements, or six (6) fuel rod clusters, as damaged fuel or debris are loaded into the failed fuel can. Damaged TRIGA fuel elements or fuel debris (up to two equivalent elements of maximum 1.5 inch diameter) shall be transported in a failed fuel can (one sealed failed fuel can per cell).

Finding: Seventeen failed fuel elements were loaded into failed fuel cans in compliance with the CoC and SAR limitations. Eleven failed fuel cans were purchased by the licensee. Eight were long cans (47.6") and three were short cans (32.5"). Of these, ten were loaded with failed fuel, leaving one long can as an extra. The ten cans used were engraved with serial numbers. The short failed fuel cans were numbered CAN-GSF-130-33-512 thru 514. The long failed fuel cans were numbered CAN-GSF-130-47-622 thru 628. The following material was loaded into failed fuel cans. For the short cans, can #512 contained fuel element FDT-1E (93%), can #513 contained fuel element FDT-7E (93%), and can #514 contained the fuel rubble from storage Drum 53 from fuel elements FDT-2E (93%) and FDT-4E (93%). For the long cans, can #622 contained fuel element LEU-4 (20%), can #623 contained two fuel elements, 1590 (20%) and 1676 (20%) and can #624 contained six 1/2" fuel elements [FPT-1 (93%), FPT-2 (93%), FPT-3 (93%), RPT-1 (93%), LEU-1 (20%) and LEU-2 (20%)]. Can #625 contained fuel element 6368 (70%), can #626 contained fuel element 6415 (70%) in a sealed FLIP fuel tube (sometimes called a TRIGA can), can #627 contained fuel element 6372 (70%) in a sealed FLIP fuel tube and can #628 contained fuel element 6326 in a sealed FLIP fuel tube. Loading of all ten failed fuel cans was observed by the NRC. The licensee's QA representative and the two fuel examiners from CH2M-WG Idaho observed the fuel loading and independently verified and photographed the content of each failed fuel can.

Documents Reviewed: (a) GA Procedure DDP-1.16 "Fuel Element Cropping and Lanyard Installation," Issue A

Category: Fuel Verification/Loading **Topic:** Acceptable Fuel Type

Reference: CoC 9225 Condition 5(b)(1)(vi)(c)/SAR Sect 1.2.3

Requirement: General Atomics TRIGA fuel elements acceptable for loading in the nonpoisoned TRIGA basket shall meet the criteria listed in the CoC Table 5(b)(1)(vi)(c).

Finding: The GA TRIGA fuel elements assigned for shipment in the NAC-LWT shipping cask met the requirements of the CoC Table 5(b)(1)(vi)(c) "Type and Form of Material." General Atomics TRIGA fuel elements included: (a) standard fuel elements - either aluminum clad or stainless steel clad; (b) instrumented fuel elements (IFE) - similar to standard fuel elements (aluminum clad or stainless steel clad), but containing thermocouple instrumentation; and (c) fueled follower control rod elements [FFCR] (aluminum or stainless steel clad) - poison rods followed by fuel in a single tube. The Mark I fuel consisted of 8.5% wt uranium with 20% enriched U-235. Both Aluminum (1100F Al) and stainless steel (304 SS) clad assemblies were used in the Mark I. The Mark F fuel contained 8.5% or 30% wt uranium with U-235 enrichments of 20%, 70% and 93%. The fuel used in the Mark F reactor was stainless steel (304 SS) clad. The Mark III reactor had been decommissioned and it's fuel used with the Mark F reactor. The fuel listed above for the Mark F reactor encompasses the Mark III fuel types.

General Atomics submitted initial loading plans and specific fuel information to NAC related to the fuel stored at the GA facility. NAC reviewed the fuel information against CoC 9225, Revision 55 and confirmed the acceptability of the fuel for use in the NAC-LWT cask in a letter dated April 5, 2010. General Atomics submitted information to DOE related to the fuel and the loading plans. CH2M-WG, as the contractor for DOE, responded on July 8, 2010 approving the planned fuel shipment.

CoC Table 5(b)(1)(vi)(c) limited the fuel elements to 13.2 lbs. The 263 fuel elements ranged in weight from 3.6 kg (7.9 lbs) to 0.4 kg (0.88 lbs). The CoC table limited the maximum element fuel length to 47.74". This is the maximum length the upper basket can hold. The upper basket is the longest basket at 48.3". The bottom basket is shorter at 34.7" and the intermediate baskets are 31.5" long. The fuel elements ranged in length from 23.25" to 47.13". Of the 263 fuel elements, 42 were either of the instrumented (IFE) variety or the fueled follower control rod (FFCR) variety. These fuel elements were too long to fit inside the baskets because of tubes extending beyond the portion of the fuel element's active fuel portion. These fuel elements were cropped to less than the 47.74."

Fuel cladding was either aluminum or stainless steel, with most fuel elements being stainless steel. The CoC table required a minimum clad thickness of 0.01". The GA TRIGA fuel clad thickness ranged from 0.02" to 0.03." Maximum element diameter allowed by the CoC was 1.5". The outside diameter for the GA TRIGA fuel ranged from 0.506" to 1.478." Active fuel length was limited to 15." The GA TRIGA fuel was all at or less than 15."

The CoC established limits on the initial uranium content and the grams of U-235 per fuel element. For fuel enriched up to 95% U-235, the maximum initial uranium content/element was limited to 186 grams with the U-235 limited to 175 gm. For the eleven 93% enriched GA TRIGA fuel elements, the maximum initial uranium content was 186 gm with a maximum U-235 content of 173 gm. For fuel elements enriched up to 71% U-235, the CoC limited the uranium content/element to 198 gm with a U-235 limit of 138 gm. For the 117 GA TRIGA fuel elements enriched to 70%, the highest uranium content was 196.19 gm with a maximum U-235 content of 137.29 gm. For fuel elements enriched up to 25% U-235 with stainless steel cladding, there were two sets of limits specified in the CoC. For fuel with an initial uranium content of 845 gm and U-235 enrichment up to 169 gm, the fuel element could be placed in any basket location. If the fuel element exceed this value up to a limit of 1447 initial uranium content and U-235 enrichment up to 275 gm, the fuel element was limited to the top or bottom baskets and a maximum of three elements were allowed in each cell. Of the 83 stainless steel TRIGA fuel elements enriched to approximately 20%, only one exceeded the 845 gm/169 gm limit. This fuel element (LEU-4) had an initial uranium content of 1361 gm and a U-235 enrichment of 270.78 gm. Fuel element LEU-4 was placed into failed fuel can 622 and assigned to the top module, cell #4 of cask #3. This failed fuel can was the only fuel placed in cell #4. The remaining 82 stainless steel fuel elements had a maximum initial uranium content of 841.52 gm with a U-235 enrichment of 165.90 gm. For the aluminum clad GA TRIGA fuel elements, all had an initial enrichment of approximately 20%. The maximum initial uranium content/element listed in the CoC was 205 gm with a U-235 maximum enrichment of 41 gm. There were 52 aluminum clad TRIGA fuel elements. These had a maximum initial uranium content of 198 gm with a U-235 enrichment of 39 gm.

For all GA TRIGA fuel elements, the limit in the CoC for hydrogen to zirconium ratio was 2.0. For the 263 fuel elements, the ratio was either 1.0 or 1.65. The maximum average burnup limit listed in the CoC for the LEU fuel (i.e. fuel with maximum U-235 content of 25%) was 151,100 MWd/MTU. For the 71% enriched (max) fuel, the burnup limit was 583,000 MWd/MTU and for the 95% enriched (max), the burnup limit was 460,000 MWd/MTU. RSD Form 434.28A provided information for each fuel element for total uranium (in grams) and element burnup (in MW-days). Based on RSD Form 434.28A, calculations were performed by NAC and independently by the NRC inspectors to verify that the GA fuel met the requirements in the CoC Table. Of the LEU (20% enriched) fuel, the maximum burnup was on fuel element 6955 at 121,865 MWd/MTU. For the 70% enriched fuel, the maximum burnup was on fuel element 6407 at 344,812 MWd/MTU. For the 93% enriched fuel, the maximum burnup was on fuel element FPT-2 at 40,241 MWd/MTU. All burnup values were bounded by the CoC limits.

The enrichment values listed in the CoC were met for all the GA fuel. The fuel was either 20%, 70% or 93% enriched. The minimum cooling time listed in the CoC was 90 days. The GA fuel cooling time ranged from 13 years to 46 years.

NAC reviewed the data spread sheets provided by GA for the 263 fuel elements (RSD Form 434.28 and 434.28A) and confirmed that the GA TRIGA fuel planned for the three shipments met the NAC-LWT Certificate of Compliance 9225, Rev 55 requirements. An independent review was also performed by the NRC using these data sheets and also concluded that all the GA TRIGA fuel was encompassed by the specifications in the NAC-LWT Certificate of Compliance.

Documents Reviewed: (a) Letter (DDE:005:JSG:10) from John Greenwood, GA to Dale Luke, CWI for USDOE "Submittal of GA Proposed Shipment Contents Required Shipper Data (RSD) Form 434.30, Revision 0," dated July 14, 2010 (b) Surveillance No PC-59066 "Annual TRIGA Fuel Inventory" TRIGA Project Surveillance Report for Project 2514312 dated June 21, 2010 (c) Required Shipper Data (RSD) Form 434.28 "Fuel and Packaging Required Shipper's Data Form," Rev 4 with Attachment 1 "Fuel Element Description Table," dated July 1, 2010 (d) Required Shipper Data (RSD) Form 434.28A "Fuel Unit Required Shipper's Data Form," Rev 1 (e) GA Procedure DDP-1.16 "Fuel Element Cropping and Lanyard Installation," Rev A (f) Letter (CCN 310470) from Philip Breidenbach, CH2M-WG to Maria Mitchell, DOE entitled "Contract No. DE-AC07-05ID14516-Approval of the General Atomics Fuel and Packaging Required Shippers Data," dated July 8, 2010 (g) Letter from James Adam, NAC International to John Greenwood, GA entitled "NAC-LWT Compliance," dated April 5, 2010 (h) E-mail from Holger Pfeifer, NAC to James Adam, NAC entitled "Documents Requested by NRC," dated August 10, 2010

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Category: Fuel Verification/Loading **Topic:** Damaged Fuel Limited to Top or Bottom Baskets

Reference: NAC-LWT SAR Section 1.2.3.1.1

Requirement: Damaged TRIGA fuel elements and TRIGA fuel debris shall be loaded into a top or base basket module only.

Finding: Damaged TRIGA fuel elements and TRIGA fuel debris was assigned for loading only in the top and bottom basket modules. All damaged fuel had been placed in sealed failed fuel cans. There was a total of ten failed fuel cans requiring placement into the three shipping casks. RSD Form 434.30, Attachment 1 "NAC-LWT Basket Loading Verification Data Sheet - TRIGA Fuel" provided the assigned loading positions for all 15 baskets to be placed into the three shipping casks (five baskets per cask). Failed fuel cans 625, 626, 627 and 628 were assigned to the top module of cask #1. Failed fuel can 623 was assigned to the top module of cask #2. Failed fuel cans 512, 513 and 514 were assigned to the bottom module of cask #3. Failed fuel cans 622 and 624 were assigned to the top module of cask #3. All ten failed fuel cans were appropriately assigned.

Documents Reviewed: (a) Letter (DDE:005:JSG:10) from John Greenwood, GA to Dale Luke, CWI for USDOE "Submittal of GA Proposed Shipment Contents Required Shipper Data (RSD) Form 434.30, Revision 0," dated July 14, 2010

Category: Fuel Verification/Loading **Topic:** Drum 53 HEU

Reference: Procedure DDP-1.19

Requirement: Two previously sectioned high enriched uranium (HEU) fuel elements (FDT-2E and FDT-4E) are stored in Drum 53 in the Mark F Reactor room. The licensee performed an examination and documentation of the condition of these two fuel elements.

Finding: The fuel elements in Drum 53 had been previously inspected in the GA hot cell facility in January 1988. At that time the gamma dose rate for the two loaded and unshielded cans containing the fuel was measured. For the fuel rubble from fuel element FDT-2E, the dose rate was 140 mR/hr at the surface of the can and 5 mR/hr at 1 meter. For the fuel rubble from fuel element FDT-4E, the dose rate was 130 mR/hr at the surface of the can and 5 mR/hr at 1 meter. Both of these fuel elements contained 93% enriched fuel. The fuel examination conducted of Drum 53 during the fuel examination project in February 2010 was intended to verify the content of Drum 53 in the presence of the CH2M-WG fuel examiners and record the condition of the fuel rubble. Procedure DDP 1.19 was used to perform the examination. The aluminum cans containing the fuel from Drum 53 were removed from the drum and transferred to the Mark F fume hood. Only one can at a time was removed from Drum 53 and examined. The cans were opened and the contents removed, placed in plastic bags, labeled, examined and photographed. The fuel elements had originally been cut apart to facilitate a post irradiation examination (PIE) and consisted of rubblized fuel pieces, including some graphite material and zirconium center rods. Their cladding had been removed during the examination process. The pieces from fuel element FDT-2E measured 1.5 R/hr on contact. The pieces from fuel element FDT-4E measured 1 R/hr on contact. The licensee documented the results of the inspection in the TRF Log Book and in Procedure DDP-1.19 Table 1 "Dose Rate Measurements Taken from Drum 53." The fuel was placed back into the aluminum cans and returned to Drum 53.

During the failed fuel can loading the week of July 12, 2010, the aluminum cans were removed from Drum 53. All personnel were removed from the room, except for the individuals directly involved in the work, and the doors to the room were closed. The bags containing the fuel rubble were removed from the cans, placed in the fume hood and the outer plastic bag removed. The fuel debris was then poured from the inner plastic bag into a funnel on top of failed fuel can #514. Paper had been placed around the failed fuel can to catch any spillage and reduce the work to clean-up any loose contamination. The process worked very smoothly, with no contamination spread during the transfer of the fuel debris to the failed fuel can. Building ventilation system radiation readings were unchanged during the transfer.

Documents Reviewed: (a) GA Procedure DDP-1.19 "Inspection of HEU Fuel Elements Stored in Drum 53," Rev A (b) Letter (CCN 309924) from Philip Breidenbach, CH2M-WG Idaho, LLC to Kathleen Hain, DOE-ID "Contract No. DE-AC07-05ID14516 - Fuel Examination Report for California TRIGA, San Diego, California" dated March 2, 2010

Category: Fuel Verification/Loading **Topic:** Fuel Cropping Project

Reference: SAR Sect 1.2.3.1.1

Requirement: Both the aluminum-clad and the stainless steel-clad TRIGA fuel elements are approximately 1.5-inch diameter rods by approximately 30 inches long. The fueled follower control rod (FFCR) elements range in length from 45 inches to 66.5 inches and are cut, as required, to fit the basket length. Instrumented fuel elements (IFE) are identical to standard fuel elements with the exception of thermocouples and wires and lead-out tubing. The lead-out tubing needs to be detached prior to shipment in order for the IFE to fit into a standard element height envelope.

Finding: The 263 fuel elements ranged in length from 23.25" to 47.13." However, the instrumented fuel elements (IFE) and fueled follower control rods (FFCR) had lead out tubes or extension rods extending beyond the fuel element that required removal in order to fit into the shipping baskets. Some fuel elements had triflutes that also required removal. The licensee developed a plan for cropping and placing lanyards on the fuel elements that were too long. Procedure DDP 1.16 was developed. During the period of May 10 thru May 17, 2010, cropping of 36 fuel elements was performed. The 36 did not include all elements requiring cropping. Fuel elements scheduled for placement in a failed fuel can were not cropped until immediately prior to placement in the can during the week of July 12, 2010. There were nine that were cropped and immediately placed into failed fuel cans.

The licensee performed cropping of the IFEs and FFCRs to remove the excess length such that the IFEs and FFCRs would fit in the shipping baskets. A lanyard was attached to facilitate the ability to hook onto the elements and move them in the pool. Each shipping cask contains five baskets consisting of one bottom basket, three middle baskets and one upper basket. The three interchangeable middle baskets were 31.5" in overall length. The lower basket was slightly longer at 34.7". The upper basket was the longest at an overall length of 48.3". The longest length of fuel element that could be placed in the upper basket was 47.74". Of the 263 fuel elements, 42 were either IFEs or FFCRs and 3 were 1/2" diameter non-IFE fuel elements that required cropping to reduce their length to fit into their assigned baskets. Most were assigned to upper baskets, but three (FDT-3E, FDT-5E and FDT-6E) were assigned to a lower basket in cask #1 and were cut to slightly under 32".

The cropping and lanyarding work resulted in 0.028 person-rem to the whole body and 0.367 person-rem to the extremities. The highest individual extremity dose was 0.197 rem.

Documents Reviewed: (a) Work Authorization #3370 "TRIGA Reactor Facility Work Authorization," Appendix B "Dose Measurements of IFEs and FFCRs" dated October 30, 2009 (b) GA Procedure DDP-1.16 "Fuel Element Cropping and Lanyard Installation" Rev A (c) GA Procedure DDP 1.17 "GA TRIGA D & D Project: Dose Measurement of IFEs and FFCRs," Issue A (d) Required Shipper Data (RSD) Form 434.28 "Fuel and Packaging Required Shipper's Data Form," Rev 4

Category: Fuel Verification/Loading **Topic:** Fuel Examination Project

Reference: License R-67 Tech Spec 9.0

Requirement: In preparation for the shipping of the TRIGA spent fuel in the NAC-LWT cask, the licensee and representatives for the Department of Energy performed a visual examination of all the fuel elements to classify them as either intact or damaged. Damaged fuel elements would be placed in sealed failed fuel cans for the shipment. The facility license required that fuel storage, handling, packaging and removal shall be in accordance with approved procedures.

Finding: During February, 2010 an examination of all TRIGA fuel stored at the GA site was conducted to verify the condition of the fuel as intact or damaged. NRC inspectors were onsite and observed the fuel examination activities. Operating records had indicated only one leaker that had been detected during reactor operations. Numerous fuel elements were observed during the fuel examination project to be scratched with surface blemishes. The aluminum elements had more blemishes and scratches than the stainless steel elements.

Handling and movement of the fuel was by the GA operations staff, all of which were currently trained and certified. Standard Operating Procedure - TRIGA Mark F Reactor Facility, Section 10 "Fuel Movement" and Work Authorization #3370 were used for the fuel handling and fuel examination effort. The fuel examination, using a remote camera and a bright underwater light, was performed by qualified fuel examiners from CH2M-WG Idaho, LLC in accordance with Procedure PLN-218. Procedure PLN-218 provided the examination process for the GA TRIGA fuel and had been incorporated into Work Authorization #3370, Appendix E. Training and certification records for the two fuel examiners were current.

The GA fuel handlers would select a fuel element and move the element from it's storage location to a viewing area where the fuel element was placed in a holder. The camera was moved the length of the fuel element. A DVD recording was made of the fuel element examination. The CH2M-WG Idaho fuel examiners would observe the camera images during the movement and would stop and take additional images of any suspected area. The fuel element would then be rotated 120 degree and the process repeated. A third rotation of 120 degree would then be performed and the final set of images taken. A ruler was stationed beside the fuel element such that the location of any areas of concern could be plotted on a paper record. The ruler and camera were able to easily distinguish sizes as small as 1/16 inch.

Criteria for damaged fuel was included in Procedure PLN-218, Appendix D "Packaging Criteria for Storage at IFSI." The criteria was consistent with NRC guidance in Interim Staff Guidance (ISG)-1 "Damaged Fuel and NEI Protocol for Classification," Rev 2 (ADAMS Document: ML71420268). Any pinhole leaks or cracks observed during the fuel examination required the fuel element to be placed in a failed fuel can for shipment.

As each fuel element was examined, the serial number was confirmed. The serial numbers on the stainless steel elements were easy to read. However, the aluminum elements were difficult to read, and in some cases, it took several attempts to decipher the numbers. The aluminum elements were all < 20% enriched. Fuel elements 6326, 6372 and 6415 had been segmented as part of a fuel examination by GA. These fuel elements did not have serial numbers and were stored in cans that were labeled with the serial numbers. The 1/2" fuel elements had serial numbers stamped on the top of the element on the flat side of a tri-flute. A tri-flute is a metal portion providing fins to balance water flow around the fuel element. The tri-flutes on these fuel elements are removed to facilitate packaging for shipment. When the tri-flutes are removed, the serial numbers will no longer be on these fuel elements. All six of the 1/2 fuel elements were scheduled to go into the same failed fuel can. Four were 93% enriched and two were 20% enriched.

A report was issued by CH2M-WG Idaho, LLC on March 2, 2010 detailing the results of the fuel examination. The report provided a summarized listing of the 263 fuel elements. Six were special test series fuel elements with 1/2" diameters. The remaining fuel elements were 1 1/2" in diameter. Fifty-two of the 263 fuel elements were aluminum clad. The remaining, except for two, were stainless steel clad. Two fuel elements (FDT-2E and FDT-4E) were special test series fuel elements that were originally stainless steel clad, but had been cut up or broken into pieces. These two fuel elements were in special cans stored in Drum 53, located in the Mark F spent fuel pool room. All 263 fuel elements were determined to meet the requirements for storage at the DOE facility in Idaho, either as is or when placed in a sealed failed fuel can.

The CH2M-WG Idaho report identified 17 fuel elements that required placement in failed fuel cans. A review of the loading plan for the three casks, as described in RSD Form 434.30, confirmed the fuel elements identified in the CH2M-WG report had been assigned to failed fuel cans. Fuel element 6368 (stainless steel) had been originally identified as a leaker by the licensee and was designated as requiring placement in a sealed failed fuel can. The fuel examination confirmed numerous pits on the fuel element surface. Fuel element 1590E (aluminum) was identified as having a dent on the bottom of the fuel element which breached the cladding. This fuel element was designated as requiring placement in a sealed failed fuel can. Fuel element FDT-1E (stainless steel) appeared to have a crack in the cladding near the top end. This element also had a pressure transducer that required removal, which would leave the element internals open. As such, this fuel element required placement in a sealed failed fuel can. Fuel element FDT-7E (stainless steel) had two 3/4" tubes coming out of the top that required removal to fit in the shipping cask. This may expose the graphite core of the fuel element, and as such, required placement in a sealed failed fuel can.

Fuel elements FPT-1, FPT-2, FPT-3, LEU-1 and LEU-2 were 1/2" stainless steel clad elements which required the tri-flutes to be removed. These were identified for placement into sealed failed fuel cans. Fuel elements 6326, 6372, and 6415 (all stainless steel) had been segmented as part of a fuel examination by the licensee. They were stored in sealed welded cans. The cans containing the fuel elements were identified for placement into sealed failed fuel cans. Fuel element LEU-4 (stainless steel) required the pressure line and thermocouple leads to be cut from the top and required placement in a sealed failed fuel can. Fuel element 1676E (aluminum) was observed to have numerous pitting. This fuel element was identified for placement in a sealed failed fuel can. This accounted for the 17 fuel elements placed in the ten failed fuel cans.

During the fuel examination project, radiation levels for each of the fuel elements were measured underwater using an AMP-200 detector. The reading was taken at midlevel about 4" from the fuel element. The average dose rate at 4" underwater was 523 R/hr with three reading over 1,000 R/hr including fuel element 6950 at 1,056 R/hr, fuel element 6407 at 1,025 R/hr, and fuel element 6408 at 1,024 R/hr. The calculated dose at 1 meter in air based on the 4" underwater reading for the three fuel elements listed above was about 38 R/hr.

Documents Reviewed: (a) CH2M-WG Idaho, LLC Procedure PLN-218 "Examination of TRIGA and Other Fuels," Idaho Cleanup Project, CH2M-WG Idaho, LLC, dated May 27, 2008 (b) "TRIGA Fuel Examiner Training Plan," Rev 3 (c) Course C330L002 "TRIGA Fuel Examiner Requalification Training," Rev 3 (d) Qualification Record for Kalan Robb, current as of Feb. 1, 2010 (e) Qualification Record for Dale Luke, current as of Jan. 4, 2010 (f) "Standard Operating Procedures: TRIGA Mark F Reactor Facility," Rev 5 (g) Work Authorization #3370 "TRIGA Reactor Facility Work Authorization," and Appendices A thru E (h) Letter (CCN 309924) from Philip Breidenbach, CH2M-WG Idaho, LLC to Kathleen Hain, DOE-ID entitled "Contract No. DE-AC07-05ID14516 - Fuel Examination Report for California TRIGA, San Diego, California" dated March 2, 2010 (i) Letter (DDE:005:JSG:10) from John Greenwood, GA to Dale Luke, CWI for USDOE "Submittal of GA Proposed Shipment Contents Required Shipper Data (RSD) Form 434.30, Revision 0," dated July 14, 2010

Category: Fuel Verification/Loading **Topic:** Fuel Limited to Top & Bottom Baskets

Reference: CoC 9225 Condition 5(b)(2)(vii)(c)

Requirement: Loading of TRIGA high enriched uranium (HEU) and low enriched uranium (LEU) fuel elements having >138 gm and >169 gm initial U-235 mass content, respectively, are limited to top and bottom basket modules and up to three rods per basket cell. A minimum of one TRIGA dummy rod per NAC Drawing No. 315-40-085 shall be installed in place of a TRIGA fuel element to limit the maximum number of rods per cell to three.

Finding: The HEU fuel with initial U-235 mass content >138 gm and the LEU fuel with initial U-235 mass content >169 gm were assigned to the top and bottom basket modules with a maximum of three rods (fuel elements) per basket cell. One dummy rod was assigned to the cell except when the fuel was in a failed fuel can. There were seven HEU fuel elements that exceeded the 138 gm limit. All seven were assigned to the bottom basket of cask #3. Of these, FDT-3E, FDT-5E, FDT-6E and a dummy rod were assigned to cell #1. FDT-1E was in sealed failed fuel can 512 and assigned to cell #2. No other fuel was assigned to cell #2. FDT-2E and FDT-4E in sealed failed fuel can 514 (the rubble from Drum 53) were assigned to cell #3 with no other fuel assigned to that cell. FDT-7E in sealed failed fuel can 513 was assigned to cell #5 with no other fuel assigned to that cell. Cell #6 had no fuel assigned to it.

Only one LEU fuel element exceeded the 169 gm limit. LEU-4 in sealed failed fuel can 622 was assigned to the top basket in cask #3. No other fuel was assigned to the same cell (cell #5). The top basket also included the remaining four 93% enriched fuel elements, in can 624, assigned to cell #2.

Documents Reviewed: (a) Letter (DDE:005:JSG:10) from John Greenwood, GA to Dale Luke, CWI for USDOE "Submittal of GA Proposed Shipment Contents Required Shipper Data (RSD) Form 434.30, Revision 0," dated July 14, 2010 (b) Surveillance No PC-59066 "Annual TRIGA Fuel Inventory" TRIGA Project Surveillance Report for Project 2514312 dated June 21, 2010 (c) Required Shipper Data (RSD) Form 434.28 "Fuel and Packaging Required Shipper's Data Form," Rev 4 with Attachment 1 "Fuel Element Description Table," dated July 1, 2010

Category: Fuel Verification/Loading **Topic:** HEU Fuel Storage

Reference: SOP Mark F Sect 10.6.1.2

Requirement: High enriched uranium (HEU) fuel enriched to 93% shall be stored with a minimum of two water filled spaces between elements in the fuel storage racks. No fuel of another enrichment shall be placed in between them at any time. The HEU fuel shall be stored in the east-west fuel storage canal.

Finding: The HEU fuel elements were stored in a special location in the east-west portion of the Mark F spent fuel pool and separated by at least two water filled spaces between the fuel elements. Surveillance No. PC-59066 identified eleven (11) fuel elements as containing HEU. The location of all fuel elements in the spent fuel pool was posted on a large wall board. Nine of the HEU fuel elements were shown on the wall board to be stored in rack NCI-48. These were fuel elements FDT-1E, FDT-3E, FDT-5E, FDT-6E, FDT-7E, FPT-1, FPT-2, FPT-3TC, and RPT-1C. As shown on the wall board, each of the nine HEU fuel elements were placed in locations with two empty spaces between them. During the failed fuel can loading the week of July 12, 2010, the pool storage area for the HEU was open. The NRC Inspector observed that the fuel elements in that location had two empty spaces between them. The other two HEU fuel elements (FDT-2E and FDT-4E) were the rubblized fuel that was stored in Drum 53.

Documents Reviewed: (a) TRIGA Project Surveillance Report for Project 2514312: Surveillance No PC-59066 "Annual TRIGA Fuel Inventory," dated June 21, 2010 (b) Required Shipper Data (RSD) Form 434.28 "Fuel and Packaging Required Shipper's Data Form," Rev 4 with Attachment 1 "Fuel Element Description Table," dated July 1, 2010

Category: Fuel Verification/Loading **Topic:** Limits on Number of Fuel Elements/Cask

Reference: CoC 9225 Condition 5(b)(2)(vii)(c)

Requirement: Up to 120 intact fuel elements are allowed in the TRIGA fuel package with non-poisoned basket. Up to four fuel elements per basket cell are allowed. Loading is allowed only in the six periphery cells. TRIGA fuel elements or sealed cans may not be loaded in the center cell of the non-poisoned basket.

Finding: Each NAC-LWT shipping cask (fuel package) contained 5 fuel baskets. Each basket held up to 24 intact fuel elements (i.e. 6 cells with 4 fuel elements/cell), resulting in a total of 120 fuel elements per shipping cask. For three shipments, a total of 360 fuel elements could be shipped. GA has a total of 263 fuel elements. Three different fuel basket (module) designs were used with the NAC-LWT cask - a base basket, an intermediate basket and a top basket. There were five baskets in each shipping cask: a top basket, bottom basket and three intermediate baskets. The three intermediate baskets are interchangeable, but the base and top baskets are required to be in their proper position. Each basket is designed with seven cells (fuel positions) which can hold up to four fuel elements/cell. For the GA TRIGA fuel, the center cell of each basket is blocked by a welded stainless steel baffle that prevents loading of that cell, resulting in only six usable cells.

RSD Form 434.30, Attachment 1 "NAC-LWT Basket Loading Verifications Data Sheet - TRIGA Fuel" provided the assigned location for the 263 GA TRIGA fuel elements. Cask #1 was assigned 73 intact fuel elements and four failed fuel cans each containing one fuel element. The bottom module was empty. The failed fuel cans were all in the top module. Cask #2 was assigned 111 intact fuel elements and one failed fuel can containing 2 failed fuel elements. The failed fuel can was in the top module. Cask #3 had 62 intact fuel elements with 5 failed fuel cans. Three of the failed fuel cans contained one failed fuel element each. One failed fuel can contained the six 1/2" fuel elements and one failed fuel can contained the rubble from the two rubblized fuel elements that had been stored in drum 53. The failed fuel cans were in the top and the bottom baskets. The center intermediate basket did not contain any fuel elements. For all three casks, this gave a total of 263 fuel elements based on 77 fuel elements in cask #1, 113 fuel elements in cask #2 and 73 fuel elements in cask #3.

Documents Reviewed: (a) NAC-LWT SAR Section 1.2.3.1.2 "TRIGA Fuel Baskets and Damaged Fuel Cans" Rev 41 (b) Letter (DDE:005:JSG:10) from John Greenwood, GA to Dale Luke, CWI for USDOE "Submittal of GA Proposed Shipment Contents Required Shipper Data (RSD) Form 434.30, Revision 0," dated July 14, 2010

Category: Fuel Verification/Loading **Topic:** Maximum Heat Load

Reference: CoC 9225 Condition 5(b)(2)(vii)(c)

Requirement: Maximum decay heat shall not exceed 7.5 watts per TRIGA fuel element (or equivalent for damaged fuel) and 900 watts per package.

Finding: The maximum decay heat had been calculated for all 263 fuel elements. No fuel elements exceeded the 7.5 watt limit and no package (cask) exceeded the 900 watt limit. Calculated decay heat values for the fuel elements were provided on RSD Form 434.28A based on a post irradiation date of October 1, 2010. The highest decay heat for any one fuel element was 0.92 watts. Most fuel elements were below 0.1 watt. Since each cask can hold a maximum of 120 fuel elements, the 900 watt limit is met.

Documents Reviewed: (a) Required Shipper Data (RSD) Form 434.28A "Fuel Unit Required Shipper's Data Form," dated October 19, 2010

Category: Fuel Verification/Loading **Topic:** Spent Fuel Pool Minimum Water Level

Reference: License R-67, Tech Spec 4.2

Requirement: The minimum level of the pool water shall be 12 inches below the reference mark near the top of the pool.

Finding: The water level for the spent fuel pool at the start of fuel handling activities for loading the first cask was at +2.68 inches above the reference mark, well above the minimum water level requirement.

Documents Reviewed: Reading taken by direct observation.

Category: Fuel Verification/Loading **Topic:** Spent Fuel Pool Sampling

Reference: License R-67 Tech Spec 4.4/SOP Mark F Sect 1.3.4

Requirement: On a quarterly basis the pool water shall be sampled and analyzed for the presence of fission products at a level indicating potential fuel leakage. The radioactivity in the pool shall be maintained at a level below 0.1 microCi (μCi) per cubic centimeter (cc) as determined by gamma spectrometry and gross alpha/beta activity concentration.

Finding: Quarterly pool sampling was being performed with pool radioactivity concentrations significantly below the technical specification limits. The 1st and 2nd quarter results for 2010 were selected for review. The radioactivity in the pool was below the 0.1 μCi/cc limit as determined by gamma spectrometry for gross alpha and beta activity concentration. The 1st quarter pool lab sample results were 1.33 x 10(-8) μCi/cc alpha activity and 3.63 x 10(-8) μCi/cc beta activity. The 2nd quarter pool lab sample results were 7.70 x 10(-9) μCi/cc alpha activity and 3.35 x 10(-7) μCi/cc beta activity. The 1st quarter samples were obtained on January 6, 2010. The 2nd quarter samples were obtained on April 7, 2010. These results verify that the fuel examination operations conducted in February 2010 caused no significant increase in fission products to the pool as a result of handling and moving the fuel.

In addition to the periodic pool sampling, a radiation monitor associated with the pool water recirculation system provided ongoing water radiation levels. Readings were typically around 0.2 mR/hr.

Documents Reviewed: (a) Standard Operating Procedure: "TRIGA Mark F Reactor Facility," Rev 5 (b) Standard Operating Procedures: TRIGA Mark F Reactor Facility "Quarterly Maintenance Checklist," Rev 5 for 1st Quarter 2010 dated January 6, 2010 (c) Standard Operating Procedures: TRIGA Mark F Reactor Facility "Quarterly Maintenance Checklist," Rev 5 for 2nd Quarter 2010 dated April 7, 2010 (d) GA Canberra Series 5 XLB alpha-beta lab results dated January 11, 2010 (e) GA Canberra Series 5 XLB alpha-beta lab results dated April 9, 2010

Category: Helium Leak Testing **Topic:** Helium Leak Test Acceptance Standard

Reference: NAC-LWT SAR Section 8.1.3

Requirement: The helium leak test is performed using helium inside the cask cavity and a helium mass spectrometer connected to the test port of the lid or one of the port covers. The mass spectrometer has a minimum sensitivity such that it is capable of detecting a leakage rate of at least 1 x 10(-9) reference cc/sec and is calibrated before and after the test with a standard having a known leakage rate between 4 x 10(-7) and 1 x 10(-9) reference cc/sec. The procedure is performed between 40 degree F and 125 degree F and is temperature corrected.

Finding: The helium leak test was performed on the NAC-LWT cask lid consistent with the acceptance standards specified in the NAC-LWT SAR and ANSI N14.5-1997. NAC Procedure 315-P-12, Section 1.0 required helium leak testing of the NAC-LWT packages in accordance with ANSI N14.5-1997. Section 4.1 required the minimum instrument sensitivity to be 1.0 x 10(-9) std cc/sec. Section 4.2 required a proper helium mass spectrometer leak detection (HMSLD) instrument calibration prior to and after all inspections with a calibrated helium leak source in the range of 4 x 10(-7) to 1 x 10(-9) ref cc/sec.

An Accu-Flow calibrated leak source, GPP-8-He-QF25, serial # 2267 was used to verify proper operations of the helium leak detector. The Accu-Flow leak source was calibrated to a leak rate of 2.58 x 10(-8) atm-cc/sec at 22.8 degree C with a 4% per degree temperature compensation. Calibration due date was November 18, 2010. Section 5.4 of Procedure 315-P-12 required temperature compensation if the ambient temperature was different than the temperature indicated on the calibrated leak source data sheet. A calculation data report form in Appendix B of Procedure 315-P-12 provided the formula for the temperature compensation calculations of the leak source. The temperature at the time of the cask lid leak test for the first cask was 24.5 degree C. The temperature compensated leak rate was calculated to be 2.76 x 10(-8) atm-cc/sec at 24.5 degree C. This value compensated for the difference in temperature between the 22.8 degree C and the 24.5 degree C at a 4%/degree C rate.

An ASM 142D leak detector, serial # MSLD-02 was used to measure the leak rate on the lid for the first cask loaded. The leak detector was required to be within +/- 0.4 x 10(-8) cc/sec of the calibrated leak source. The leak detector measured 2.9 x 10(-8) cc/sec from the calibrated leak source. This was within the 0.4 x 10(-8) cc/sec tolerance for the temperature compensated leak rate value of 2.76 x 10(-8) atm-cc/sec. The sensitivity of the leak detector was calculated to be 0.95 x 10(-10) std cc/sec.

Documents Reviewed: (a) NAC Procedure 315-P-12 "Helium Leakage Testing of the NAC-LWT Cask", Rev 2 (b) Vacuum Technology Certificate of Calibration Report # 2267-ACAL-COMP-1-47453 for the Accu-Flow calibrated leak source, Serial # 2267 (c) RRL NDT Consulting, LLC Certification Record for Walter Yauch as a Level III Leak Tester (d) American National Standards Institute (ANSI) N14.5-1997 "Leak Tests on Packages for Shipment"

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Category: Helium Leak Testing **Topic:** Helium Leak Test on Closure Lid

Reference: CoC 9225 Cond. 9/SAR 8.1.3, 8.1.3.1, Table 8.2-1

Requirement: Perform a helium leakage rate test on the closure lid in accordance with SAR Section 8.1.3.1 "Closure Lid Leakage Rate Test" prior to each loaded transport. The lid seal helium leak rate shall be ≤ 2.0 x 10(-7) cc/sec.

Finding: Cask #2 was the first cask loaded with GA fuel. The cask lid was helium leak tested prior to shipment in accordance with the SAR and met the leak rate requirement. NAC Procedure 315-P-12, Section 8.1 required the closure lid leakage rate test results to be ≤ 2 x 10(-7) cc/sec prior to shipment. Cask #2 (first cask loaded) was leak tested on August 30, 2010 using an ASM 142D leak detector, serial # MSLD-02. The temperature at the time of the test was 24.5 degree C. Test results showed a temperature corrected leak rate of 4.9 x 10(-10) cc/sec. The leak testing was performed by a certified Level III leak test individual with a certification expiration date of May 14, 2012.

Documents Reviewed: (a) NAC Procedure 315-P-12 "Helium Leakage Testing of the NAC-LWT Cask," Rev 2
(b) American National Standards Institute (ANSI) N14.5-1997 "Leak Tests on Packages for Shipment"

Category: Helium Leak Testing **Topic:** Lid/Port Cover Seal Replacement & Helium Leak Test

Reference: CoC 9225 Cond. 9/SAR Sect 8.1.3.2, Table 8.2-1

Requirement: Replace the alternate port cover seal annually and perform a helium leakage rate test. If the alternate port cover Viton O-ring containment face seal is field replaced, such as prior to a shipment, the helium leakage rate test shall be performed. The seal helium leak rate shall be less than or equal to 2.0 x 10(-7) cc/sec.

Finding: The annual replacement of the port cover seals and helium testing of the O-rings was successfully completed for all three NAC casks used for the GA fuel shipments. Procedure 315-P-03 required certain annual maintenance to be performed on the NAC-LWT casks. This included visual inspection of the drain and vent port covers and replacement of the O-ring seals as described in Section 6.11.5. There are two licensed port cover designs used on the NAC-LWT casks. They are referred to as "alternate port covers" and as "alternate B port cover." Each of these designs have both a drain and a vent associated with the port cover. Casks #7 and #8 were qualified to use either the alternate port covers (drain and vent) or the alternate B port covers (drain and vent). Cask #2 was qualified for only the alternate port covers. For the GA TRIGA fuel shipments, only the alternate port covers were required. Procedure 315-P-03, Section 6.12 required helium leak testing on the O-ring seals after the port covers were inspected and the seals replaced.

The NAC-LWT Annual Maintenance Data Package for Casks #2, #7, and #8 documented that the above seal replacement had been performed and listed the new serial part numbers for the newly installed O-rings. The data packages also documented the results of the helium leak tests on the vent and port covers for the three casks. Cask #2 alternate port cover drain helium leak rate was 5.8 x 10(-10) std cc/sec and the vent was 1.0 x 10(-9) std cc/sec. Cask #7 alternate port cover drain was 8.0 x 10(-8) std cc/sec and the vent was 7.1 x 10(-8) std cc/sec. Cask #7 alternate B port cover drain was 1.0 x 10(-10) std cc/sec and the vent was 9.0 x 10(-9) std cc/sec. Cask #8 alternate port cover drain was 3.9 x 10(-8) std cc/sec and the vent was 1.8 x 10(-7) std cc/sec. Cask #8 alternate B port cover drain was 3.0 x 10(-9) std cc/sec and the vent was 1.0 x 10(-10) std cc/sec. All leak tests met the ≤ 2.0 x 10(-7) cc/sec requirement.

Documents Reviewed: (a) NAC Procedure 315-P-03 "NAC-LWT Annual Maintenance Procedure," Rev 21 (b) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #2, dated October 2009 (c) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #7, dated May 2010 (d) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #8, dated May 2010 (e) American National Standards Institute (ANSI) N14.5-1997 "Leak Tests on Packages for Shipment"

Category: Material Accountability **Topic:** Material Balance Report

Reference: 10 CFR 74.13(a)

Requirement: Each licensee possessing special nuclear material in a quantity totaling one gram or more of contained uranium-235, uranium-233 or plutonium shall complete and submit Material Balance Reports concerning special nuclear material that the licensee has received, produced, possessed, transferred, consumed, disposed or lost.

Finding: The licensee had submitted the required material balance reports to the NRC. The licensee's material balance report for the period of October 1, 2009 to March 31, 2010 listed the quantity of inventory for the uranium onsite at the beginning of the reporting period. Record keeping adjustments were made to the GA inventory during this reporting period to account for new calculations that had been performed to determine the End-Of-Life (EOL) composition of the fuel based on the core exposure histories and fuel element core location. The calculations determined that the uranium content should be reduced by 534.16 gms and the U-235 inventory increased by 57.16 gms. These adjustments were factored into the beginning values for the next semiannual material balance report dated April 1, 2010 though September 30, 2010. During the April - September 2010 reporting period, the three fuel shipments were completed, reducing the inventory by 56,585 gms uranium and 18,464 gms U-235. As a result of the three shipments of TRIGA fuel, the inventory onsite as of September 30, 2010 included only material licensed to GA under their California license RML 0145-37. All uranium licensed by the NRC to GA under their SNM license has been removed from the site.

For plutonium, the beginning quantity on the material balance report for the period of October 1, 2009 through March 31, 2010 was reviewed. The End-Of-Life adjustment made during this period was 21.54 gms plutonium added and 15.02 gms Pu-239 subtracted. These adjustments were factored into the beginning values on the next semiannual material balance report dated April 1, 2010 through September 30, 2010. The three fuel shipments included all the plutonium, leaving a balance of zero on the material balance report as of September 30, 2010.

Documents Reviewed: (a) Letter (DDI:001:VJB:09R) from V.J.Barbat, GA to J.S. Greenwood, GA entitled "Justification for TRIGA SNM Inventory Adjustments," dated February 3, 2010 (b) DOE/NRC Form 742 "Material Balance Report," dated from October 1, 2009 to March 31, 2010 for Material Type "50" (Pu) (c) DOE/NRC Form 742 "Material Balance

Report," dated from April 1, 2010 to September 30, 2010 for Material Type "50" (Pu) (d) DOE/NRC Form 742 "Material Balance Report," dated from October 1, 2009 to March 31, 2010 for Material Type "20" (Uranium) (e) DOE/NRC Form 742 "Material Balance Report," dated from April 1, 2010 to September 30, 2010 for Material Type "20" (Uranium)

Category: Material Accountability **Topic:** Nuclear Material Transaction Report

Reference: 10 CFR 74.15(a)

Requirement: Each licensee who transfers or receives special nuclear material (SNM) in a quantity of one gram or more of contained uranium-235, uranium-233 or plutonium shall complete a Nuclear Material Transaction Report. Each licensee who transfers the material shall submit the Nuclear Material Transaction Report to the NRC no later than the close of business the next working day.

Finding: For the first fuel shipment, the licensee had completed the required Nuclear Material Transaction Report and submitted it to the NRC in accordance with the required time frame of no later than the close of business the following work day. The document listed the quantity of SNM that was contained in the first shipment.

Documents Reviewed: (a) DOE/NRC Form 741 "Nuclear Material Transaction Report," dated September 1, 2010

Category: Material Accountability **Topic:** SNM Transfer Record Retention

Reference: 10 CFR 74.19(a)(4)

Requirement: Each record of transfer of SNM to other persons must be retained by the licensee who transferred the material until the Commission terminates the license authorizing the licensee's possession of the material.

Finding: Quality Assurance Program Document QAPD-09009, Section 18 "QA Records," Section 18.4 stated "All records listed in Table 5 shall be maintained for at least five years after the completion of the project, or as otherwise required by 10 CFR 71.51, 10 CFR 74.19 and 40 CFR 268. Surveillance reports for annual inventory of spent TRIGA fuel and spent TRIGA fuel transfer and shipment documents shall be retained for five years or until the TRIGA reactor facility licenses are terminated, whichever is longer." Table 5 "QA Records Retention Responsibilities," listed Spent TRIGA Fuel & Waste Shipment Folders as required records.

Procedure 1013, Section 7.0 "SNM Transactions," stated "Records for SNM transactions, e.g. Receipt and Shipment of SNM, shall be retained pursuant to the requirements of US NRC regulations. In particular, 10 CFR Part 71 "Packaging and Transportation of Radioactive Material" and Part 74 "Material Control and Accountability of SNM." Record retention requirements are specified in 10 CFR 71.91 "Records" which requires records to be retained for a period of 3 years after shipment and in 10 CFR 74.19 "Recordkeeping" which requires records of receipt, acquisition or physical inventory to be retained for 3 years following transfer or disposal of the material. Records of transfers to other persons must be retained until the Commission terminates the license.

Documents Reviewed: (a) Quality Assurance Program Document QAPD-09009 "Site Decontamination and Decommissioning," Issue F (b) Health Physics Procedure 1013 "Records Management and Storage," Issue E

Category: Procedures **Topic:** Field Change

Reference: GA Procedure DDP-1.16

Requirement: Field changes may need to be implemented during the work evolution, as alternate methods may be developed on a case-by-case basis. All field changes shall be documented, as pin & ink changes, and shall be reviewed and approved by the TRIGA reactor facility Physicist-in-Charge (or designee), the GA Health Physics Manager (or designee), and the cognizant Project Quality Engineer (or designee).

Finding: The field change process was not used for a leak test of the drain plug on failed fuel can #625. This has been identified as a Non-Cited Violation (NCV).

Vacuum drying of failed fuel can #625 containing fuel element 6415 in a sealed FLIP fuel tube was not successful during the first attempt. After approximately 5 hours of vacuum drying the can, the licensee stopped the drying process, pressurized the can slightly with helium and conducted a soapy bubble test of the sealed areas on the top of the can and on the manifold. No bubbles were detected. The can was then raised out of the shielded work station and the drain plug was tested. The drain plug was found to be leaking.

The failed fuel was loaded into the failed fuel can in accordance with GA Procedure DDP-1.16 and NAC Procedure 315-P-15. Procedure DDP-1.16, Section 2 "Scope" allowed for deviations from the approved procedure through the use of field changes, and defined the approval process for the field change. Procedure 315-P-15, Section 4.0 "Sealed Failed Fuel Canister Loading Procedures" was used to load, vacuum dry, helium backfill and seal the failed fuel cans. This procedure did not have provisions for testing for a leaking canister or leaking drain plug. As such, a field change in accordance with Procedure DDP-1.16 would be required to perform the leak test. The licensee did not implement the field change process or obtain the required approvals before conducting the leak test. The Mark F reactor Technical Specification 8.3 stated "Written instructions shall be in effect for the following items." Item (c) under this technical specification listed the handling of fuel element or control element with fuel follower as requiring written procedures. Contrary to the requirement in the technical specifications to conduct fuel handling operations in accordance with written instructions, a violation of Procedure DDP-1.16 requirements to use the field change process and obtain the required approvals for any change is identified. Because of the low safety significance of this issue and because it was entered into the licensee's corrective action program, the NRC is treating the violation as Non-Cited Violations (NCV) consistent with Section 2.3.2 of the Enforcement Policy.

Documents Reviewed: (a) GA Procedure DDP-1.16 "Fuel Element Cropping and Lanyard Installation," Issue A
(b) NAC Procedure 315-P-15 "NAC-LWT TRIGA Basket Loading Procedure," Rev 2
(c) "Technical Specifications for TRIGA Mark F Reactor," Facility License No. R-67, dated August 12, 1999

Category: Procedures **Topic:** Level of Procedure Detail/Approvals

Reference: NAC-LWT SAR Section 7

Requirement: NAC-LWT SAR Chapter 7 describes the generic operating procedures for loading, unloading and preparing the NAC-LWT package for transport. These procedures shall be implemented to ensure the package is used in accordance with Certificate of Compliance 9225 for the NAC-LWT packaging. Additional operations and/or modifications to these procedures to address site-specific conditions may be required for each user's facility. These additional operations and/or modifications will be documented in site-specific procedures.

Finding: The NAC and GA procedures used for the fuel shipping project contained an adequate level of detail for conducting operations and had been approved by the appropriate levels of management. Selected NAC and GA procedures were reviewed related to the handling of the spent fuel, loading of the spent fuel into the shipping casks, loading the shipping casks onto the transport vehicle and performing the required surveillances of the shipping cask prior to shipping. Also reviewed were procedures related to the fuel examination effort, fuel cropping and lanyarding effort and placing the damaged fuel into the failed fuel cans. Numerous other supporting procedures were reviewed during the site inspections to verify requirements in the GA TRIGA reactor facility license and requirements in 10 CFR Part 20, Part 50 and Part 71 had been adequately incorporated into procedures such as quality assurance requirements, records requirements, surveillance requirements, etc.

NAC provided procedures for the work activities they were contracted to perform. This included sealing, drying and helium backfilling the failed fuel cans, loading the baskets into the shipping cask, and loading and securing the shipping cask onto the transport vehicle. These procedures had been reviewed and approved by the appropriate NAC organizations such as engineering, operations, licensing and QA, depending on the content and purpose of the procedure. The licensee performed a review of the NAC procedures planned for use at GA to verify compatibility with the GA TRIGA reactor facility license requirements. Licensee procedures were used for fuel movement activities including loading the damaged fuel into the failed fuel cans, loading the spent fuel into the baskets and moving the loaded baskets out of the TRIGA Mark F reactor facility. The licensee also developed an ALARA plan and procedures to support the radiological work by both the GA and NAC personnel.

For both the GA and NAC procedures, the corresponding requirements from the NAC Safety Analysis Report (SAR) were reviewed by the NRC to ensure requirements had been incorporated from the SAR into the appropriate procedures. SAR sections reviewed included: Section 7.1.6 "Procedure for Dry Loading of TRIGA Fuel Basket Modules and GA IFM Modules in the NAC-LWT Cask," SAR Section 7.1.7 "Procedure for Loading TRIGA Damaged Fuel or Fuel Debris into a TRIGA Sealed Damaged Fuel Can," SAR Section 7.2.4 "Procedure for Dry Unloading MTR, TRIGA, DIDO, ANSTO, or PULSTAR Fuel Contents," SAR Section 8.1.3 "Leak Tests," SAR Section 8.1.4 "Component Tests," and SAR Section 8.2 "Maintenance Program." The GA and NAC procedures were found to comply with the requirements in the SAR, however, a number of steps in the SAR sections were not explicitly stated in the NAC procedures, in particular, steps from the SAR unloading section. These differences were discussed with GA management and with NAC.

The licensee developed two main procedures for conducting the fuel movement activities. Procedure DDP-1.16 was developed to remove (crop) extension tubes, conduit and other portions of the fuel elements that extended beyond the fuel region that prevented the fuel elements from fitting into the baskets. Procedure DDP-1.16 was also used for moving the damaged fuel from the fuel pool into the failed fuel cans.

Procedure DDP-1.15 was used in conjunction with the NAC procedures and enveloped all the activities necessary for the fuel loading project from initial start-up of the effort through the placement of the shipping cask onto the transport vehicle and the preparation of the shipping papers. This procedure also included the requirements for the dry run activities conducted prior to the loading of the first cask to ensure equipment was available, heavy lift activities would be smoothly conducted, movement of the baskets into the spent fuel pool were performed safely with minimal risk of adversely affecting the spent fuel in the pool, radiological concerns were identified to the workers and everyone was familiar with the planned activities for successfully loading the shipping casks. The dry run was of particular value in positioning the crane in relationship to the fuel building and the NAC-LWT cask to provide for safe lifting and movement of the fuel while in the individual baskets.

Both of the licensee's procedures used for the fuel loading activities were very thorough and provided a well documented package showing compliance with the NAC-LWT technical specifications and the TRIGA Mark F license and technical specifications. The procedures included detail steps for conducting the operations necessary to perform the cropping and lanyarding of the fuel elements that were too long, loading the damaged fuel into the failed fuel cans, loading the intact fuel and the failed fuel cans into the baskets, tracking the fuel loaded against the approved loading plan and load the baskets into the NAC-LWT cask. Appropriate hold points and QC sign-offs were throughout the procedure.

Documents Reviewed: (a) NAC-LWT Safety Analysis Report (SAR), Rev 41 (b) NAC Procedure 432-P-01 "Dry and Intermediate Transfer System Operating Procedure," Rev 6 (c) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14 (d) NAC Procedure 315-P-03 "NAC-LWT Annual Maintenance Procedure," Rev 21 (e) NAC Procedure 315-P-12 "Helium Leakage Testing of the NAC-LWT Cask," Rev 2 (f) NAC Procedure 315-P-15 "NAC-LWT TRIGA Fuel Basket Loading Procedure," Rev 2 (g) GA Procedure DDP-1.16 "Fuel Cropping and Lanyard Installation," Issue A (h) Standard Operating Procedures: TRIGA Mark F Reactor Facility," Rev 5 (i) GA Procedure DDP-1.15 "Packaging of TRIGA SNF For Transport from GA to INL," Issue A (j) Internal GA correspondence from Paul Pater to Distribution entitled "ALARA Plan for TRIGA Fuel Loading Operations," dated August 20, 2010 (k) Health Physics Procedure No. 16 "Shipment or Transfer of Radioactive Material," Issue AT

Category: Procedures **Topic:** Unloading Procedures

Reference: NAC-LWT SAR Section 7.2

Requirement: Procedures for unloading the fuel shipment at the GA site should be developed consistent with NAC-LWT SAR Section 7.2. These procedures should address removing a loaded cask from the transport vehicle, returning the baskets to the spent fuel pool, removing the fuel assemblies from the baskets and returning them to storage locations in the pool.

Finding: Procedures for unloading the GA spent fuel at the GA TRIGA reactor facility were developed as a contingency should a problem arise with the cask. Instructions for unloading the cask were included in NAC Procedure 315-P-02 and in GA Procedure DDP-1.15. The SAR, Section 7.2.4 "Procedures for Dry Unloading of MTR, TRIGA, DIDO, ANSTO, or PULSTAR Fuel Contents" provided the steps to remove the shipping cask from the transport vehicle and open the shipping cask lid.

NAC Procedure 315-P-02, Section 8.1.4 "Dry Unloading Operations" provided the procedural steps to implement the SAR Section 7.2.4 requirements. The critical unloading steps identified in the SAR were incorporated into Procedure 315-P-02. Fuel unloading at GA would be successfully implemented thru the use of this NAC procedure. However, there was not a one-to-one correlation between the SAR steps and the Procedure 315-P-02 steps. Some of the SAR steps were not reflected in the procedure. These minor discrepancies were discussed with the licensee and with NAC.

NAC Procedure 315-P-02, Section 8.1.4 was designed to position the shipping cask and remove the baskets. The licensee had developed Procedure DDP-1.15 for moving the baskets to the spent fuel pool area, placing the baskets into the spent fuel pool, one at a time, and returning the spent fuel to assigned storage locations in the pool storage racks. This process was described in Section 8.3 of Procedure DDP-1.15. The licensee's fuel handlers would perform any fuel handling operations to place the spent fuel back into the spent fuel pool racks.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14 (b) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transportation from GA to INL," Issue A (c) NAC-LWT Safety Analysis Report (SAR), Rev 41

Category: QA **Topic:** QA Audits

Reference: TRIGA Decommissioning Plan, Sect 1.2.4

Requirement: A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program.

Finding: The licensee was implementing an audit and surveillance program that included the fuel shipping activities. Overall, the audit program was adequately implemented, however several discrepancies were noted: (a) Documentation for the annual visual test for two of the auditors had expired January 7, 2010 and January 12, 2010, respectively. Their re-examination was completed in February 2010 (no specific date provided). Both passed the vision test as documented in the February 2010 examination report, but the reexamination should have been performed prior to their expiration date. No definition of "yearly" was found in the QA program that allowed for a grace period, such as +/- 3 months, that would typically be incorporated into a licensee's definition of "yearly." (b) The 2010 recertification listing for the lead auditors, attached to a January 14, 2010 memo from D. Pettycord, was not dated nor was the training record. The form used for listing the recertified auditors names did not have a form identification number or revision number. (c) The audit checklist for Audit 10001 was not signed as required by Procedure QDI 81-3, Section 11 "Audit Records."

Internal audits were being conducted by the licensee of the decommissioning and decontamination project. Procedure QP 18, Section 3.2 stated "Each element of the GA QA program and each organization involved in safety-related activities shall be scheduled for audit at least once each year or at least once during the life of the activity, whichever is shorter." This requirement was also specified in Procedure QDI 18-4, Section 4.1.1. The decommissioning and decontamination program audit (Audit 10001) performed in March 2010 only reviewed selected criteria and did not cover all 18 criteria of the QA program. Just prior to the spent fuel shipping activities, an internal audit was conducted which reviewed the quality criteria missing from previous audits. The audit was performed from August 24 thru August 26, 2010 and documented in Audit Report No. 10013. No findings or observations were noted.

Documents Reviewed: (a) Quality Division Instruction (QDI) 18-3 "Audit Performance" Issue J (b) Quality Division Procedure (QDI) 18-4 "Maintenance of the Audit Program," Issue B (c) Audit Report 08001 "Audit of the Low Level Radioactive Waste Decommissioning and Decontamination Project," including associated audit checklist, audit plan and audit 08001 Decommissioning and Decontamination Checklist (d) Audit Report 09004 "Audit of the Low Level Radioactive Waste Decommissioning and Decontamination Project," including associated audit checklist, audit plan and audit 09004 Decommissioning and Decontamination Checklist (e) Audit Report 010001, "Audit of the Low Level Radioactive Waste Decommissioning and Decontamination Project," including associated audit checklist, audit plan and audit 10001 Decommissioning and Decontamination Checklist dated March 31, 2010 (f) Audit Report 09009, "Audit of Quality Assurance Division," performed October 27-28, 2009 (g) Quality Assurance Program GA-A13010A (GA-LTR-11) Amendment 15, dated June 2006 (h) Quality Assurance Manual, Rev E (i) Audit Report 09002 " Audit of NAC International," performed March 17-18, 2009 (j) "GA Safety-Related Quality Assurance Approved Suppliers Listing" issued July 12, 2010 (k) Audit Schedules for 2009, dated February 12, 2009 (l) Audit Schedule for 2010, dated February 22, 2010 (m) Quality Procedure QP 18 "Audits," Issue E (n) Memo to Distribution from D. Pettycord entitled "Annual Assessment of Lead Auditor Certification," dated January 14, 2010 (o) Quality Vision Tests for K. Partain dated January 7, 2009 and February 2010 (p) Quality Vision Tests for V. Nicolayeff dated January 12, 2009 and February 2010 (q) Audit Report No. 10013 conducted between August 24 to August 26, 2010

Category: QA **Topic:** Receipt Inspection

Reference: TRIGA Decommissioning Plan, Sect 1.2.4

Requirement: The GA quality assurance program meets the requirements of 10 CFR Part 71 "Transportation of Radioactive Material," Subpart H "Quality Assurance." Subpart H (10 CFR 71.129) requires, in part, that measures shall be established to indicate, by use of markings, such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the package. These measures must provide for the identification of items that have satisfactorily passed required inspections and tests.

Finding: The GA receipt inspection QA procedures were not followed for the initial receipt of the baskets. The yellow "Hold for Inspection" stickers were not placed on the baskets received at the TRIGA reactor facility. This has been identified as a Non-Cited Violation (NCV).

The Mark F Technical Specification 9.0(b) stated that decommissioning shall be performed in accordance with an approved Decommissioning Plan. The GA TRIGA Reactor Facility Decommissioning Plan, Section 1.2.4 "Program Quality Assurance," stated that the currently approved GA quality assurance program meets the requirements of 10 CFR Part 71 "Packaging and Transportation of Radioactive Material," Subpart H "Quality Assurance." Section 1.2.4.1 of the Decommissioning Plan stated that the GA Corporate Quality Assurance Program had been reviewed and accepted by the NRC. The licensee had developed a series of quality procedures (QP) as part of the corporate QA program. In addition to the Corporate QA plan, the Decommissioning Plan stated that a Quality Assurance Program Document (QAPD-09009) had been developed specific to the decommissioning activities for the TRIGA facility. The QAPD had a number of associated implementing procedures, referred to as quality division instructions (QDIs). The QAPD included numerous references not only to the QDI procedures but also to the QP procedures as acceptable means for implementing QA requirements for the TRIGA decommissioning activities.

When equipment and supplies were received for the TRIGA facility that required receipt inspection under the QA program, Procedure QDI 10-1 provided instructions for performing the receipt inspection. Section 2.2.1 of that procedure required a yellow "Hold for Inspection" sticker to be placed on all items that required a receipt inspection and to enter the pertinent information into the Accounts Payable/Purchase Order Millennium data base. After the Project Quality Engineer performed the receipt inspection, Section 2.3.3 of Procedure QDI 10-1 required that an "Accept Tag (GA 1022)" be attached to the item.

In addition to the requirements in Procedure 10-1, Procedure QDI 8-2, Section 3.1.1 also contained the requirement for the yellow sticker to be placed on items waiting inspection and Section 3.1.2 required an acceptance tag be placed on the item after passing receipt inspection until the item was used or consumed.

A receipt inspection plan for the failed fuel cans and the baskets was prepared and issued as Receiving Instruction Plan Production Control #059702. When the 15 baskets to be used with the three NAC-LWT casks were received at GA in crates, the crates were stored at the TRIGA reactor facility, but no yellow "Hold for Inspection" stickers were placed on the crates in compliance with Procedures QDI 10-1 and QDI 8-2. When the eleven failed fuel cans were delivered to the GA site, receipt inspection was performed on the failed fuel cans by GA. However, the original green acceptance tags from the fabricator, Columbiana Hi Tech, were left on the cans without being replaced with GA acceptance tags (GA 1022) in compliance with Procedures QDI 10-1 and QDI 8-2.

Failure to place the yellow stickers on the crates containing the baskets and failure to replace the Columbiana green tags with GA acceptance tags as specified in Procedures QDI 10-1 and QDI 8-2 was initially identified as a violation of quality assurance requirements by the NRC inspectors as they applied to the TRIGA fuel shipment project. The licensee entered these two issues into their corrective action system as Nonconformance Reports (NR) 19625 and 21057.

Upon further discussions between the licensee and the NRC, it was identified that QAPD-09009, Section 11 "Inspection, Observation, Characterization and Radiological Surveys," references Procedure QP-10 and QAPD-09009, Section 15 "Inspection, Testing and Operational Status," references Procedure QP-14. In reviewing the two procedures applicable to this situation, Procedure QP-10, Section 2.6.9 provided that "Accepted items shall be tagged as described in QP-14 until they are placed in stock or issued for use." Since the failed fuel cans were delivered directly to the TRIGA facility, inspected upon receipt, and possession of the cans was immediately transferred to the TRIGA staff, the acceptance tags were not required. The licensee then closed NR 19625.

The remaining issue related to the yellow "hold for inspection" tags remained as a violation of procedures. Because of the low safety significance of this issue and because it was entered into your corrective action program, the NRC is treating this violation as a Non-Cited Violations (NCV) consistent with Section 2.3.2 of the Enforcement Policy.

Documents Reviewed: (a) Quality Assurance Program Document QAPD-09009 "Site Decontamination and Decommissioning," Issue F (b) Document PC-000482/3 "GA TRIGA Reactor Facility Decommissioning Plan," dated July 1999 (c) "Technical Specifications TRIGA Mark F Reactor," Facility License No. R-67, dated August 12, 1999 (d) Quality Division Instruction QDI 8-2 "Identification and Control of Items," Issue B (e) Quality Division Instruction QDI 10-1 "Receiving Inspection," Issue H (f) Quality Procedure (QP)-10 "Inspection," Rev E (g) Quality Procedure (QP)-14 "Inspection, Testing and Operating Status," Rev E (h) Receiving Instruction Plan (RIP) Production Control (PC) #059702 "Failed Fuel Cans & Baskets Supplied by Columbiana Hi Tech" (i) Nonconformance Report (NR) 19625 "Acceptance Tags," dated July 29, 2010 (j) Nonconformance Report (NR) 21057 "Receipt Inspection Hold Tags," dated July 26, 2010

Category: Radiological **Topic:** Air Monitoring - Continuous

Reference: License R-67 Tech Spec 6.2(b)/SOP Mark F Sect 3.2

Requirement: When the facility is not secure, a continuous monitoring system for airborne activity having a readout and audible alarm which can be heard in the facility is required. This system is required to be calibrated annually. The flow meter on the flow totalizer should read 100 +/- 50 liters/minute.

Finding: The continuous air monitoring system was calibrated and operational during the failed fuel activities conducted the week of July 12, 2010 and during the loading of the baskets and movement of the baskets out of the building for the first cask loading August 28-29, 2010. A Ludlum Model 333-4 continuous air monitor (CAM) located in the Mark F control room collected air from the Mark F spent fuel pool room. Typically, the sampling location for the CAM hose was near the wall adjacent to the control room and the spent fuel pool. But when the cropping was performed and when the fuel debris was moved into a failed fuel can, the sampling hose for the CAM was positioned near the work areas to take a representative air sample. The CAM flow rate was set at 65 liters/min. Calibration due date was December 16, 2010. Alarm set points for the Mark F CAM were identified in Emergency Procedure TRF-EP-01, Section 12.1 "Response to MKF CAM Alarm" as 10,000 cpm for an annunciator alert and 35,000 cpm to actuate an alarm. Typical CAM readings throughout the week of failed fuel activities was around 700 cpm with a slightly higher reading of approximately 1,000 cpm at the beginning of the day due to the buildup of naturally occurring radon during the night. Work activities during the day included having the doors open, which reduced the radon levels.

A portable air sampler was also stationed in the Mark F spent fuel pool area adjacent to the work area where the fuel elements were moved from the transfer cask into the failed fuel can. This system was an SAIC Radeco pump and filter, serial number 5131 with a calibration due date of November 9, 2010.

Documents Reviewed: (a) Emergency Procedure TRF-EP-01 "Emergency Procedures - TRIGA Reactor Facility," Rev 22

Category: Radiological **Topic:** ALARA

Reference: 10 CFR 20.1101

Requirement: The licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiological protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA).

Finding: ALARA controls were applied to the activities observed by the NRC during the loading of failed fuel into the failed fuel cans and during the loading of the TRIGA fuel into the NAC-LWT cask for the first fuel shipment. For the loading of the ten failed fuel cans during the period of July 12-16, 2010, no ALARA plan had been developed. Discussions had been held between the operations staff and the health physics staff on the best ways to move the spent fuel from the pool to the failed fuel cans. An efficient layout for moving the spent fuel had been set up and a shielded work area using concrete blocks and lead blankets was provided for the drying and helium backfill work on the failed fuel cans. Access was controlled to the building by a security officer. Everyone in the building had to review and sign the radiation work permit and received a briefing on the precautions necessary while the fuel was being moved.

The work activities in loading the failed fuel cans included lifting the failed fuel elements out of the spent fuel pool while in a shielded transfer cask using the building crane. The cask was moved to a place near the work area where the failed fuel cans had been pre-positioned in a shielded arrangement using concrete blocks and lead blankets. A long handled tool with a hook was then used to lift the fuel element out of the cask and over into the failed fuel can. During the transfer, which took about 15-30 seconds, the fuel element radiation level at 2" was measured and helium gas was blown onto the fuel element to dry it. Two long handled tools were available, one with an approximately three foot handle and one with a six foot handle. Dose rates were taken of the fuel element as it was being transferred using a teletector. Readings of 90 to 100 R/hr at 2" were measured on three of the fuel elements (6326, 6372 and 6415). Fuel element 6368 was expected to be the hottest of the damaged fuel elements, so movement from the cask into the failed fuel can was expediently completed with no delay to take a dose reading or to dry it. After the fuel element was sealed in the failed fuel can and was being moved back to the spent fuel pool using the building crane, a reading using the teletector was performed. The failed fuel can containing fuel element 6368 measured 880 R/hr at 2 inches.

As the work progressed to load the failed fuel cans on the first day, several ALARA improvements were identified. Examples of improvements made during the loading to reduce personnel exposures included: (1) Shield blocks around the failed fuel can work area to shield the workers had been stacked based on the short failed fuel cans which resulted in the long failed fuel cans extending above the shielding. Additional shielding was added to increase the height of the shielded area and additional concrete blocks were added as a second shield wall between the failed fuel cans and the workers. (2) NAC personnel handling the failed fuel cans during the drying and helium backfill were provided finger rings and reduced the amount of time their hands were above the fuel element. (3) The short handling tool had the better attachment for handling the fuel elements and was initially selected over the longer handled tool for handling the fuel elements. The long handling tool was modified and used instead of the short handling tool. (4) The person recording the data for the activities (i.e. radiation readings, fuel element number, failed fuel can number, drying values, etc) was positioned less than 10 feet from where the fuel was being handled. This was appropriate for some of the activities, but much of the time when not needed, the individual had remained at the work station instead of moving back away from the radiation source. The individual began limiting her time at the work station and used a low background area as her waiting area. (5) While the bolts were being placed on the failed fuel can lid and being torqued, the second person held the failed fuel can by hand to keep it steady. Holding the failed fuel can by hand was replaced with the long handling tool. (6) During work activities, individuals had full access to all areas of the room to observe the work. The health physics staff conducted several surveys during the work activities and identified certain areas where the background radiation levels were low. Non-essential personnel were directed to remain in those areas or in the adjacent room and to watch from the doorway. (7) Initially, the senior reactor operator and one of the radiation protection personnel took the lead on handling the fuel elements. After handling a couple of the higher dose rate fuel elements, they began rotating with other staff personnel to reduce their individual radiation dose and spread the dose among several workers.

Throughout the work activities, close attention was paid to the electronic dosimeters and numerous "checks" were performed by the health physics staff to ensure no individual was receiving a high dose during the work activities. The lead operations person and the lead radiation safety person were receiving approximately 50 mR/day on their electronic dosimeters. Health physics coverage was provided by at least two and sometimes three radiation protection personnel using teletectors and other appropriate radiation detection devices. All workers were badged. Personnel working close to the spent fuel with their hands had finger rings. Continuous air monitoring of the room was performed as well as a portable air monitor set up near the work area. Periodic smears of the work areas were performed to ensure no contamination spread was occurring. Personnel leaving the area surveyed out on the hand and foot monitor. Prior to work starting each morning, the Manager of the TRIGA Facility provided a safety briefing related to "lessons learned" from the previous day, work planned for today, safety requirements for the work, radiological conditions expected during the work activities, and a reminder that everyone had "stop work authority" and were expected to exercise that right at anytime an unsafe activity was observed. The health physics staff would provide additional specific radiation information related to the days work, as appropriate.

TLD results for the three and one-half days of loading the failed fuel cans were 114 mrem to the lead operations person, 141 mrem to the lead radiation safety person, 53 mrem to one of the NAC personnel performing drying and helium backfill activities and 57 mrem to the other NAC person. The highest extremity dose to the hands was 436 mrem to one of the NAC personnel and 272 mrem to the other NAC person. The lead operations person received 144 mrem to the hand and the lead health physics person received 228 mrem.

For the loading of the spent fuel into the NAC-LWT casks during August 26-31, 2010, the licensee had developed an ALARA plan dated August 20, 2010. The ALARA plan specified requirements for dosimetry, pre-job ALARA briefings, contamination controls, training, air sampling and area postings. Since there would be a number of observers, including the personnel from the NRC and DOE-Idaho, the licensee set up a remote TV arrangement in a nearby trailer. This allowed personnel to watch loading activities occurring in the pool area and not be in the radiologically controlled area. This was especially beneficial since some of the intact fuel elements were projected to read 20-40 R/hr at 1 meter unshielded. The highest actual reading during the loading of the baskets for the first shipment occurred with basket #5 which contained sealed failed fuel can #625 (failed fuel element # 6368). A reading of 14 R/hr at 1 meter from the failed fuel can was recorded as it was moved from its dry storage location and placed back into the spent fuel pool and into the basket. The hand and foot monitor in the adjacent room alarmed and the ventilation system alarm sounded. Readings as high as 500 mR/hr were measured on the Mark F area radiation monitors. When the basket was removed from the pool inside the interim transfer system (ITS) cask, the radiation level on the ITS side on contact was 1 R/hr.

For the first cask loading August 28-29, 2010, the total of the electronic dosimeter readings was 138 mrem. The highest reading of any individual on August 28 was 16 mrem. The highest on August 29 was 62 mrem.

Documents Reviewed: (a) Internal GA Correspondence from Paul Pater to Distribution entitled "ALARA Plan for TRIGA Fuel Loading Operations," dated August 20, 2010

Category: Radiological **Topic:** Area Radiation Monitor

Reference: License R-67 Tech Spec 6.2(a)/SOP Mark F Sect 3.1

Requirement: An area radiation monitoring system shall be operable capable of activating the evacuation alarm. This system is required to be calibrated annually. A pair of radiation monitors - a low level unit with a 20 mR/hr alarm set point and a high level unit with a 5 R/hr alarm set point are utilized for each facility (Mark F and Mark I).

Finding: The area radiation monitoring system was verified as operational during the failed fuel activities conducted the week of July 12, 2010 and the loading the first cask on August

28-29, 2010. The area radiation monitoring system consisted of five detectors. Detectors MK1-5 and MK1-20 were located in the Mark I reactor room (decommissioned) adjacent to the Mark F control room. Detectors MKF-5 and MKF-20 were located in the Mark F spent fuel pool room above the entrance door. The criticality detector was located on the east wall of the Mark I reactor room. Read-outs for the five detectors were in the Mark F control room. All five detectors were operational and calibrated. Calibration due dates were July 6, 2011 for the MK1-5 and MK1-20 detectors, January 6, 2011 for the MKF-20 detector, June 23, 2011 for the MKF-5 detector and November 13, 2010 for the criticality detector. Normal readings on all five detectors was typically less than 1 mR/hr.

Because the spent fuel elements were lifted out of the pool water to be transferred into the failed fuel cans during the work on July 12, 2010, the alarm set points on the area radiation monitors were set to higher values to avoid alarming. This was acceptable due to the continuous health physics monitoring provided during the work activities. The failed fuel cans, once sealed, were placed in dry storage racks that were shielded. During the loading of the baskets August 28-29, 2010, the failed fuel cans were retrieved from their dry storage area and placed back into the pool to be inserted into the baskets. Once the baskets were filled, they were lifted out of the pool to be moved outside for placement in the NAC-LWT. During this time, the area radiation monitors again required a higher setting to avoid alarming while health physics personnel provided radiation monitoring coverage. Increasing the area radiation monitor alarm set points was performed in accordance with Procedures DDP-1.15 and DDP-1.16.

Procedure DDP-1.16, Attachment A "CWAS Alert/Alarm Settings and Resetting Forms," provided a table of alarm set points for the ARMS. Procedure DDP-1.15, Section 6.1.3, Table 6.1 "Form for Raising of Alert and Alarm Set-Points on the TRF CWAS," and Attachment 13.3 "CWAS Alert/Alarm Set-point Increased," also provided alarm set points. The values in Procedure DDP-1.15 and Procedure DDP-1.16 were identical. The Mark F RAM Low (MKF-20) and the Mark I RAM Low (MK1-20) were normally set at 20 mR/hr. During fuel movement activities, the alarm set point was reset to 1 R/hr. The Mark F RAM High (MKF-5) and Mark I RAM High (MK1-5) were normally set at 5 R/hr. This setting was not changed during fuel movement activities. The criticality detector, normally set at 20 mR/hr, was reset to 1 R/hr during the fuel movement activities. The ARMS also alarm in the security center. Prior to fuel movement activities, the readings on the ARMS ranged between 0.5 to 0.8 mR/hr. The radiation levels outside the building taken during a survey by the NRC inspector averaged 0.01 mR/hr. The last failed fuel can loaded (can #625 with failed fuel element 6368) was one of the hottest fuel elements reading 85 R/hr on the top of the transfer cask coming out of the pool. No survey reading was taken of the fuel element during insertion into the failed fuel can due to ALARA concerns of holding the fuel element in the air while a reading was taken. The readings on the MKF-5 and MKF-20 radiation detectors were 600 mR/hr during the transfer. A reading of 880 R/hr was later measured 2" from the side of the failed fuel can when it was being moved back into the pool.

A klaxon horn was connected to the radiation monitoring system such that an alarm on the criticality detector or a combination of both the 20 mR/hr and 5 R/hr alarm for the MK1 system or the MKF system would activate the klaxton. Emergency Procedure

TRF-EP-01, Section 12.3 "Response to Klaxon Alarm" described the alarm function of the Klaxon. Personnel had been directed during the pre-job briefing that when the klaxton sounded, they were to immediately stop all work activities, evacuate the building and proceed to the assembly area.

Documents Reviewed: (a) Standard Operating Procedure: "TRIGA Mark F Reactor Facility," Rev 5 (b) GA Procedure DDP-1.16 "Fuel Element Cropping and Lanyard Installation," Rev A (c) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transportation from GA to INL," Issue A (d) Emergency Procedure TRF-EP-01 "Emergency Procedures - TRIGA Reactor Facility," Rev 22

Category: Radiological **Topic:** Contamination Levels on Shipment

Reference: 10CFR71.87(i)/49CFR173.443

Requirement: The level of removable radioactive contamination on the external surface of each package offered for shipment is as low as reasonably achievable, and within the limits specified in DOT regulations in 49 CFR 173.443. DOT regulations 49 CFR 173.443 specify a limit of 220 disintegrations per minute (dpm) per square cm beta-gamma and low toxicity alpha and 22 dpm/square cm for all other alpha. The sample shall be taken over a 300 square cm surface with an absorbent material. Sufficient measurements must be taken in the most appropriate locations to yield a representative assessment. Wipe efficiency may be assumed to be 0.10.

Finding: Cask surface contamination survey results were in compliance with 49 CFR 173.443 limits for the first shipment of GA spent fuel. No contamination was found. Procedure DDP-1.15, Section 9.2.18.2 required a contamination survey of the exterior of the shipping cask and the ISO container interior. This survey was performed in accordance with Health Physics Procedure No. 16. Section 11.4 of the procedure required a wipe survey of the loaded vehicle prior to shipment. The wipe survey included multiple survey locations that exceeded the total 300 square cm requirement. Procedure No. 16, Section 9.2 provided additional details concerning the wipes and contained the wording from 49 CFR 173.443 including the limits of 220 dpm/square cm beta-gamma and low toxicity alpha and 22 dpm/square cm for all other alpha. Section 9.2 directed the person taking the surveys to take three 100 square cm swipes of different areas, add the counting results together, then divide by 300 to get the average dpm/square cm value to compare to the shipping limits. The cask was smear surveyed on August 29, 2010 after it was placed in the ISO container. No contamination was found during the surveys of the NAC-LWT cask or the ISO container.

As a precaution, the licensee had performed a thorough smear survey of the cask and associated equipment upon arrival at the GA site to ensure that the cask did not arrive already contaminated. Surveys showed minimal (near background) contamination levels on the cask with most of the surveys showing no residual contamination.

Documents Reviewed: (a) Health Physics Procedure No. 16 "Shipment or Transfer of Radioactive Materials,: Issue AT (b) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transportation from GA to INL, Issue A (c) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14

Category: <u>Radiological</u> **Topic:** <u>External Radiation Levels on Shipment</u>

Reference: 10 CFR71.47/10CFR71.87(j)/ 49CFR173.441

Requirement: For exclusive use shipments, the radiation levels must not exceed the following : 1) 200 mrem/hr on the external surface of the package, unless the shipment is made in a closed transport vehicle, the package is secured within the vehicle so that its position remains fixed during transport, and there are no loading and unloading operations between beginning and end of the transportation, in which case the limit is 1,000 mrem/hr, 2) 200 mrem/hr at any point on the outside surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat bed style vehicle, at any point on the vehicle planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle, and 3) 10 mrem/hr at any point 2 meters from the outer lateral surface of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat bed style vehicle, at any point 2 meters from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle, and 4) 2 mrem/hr in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear dosimetry devices in conformance with 10 CFR 20.1502.

Finding: Radiation levels on the exterior of the shipment were in compliance with 10 CFR 71.47 limits for the first shipment of GA spent fuel. The 1,000 mrem/hr limit applied to the surface of the shipping cask because the NAC-LWT cask was in a closed transport vehicle, the package was secured within the vehicle so its position remained fixed during transport, and there were no loading and unloading operations between the beginning and end of the transportation. NAC Procedure 315-P-02, Section 7.2.2.23 required a radiological surveys of the shipping cask to verify compliance with 10 CFR 71.47 and 71.87(i). Section 7.2.2.26 required a final radiological survey on the exterior to the ISO to verify compliance with 10 CFR 71.87(j), which references 10 CFR 71.47 as the acceptable radiation levels.

The GA Health Physics Procedure No. 16 was used by the radiation protection staff to perform the cask survey. This procedure specified the 10 CFR 71.47 and the 49 CFR 173.441 radiological limits in Section 9.1. Survey forms were included as attachments that provided spaces to record all required survey readings on the transport vehicle.

A preliminary survey of the first shipping cask was completed August 30, 2010 after the cask had been placed inside the ISO container. The shipping cask external dose rate (highest) was 0.4 mrem/hr on contact. The highest reading on the ISO container side surface was 0.15 mR/hr.

The official vehicle survey conducted August 31, 2010 with the ISO and shipping cask loaded on the trailer confirmed the earlier preliminary readings. The highest reading under the trailer was 0.4 mR/hr on contact. No contamination was detected on the ISO container surface.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14 (b) Health Physics Procedure No. 16 "Shipment or Transfer of Radioactive Materials," Issue AT (c) Survey No. TRF-10-027 "Transport Vehicle Survey Form," dated August 31, 2010

Category: Radiological **Topic:** Radiation Protection Program

Reference: License R-67 Sect 9.0(a)/10 CFR Part 20

Requirement: Activities conducted at the facility for monitoring, control of radioactive material and protection of personnel shall be conducted under approved procedures.

Finding: Radiological control activities conducted at the TRIGA reactor facility were performed in accordance with written and approved procedures by a qualified health physics staff. The radiation protection staff consists of a Manager and six health physics staff. The Manager, Health Physics was a Certified Health Physicists. For the fuel loading activities, arrangements were made for additional contract health physics personnel, if needed. During the failed fuel can loading activities and the loading of the NAC-LWT cask, there were always two to three health physic personnel assisting with the work activities. This was an adequate staffing level to control radiological activities.

The radiation safety program was audited annually by the Criticality and Radiation Safety Working Group. The audit reviewed health physics activities conducted at the GA site facilities, exposures to site workers, estimated exposures to the public, status of radiological training of the workers and information related to any accidents or abnormal occurrences. For calendar years 2008 and 2009, no accidents or abnormal occurrences occurred. The quality assurance organization also audited the health physics activities to verify that the radiation safety program was being implemented in accordance with the GA quality assurance plan and applicable regulatory documents.

Health physics controls were adequately implemented during the failed fuel can loading and the loading of the first NAC-LWT cask. Workers were assigned TLD dosimetry, and as appropriate, electronic alarming dosimeters; radiological surveys were continuously being performed in the work area; personnel not directly performing work were kept back away from the radiological sources in low background areas; worker briefings were conducted prior to work; contamination controls were established, though no contamination issues occurred due to the cleanliness of the spent fuel pool water; airborne monitoring was performed; contaminated waste was controlled; radiological postings and control of the radiation areas was established; and appropriate survey instruments were used. Overall, effective health physics controls were established and were adhered to by the workers.

Documents Reviewed: (a) Health Physics Procedure No. 16 "Shipment or Transfer of Radioactive Materials," Issue AT (b) Memo from Criticality & Radiation Safety Working Group (C&RS WG) to Dave Roberts entitled "Annual C&RS WG Audit Reports and ALARA Review for CY 2009," dated May 7, 2010 (c) Memo from Criticality and Radiation Safety Committee (CRSC) to Dave Roberts entitled "Annual CRSC Audit Reports and ALARA Review for CY 2008," dated June 12, 2009

Category: Radiological **Topic:** Ventilation System Operational

Reference: License R-67 Tech Spec 3.5/SOP Mark F Sect 8.1

Requirement: While fuel remains in the facility, air ventilation systems shall be operating to ensure the reactor room is maintained at a negative pressure with respect to the surrounding areas and the exhaust pathway to the environment is controlled and monitored.

Finding: The air ventilation system was operational in the Mark F spent fuel pool area during the loading of the five baskets for the first shipment of GA spent fuel. A negative pressure was maintained in the room. At the start of activities on the first day of fuel loading, the Mark F ventilation monitor was reading 0.06 mR/hr. This reading was relatively constant throughout the work activities, though it did change periodically based on whether the large door had been opened or closed. Opening the door allowed for a reduction in the naturally occurring radon gas in the building. This affect was also seen on the area radiation monitors.

Documents Reviewed: None

Category: Records **Topic:** Fuel Inventory & Transfer Records

Reference: License R-67 Tech Spec 8.5(g)

Requirement: Records and logs shall be prepared for fuel inventories and transfers and retained for a period of at least 5 years.

Finding: Record retention for records associated with fuel inventories and transfers was required for a period of at least 5 years in the Quality Assurance Program Document QAPD-09009, Section 18 "QA Records." Procedure QDI 17-10, Section 2.1 listed document storage locations. QAPD-09009, Table 5 "Quality Assurance Records Retention Responsibilities," listed all records required to be captured that are associated with the QA program. The second column in the table indicates all of the records will meet either dual storage or single storage requirements. In addition, Section 18.5 required "dual copy form media" as the primary storage requirement. If dual storage of a record was not feasible, the record shall be stored in a facility which meets the criteria for a "single facility" in ASME NQA-1.

Documents Reviewed: (a) Quality Assurance Program Document QAPD-09009 "Site Decontamination and Decommissioning," Issue E (b) Quality Division Instruction (QDI) 17-10 "QA Record Retention Period," Issue B

Category: Records **Topic:** SNM Inventory Record Retention

Reference: 10 CFR 74.19(a)(3)

Requirement: Each record of receipt, acquisition or physical inventory of SNM must be retained as long as the licensee retains possession of the material and for 3 years following transfer or disposal of the material.

Finding: The licensee's record retention requirements exceeded the retention time period specified in 10 CFR 74.19(a)(3). Quality Assurance Program Document QAPD-09009, Section 18 "QA Records," Section 18.4 stated "All records listed in Table 5 shall be maintained for at least five years after the completion of the project, or as otherwise required by 10 CFR 71.51, 10 CFR 74.19 and 40 CFR 268. Surveillance reports for annual inventory of spent TRIGA fuel and spent TRIGA fuel transfer and shipment documents shall be retained for five years or until the TRIGA reactor facility licenses are terminated, whichever is longer." Table 5 "QA Records Retention Responsibilities," listed Spent TRIGA Fuel & Waste Shipment Folders as a required record.

Procedure 1013, Section 7.0 "SNM Transactions," stated "Records for SNM transactions, e.g. receipt and shipment of SNM, shall be retained pursuant to the requirements of US NRC regulations. In particular, see 10 CFR Part 71 "Packaging and Transportation of Radioactive Material" and Part 74 "Material Control and Accountability of SNM."

Record retention requirements in 10 CFR 71.91 "Records" specify a period of 3 years after shipment and 10 CFR 74.19 "Recordkeeping" stated records of receipt, acquisition or physical inventory must be retained for 3 years following transfer or disposal of the material, and records of transfers to other persons must be retained until the Commission terminates GA's license.

Documents Reviewed: (a) Quality Assurance Program Document QAPD-09009 "Site Decontamination and Decommissioning," Issue F (b) Health Physics Procedure 1013 "Records Management and Storage," Issue E

Category: Safety Reviews **Topic:** Criticality and Safety Review Committee

Reference: License R-67 Tech Spec 8.2

Requirement: The Criticality and Safety Committee shall review activities of the facility to assure criticality and radiation safety.

Finding: The Criticality and Safety Committee provided an independent overview of site activities related to criticality and radiological safety at the TRIGA reactor facility. The Criticality and Safety Review Committee had recently changed it's name to the Criticality and Radiation Safety Working Group. The Working Group consisted of five members; four of the members acting as auditors. All of the members had appropriate independence and met the Technical Specification (TS) 8.2 requirements of the license. The Working Group met at the periodicity required by TS 8.2 and conducted required audits. The technical specification requirement to have a quorum for meetings was not defined in the group’s charter (as contained in WG-1200.0705); however, all five members had participated in the most recent audit review. The Annual Audit and ALARA Review for calendar year 2009, dated May 7, 2010, included information related to the audit performed at the TRIGA reactor facility on March 15, 2010. The audit was conducted by one of the working group members and was comprehensive. Not all TS 8.2 required functions, such as procedure review, had been captured in the working group’s charter; however, a review of various approved site documents revealed that they were being reviewed and signed by working group members where appropriate.

The annual audits conducted by the Criticality and Radiation Safety Working Group for calendar years 2008 and 2009 were reviewed. Areas covered during the annual audits included work activities at the TRIGA facility, weekly and monthly maintenance activities conducted at the TRIGA facility, completion of required biennial medical exams of the Senior Reactor Operators, health physics activities conducted onsite, occupational doses to workers for the year, implementation of radiological safety training, estimated offsite public exposures, and a review of any accident or abnormal occurrences, which there were none in 2008/2009 related to the TRIGA facility.

Documents Reviewed: (a) CP-1201.2 "Criticality and Radiation Safety Working Group (C&RS WG)," dated September 23, 2009 (b) WG-1200.0705 "Criticality and Radiation Safety Working Group (C&RS WG)," dated April 2010 (c) Memo from C&RS WG to Dave Roberts entitled "Annual C&RS WG Audit Report and ALARA Review for CY 2009," dated May 7, 2010 (d) Memo from Criticality & Radiation Safety Committee (CRSC) to Dave Roberts entitled "Annual CRCS Audit Report and ALARA Review for CY 2008," dated June 12, 2009 (e) Internal GA Correspondence from E. A. Dress to C&RS WG Audit File entitled "C&RS Working Group Audit of the TRIGA Reactor Facility," dated March 15, 2010 (f) Memo from A. Baxter and J. Razvi to C&RS WG Members entitled "C&RS Working Group Annual Audit of Health Physics Activities for CY 2009," dated February 16, 2010 (g) Memo from A. Baxter and A. Bozek to C&RS Working Group Members entitled "2010 Audit of the Nuclear Instrumentation Calibration Laboratory," dated February 17, 2010 (h) Internal GA Correspondence from E. A. Drees to CRSC Audit file entitled "CRSC Audit of the TRIGA Reactor Facility - Inspection Report," dated March 10, 2009 (i) Memo from A. Baxter and A. Bozek to CRSC Members entitled "CRSC Annual Audit of Health Physics Activities for CY 2008," dated May 7, 2009 (j) Memo from A. Baxter and A. Bozek to CRSC Members entitled "2009 Audit of the Nuclear Instrumentation Calibration Laboratory," dated April 9, 2009 (k) Internal GA Correspondence from E. A Drees to CRSC Audit File entitled "CRSC Audit of the TRIGA Reactor Facility - Inspection Report," dated March 10, 2009 (l) Internal GA Correspondence from E. A. Drees to C&RS WG Audit File entitled "C&RS Working Group Audit of the TRIGA Reactor Facility," dated March 15, 2010

Category: Safety Reviews **Topic:** Safety Reviews

Reference: License R-67 Section 2.C.1

Requirement: The licensee may make changes in the facility or procedures described in the Hazards Summary Report and conduct tests or experiments not described in the Hazards Summary Report without Commission approval, unless the proposed changes, tests, or experiments involve a change in the technical specifications incorporated in the license or an unreviewed safety question.

Finding: No safety reviews were performed related to the activities associated with the fuel examinations, cropping and lanyarding and the planned shipment of the spent fuel.

Documents Reviewed: None

Category: Special Topics **Topic:** Current License/Site Status

Reference: N/A

Requirement: The General Atomics' Torry Pines Mesa Site provides research facilities for the development and advancement of the TRIGA (Training, Research, Isotopes, General Atomics) reactor. Currently, all NRC licensed TRIGA reactors are defueled and undergoing decommissioning. Two NRC licensees are in effect at the site.

Finding: The General Atomics Torrey Pines Mesa site, located north of San Diego, California, at La Jolla, California, provided research facilities for the development and advancement of the TRIGA reactor. General Atomics continues to offer TRIGA services and equipment, but has ceased the reactor operations at the Torrey Pines Mesa site with most of the reactor facilities and equipment decommissioned. Three TRIGA reactors had operated at the TRIGA Reactor Facility (TRF) at the Torrey Pines Mesa site. These were the TRIGA Mark I (License #R-38, Docket 50-89), TRIGA Mark F (License #R-67, Docket 50-163) and TRIGA Mark III (License #R-100, Docket 50-227 - terminated). All spent fuel stored onsite (Mark I, Mark F and Mark III) after the reactors were defueled had been stored in the fuel storage canal (spent fuel pool) in the Mark F reactor facility, except for several fuel fragments stored in containers in the Mark F reactor pool area.

The Mark I reactor, initially designated as the Prototype TRIGA reactor and later as the Torrey Pines TRIGA reactor, went critical on May 3, 1958. The reactor was a light water cooled and moderated reactor with a graphite reflector. The reactor operated continuously until late 1997 with a maximum power steady state level of 250 kW (thermal). The integrated power generation during operations was estimated to be 84 MW-days. The Mark I fuel consisted of (U,Zr)H1.0 or (U,Zr)H1.6. Fuel uranium content was 8.5% wt U with 20% enriched U-235. Both aluminum (1100F AL) and stainless steel (304 SS) clad elements were used in the Mark I. On October 29, 1997, the license was changed to a Possession-Only-License (POL) in Amendment #35. Amendment #36 removed the authority from the license to possess special nuclear material. All reactor fuel elements were transferred to the Mark F license and moved from the Mark I reactor pool to the Mark F fuel storage canal. The Mark I reactor facility has been partially decommissioned and dismantled.

The Mark F reactor, initially licensed as the Flashing Advanced Irradiation Reactor (FLAIR) and later as the Advanced TRIGA Prototype Reactor (ATPR), went critical July 2, 1960. The reactor was a light water cooled and moderated reactor with a water reflector. It operated until January 31, 1995. The license was modified March 22, 1995 (effective date April 21, 1995), to a Possession Only License (POL) status in Amendment #43. The maximum steady state power level was 1.5 MW (thermal). The integrated power generation during operations was estimated to be 4,200 MW-days. The Mark F fuel consisted of (U,Zr)H1.6. Fuel uranium content was 8.5% or 30% wt U with U 235 enrichments of 20%, 70% and 93%. The fuel used for the Mark F reactor was stainless steel (304 SS) clad. The Mark F fuel is stored in the Mark F fuel storage canal.

The Mark III reactor was developed to test thermionic direct conversion devices for the production of electricity for specialized purposes. The reactor initially went critical on January 17, 1966 and operated until 1973, when it was decommissioned. The maximum core power rating was 2 MW The license was terminated in 1975. The fuel used in the Mark III reactor was later used in the Mark F reactor and transferred to the Mark F license.

The reactors at GA are Class III Research and Test Reactors in accordance with NRC Manual Chapter 2545, Section 08. Class III reactors are those that are shutdown for extended periods or have a Possession Only License (POL). The spent fuel stored at GA was safeguards Category I fuel as defined in NRC Manual Chapter 2545, Section 10. This rating was due to the amount of fuel and the enrichment of the fuel.

There were 263 fuel elements at GA stored in the Mark F fuel pool (fuel storage canal). Of these, 52 were aluminum clad and 211 were stainless steel clad. The aluminum clad elements were the older elements, all enriched to 20%. The stainless steel clad elements were 20%, 70% and 93% enriched. The fuel elements were 1 ½" in diameter, except for six stainless steel test elements which were ½" in diameter.

During the period of 1980 to 1981, thirteen shipments of spent fuel from the Mark F and Mark I reactors were shipped to DOE-Idaho. Approximately 360 fuel elements were shipped.

Documents Reviewed: (a) Facility License No. R-38 "TRIGA Mark I Research Reactor," Amendment 37, dated July 20, 2001 (ML011570122) (b) Facility License No. R-67 "TRIGA Mark F Research Reactor," Amendment 46, dated July 20, 2001 (ML011570122) (c) Letter from Alexander Adams, Jr, NRC to Dr. Keith Asmussen, GA entitled "GA Mark I and Mark F TRIGA Research Reactors - Amendment RS: Security (TAC Nos. MA9437 and MA9438)," dated July 20, 2001 (ML011570122: Security Related - Not Publically Available) (d) "Technical Specifications for TRIGA Mark I Reactor," Facility License No. R-38, dated August 12, 1999 (e) "Technical Specifications for TRIGA Mark F Reactor," Facility License No. R-67, dated August 12, 1999 (f) Document PC-000482/3 "GA TRIGA Reactor Facility Decommissioning Plan," dated July 1999 (g) Standard Operating Procedure: "TRIGA MARK F Reactor Facility," Rev 5 (h) Standard Operating Procedure: "TRIGA Mark I Reactor Facility," Rev 4 (i) Required Shipper Data (RSD) Form 434.28 "Fuel and Packaging Required Shipper's Data Form," Rev 4

Category: Special Topics **Topic:** Dry Run

Reference: N/A

Requirement: The licensee conducted a dry run practice the day prior to the first cask loading.

Finding: On August 27, 2010, the day prior to beginning the loading of the first cask, the licensee conducted a dry run practice to verify all necessary equipment and supplies were readily available and to work out the fine details of the procedure implementation. The dry run was included in Procedure DDP-1.15, Section 7. The first basket loaded in the interim transfer system (ITS) container was placed in the spent fuel pool, taken out as if it had been loaded with fuel (simulated) and lowered into the shielded transfer cask. The shielded transfer cask, now containing the empty basket was raised to the top of the NAC-LWT cask and the basket was lowered into the NAC-LWT cask. Since the first basket to be placed into the NAC-LWT cask for shipment #1 was planned to be empty, the basket was left in the NAC-LWT cask. The dry run effort provided an opportunity to align the crane with the locations where the NAC-LWT cask, ITS container and shielded transfer cask would be placed. The day's operations went very smoothly and verified the procedures for the movement of the baskets from the spent fuel pool to the NAC-LWT cask.

Throughout the loading campaign, the senior reactor operator in charge of the fuel loading activities would periodically conduct walk throughs of the next set of activities that were planned. This ongoing procedure review ensured that all the workers were following the same procedural steps and knew what each person was responsible for completing. This minimized confusion and provided for a very efficient operation.

Documents Reviewed: (a) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transport from GA to INL," Issue A

Category: Special Topics **Topic:** Transfer of Fuel to DOE

Reference: NRC Letter to GA dated Nov. 4, 2008

Requirement: The licensee, General Atomics, shall transfer the high enriched uranium and low enriched uranium stored at it's site under License Nos R-38 and R-67 to the Department of Energy for storage on behalf of General Atomics.

Finding: In 2000, the NRC began discussions with General Atomics and with the Department of Energy concerning the ongoing storage of TRIGA fuel at the Torry Pines Mesa site in La Jolla, California. Because some of the fuel was enriched to 70% and 93% U-235, and the radiation levels of some of this fuel were not sufficiently high to ensure self protection (i.e. handling of the fuel could be done without heavy shielding due to low radiation levels), the NRC and GA expressed the desire to have the fuel relocated from the GA site to a DOE facility. DOE was currently transferring high enriched uranium TRIGA fuel from university and research facilities throughout the world to a secure DOE storage facility in Idaho. DOE was doing this as part of the Global Threat Reduction Initiative.

Because the TRIGA fuel stored at the GA site was considered privately owned, DOE's acceptance of the fuel for storage required establishing a special contract and compensation paid to DOE for the storage. This was a complicated and lengthy process. In the meantime, GA voluntarily increased security measures for their TRIGA fuel storage area at the La Jolla site. In the fall of 2006, DOE informed the NRC that there may be a path to overcome the legal issues that would be associated with the storage of the GA fuel at a DOE facilities. In early 2008, DOE determined that proceeding under the Atomic Energy Act as opposed to the Nuclear Waste Policy Act could provide a path for acceptance of the fuel by DOE. During June 2008, the NRC participated in a site visit of the GA facility with DOE and representatives of the Global Threat Reduction Initiative (GTRI) of the National Nuclear Security Administration (NNSA). In a classified report, GTRI concluded it was prudent to remove the fuel from the GA site. On June 9, 2008, NRC Chairman Dale E. Klein sent a letter to DOE Secretary Samuel W. Bodman stating that based on the information provided from the GTRI site visit, the NRC felt the TRIGA fuel should be removed from the GA site.

On November 3, 2009, DOE and GA entered into an agreement specifying the terms and conditions for the receipt and storage of the GA research reactor TRIGA fuel at the DOE Irradiated Fuel Storage Facility at the Idaho National Laboratory. All TRIGA fuel was removed from the spent fuel pool at GA between August 28 and September 11, 2010. The third and final shipment of fuel was received at the Idaho Irradiated Fuel Storage Facility on September 16, 2010.

Documents Reviewed: (a) Letter from Charles L Miller, NRC to Dr. Keith Asmussen, GA entitled "Transmittal of Order Related to TRIGA Fuel Management," dated November 4, 2008 [Security Related-Not Publically Available] (b) COMSECY-08-0023 from R. W. Borchardt to the NRC Commission entitled "Update on US NRC Activities to Transfer TRIGA Fuel from General Atomics Site to the U.S. Department of Energy," dated August 1, 2008 [Official

Use Only-Security Related-Not Publically Available] (c) Letter from Dale Klein, NRC to Samuel Bodman, DOE concerning the status of TRIGA fuel at GA (no subject line), dated June 9, 2008 (d) Letter from James R Wade, DOE to Keith Asmussen, GA entitled "Agreement No DE-NE0003 - DOE Idaho Operations Office and General Atomics (OM-OCC-09-084)," dated November 3, 2009 with attached Agreement No. DE-NE0003.

Category: Training & Qualifications **Topic:** Reactor Operator Training & Qualifications

Reference: GA TRIGA Operator Requalification Program

Requirement: The GA TRIGA Facility Reactor Operator Requalification Program requires each operator to perform a minimum of five practice fuel handling manipulations each calendar year and spend a minimum of four hours per calendar quarter participating in routine facility maintenance tasks and/or related activities.

Finding: All three personnel assigned as TRIGA reactor operators were currently trained and qualified. On March 3, 1998, the NRC had approved the GA reactor operator requalification program that had been submitted to the NRC by GA for approval on December 16, 1997, as supplemented on February 24, 1998. The program was reviewed by the NRC against the criteria in 10 CFR Part 55 and found acceptable. The requalification requirements included staying current on applicable procedures, performing at least five practice fuel handling manipulations each calendar year, annual evaluations of the operators knowledge and skills related to the ongoing storage of the TRIGA fuel, and four hours per calendar quarter participating in routine maintenance tasks and/or other related activities. A biennial written exam was also required.

There were currently three personnel at GA that were maintaining their reactor operator qualifications in accordance with the NRC approved training program. Selected records for all three operators were reviewed including quarterly documentation of the four hour maintenance activities and the five fuel handling manipulations, biennial requalification exams, annual evaluations and the decommissioning training matrix which listed all the training that had been completed by the operators including the radiation worker training. Observations made by the NRC inspectors during the loading activities confirmed that the reactor operators were well qualified and skilled to perform the work associated with handling spent TRIGA fuel.

Documents Reviewed: (a) GA TRIGA Reactor Facility Reactor Operator Requalification Program, dated February 24, 1998 (b) Letter from Alexander Adams, Jr., NRC to Dr. Keith Asmussen, GA entitled "GA Reactor Operator Requalification Program," dated March 3, 1998 (c) TRIGA Reactor Facility Annual Reactor Operator Evaluations Operator Requalification Examinations, dated June 2010 for J. S. Greenwood, William Stout, and Virgil J. Barbat (d) TRIGA Reactor Facility Quarterly Maintenance Checklist, dated April 2009 and April 2010 (e) Decommissioning Training Matrix, dated July 25, 2010 (f) TRIGA Reactor Facility Annual Reactor Operator Evaluation Form, dated September 23, 2009 from W. A. Stout and V. J. Barbat (g) TRIGA Reactor Facility Quarterly Maintenance Checklist (Figure 4.3 of the Standard Operating Procedure for TRIGA Mark F Reactor Facility), Rev 5 for 4Q09, 1Q10, 2Q10 and 3Q10

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Category: Transportation **Topic:** Advanced Notification to States

Reference: 10CFR71.97

Requirement: Each licensee shall provide advanced notification to the Governor of a State, or the Governors designee, of the shipment of licensed material through, or across the boundary of the State, before the transport. Notification must be made in writing. If delivered by mail, the notification must be postmarked at least 7 days before the beginning of the 7 day period during which departure of the shipment is estimated to occur. If notification is other than by mail, the notification must reach the office of the Governor or the Governor's designee at least 4 days before the beginning of the 7 day period.

Finding: The required advanced notifications were made to the responsible state agencies for the states in which the shipment passed through. This included California, Arizona, Nevada and Idaho. The first shipment was made September 1, 2010. The state notifications had been made on August 17-18, 2010, which was 7 days prior to the beginning of the 7 day period in which the shipment is estimated to depart. The original estimated departure date was August 30-31.

Documents Reviewed: (a) Letter from Secure Transport Services to California Highway Patrol Enforcement Division, dated August 18, 2010 (b) Letter from Secure Transport Services to Nevada State Health Division, dated August 17, 2010 (c) Letter from Secure Transport Services to Arizona Radiation Regulatory Agency, dated August 18, 2010 (d) Letter from Secure Transport Services to Utah Department of Environmental Quality, Division of Radiation Control, dated August 17, 2010 (e) Letter from Secure Transport Services to Idaho State Police Commercial Vehicle Safety, dated August 17, 2010

Category: Transportation **Topic:** Annual Cask Inspection Certification

Reference: NAC-LWT SAR Section 8.1.3, 8.2, Table 8.2-1

Requirement: The completion of the annual maintenance and test program is required for each NAC-LWT cask while in service. The completion of the annual maintenance is documented on an annual inspection certification document. Each NAC-LWT cask must have a current annual certificate before it can be used. The annual maintenance program requirements are provided in Table 8.2-1 "Maintenance Program Schedule."

Finding: The required annual maintenance for the three casks (Cask #2, Cask #7 and Cask #8) used for the GA fuel shipments was completed within the calendar year in accordance with NAC Procedure 315-P-03. The required annual inspection items from SAR Table 8.2.1 were included in NAC Procedure 315-P-03. NAC-LWT Cask #2 annual maintenance operation was performed on October 30, 2009. NAC-LWT Cask #7 annual maintenance operation was performed on May 20, 2010. NAC-LWT Cask #8 annual maintenance operation was performed on May 2, 2010. The annual maintenance requirements were all signed off as completed in the respective annual maintenance data packages.

Documents Reviewed: (a) NAC Procedure 315-P-03 "NAC-LWT Annual Maintenance Procedure," Rev 21 (b) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #2, dated October 2009 (c) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #7, dated May 2010 (d) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #8, dated May 2010

Category: Transportation **Topic:** Basket Spacers & Inserts

Reference: NAC-LWT SAR Section 7.1

Requirement: Content configurations may require spacers, baskets, basket inserts, canisters, etc. to support and/or control the content geometry during transport.

Finding: The licensee utilized spacers and inserts to support the fuel elements during shipping. NAC Procedure 315-P-15, Section 3.1.5 stated "determine the height of the fuel element, cluster rod insert or failed fuel can (FFC) to be installed in each module cell position and insert spacers if required." The licensee developed a Fuel Element Description Table as Attachment 1 to RSD Form 434.28 that provided the fuel element length for all 263 fuel elements. Based on this chart and results from GA Procedure DDP-1.16, which cropped some fuel elements to suitable sizes to fit in the baskets, the licensee produced a table called Cask Spacers. The Cask Spacers table described which fuel elements required inserts during transport. The number, size and location of the axial fuel spacers were documented in RSD Form 434.31.

Documents Reviewed: (a) NAC Procedure 315-P-15 "NAC-LWT TRIGA Fuel Basket Loading Procedure," Rev 2 (b) Required Shipper Data (RSD) Form 434.28 "Fuel and Packaging Required Shipper's Data Form," Rev 4 with Attachment 1 "Fuel Element Description Table," Rev 0 (c) GA Procedure DDP-1.16 "GA TRIGA D&D Project: Fuel Element Cropping and Lanyard Installation," Rev A (d) Cask Spacer List, Rev 0 (e) Required Shipper Data (RSD) Form 434.28, "Fuel and Packaging Required Shipper's Data Form," Rev 4 (f) Required Shipper Data (RSD) Form 434.31 "Final Shipping Contents,"

Category: Transportation **Topic:** Center of Gravity for Package

Reference: CoC 9225 Condition 13/SAR Table 2.2.1-1

Requirement: Packages used to ship metallic fuel rods may be shipped in closed shipping containers. When the cask is shipped in a closed shipping container, the center of gravity of the combined cask, closed shipping container and trailer must not exceed 75 inches.

Finding: NAC International provided a calculation for the center of gravity which showed the 75 inch limit specified in the CoC Condition 13 was met. The calculated center of gravity was 72.9 inches, as shown in Table B1 of Calculation Package No. 14661-202. NAC used five TRIGA baskets configured in a NAC-LWT transport cask system with a design weight of 52,000 lbs. The calculations were performed based on the cask loaded on the trailer with a ISO container in place. The total weight of the cask, ISO container and trailer was 79,214 lbs. The height of the trailer bed was 40 inches.

Documents Reviewed: (a) NAC International Calculation Package No. 14661-202 "TRIGA Five Unit Basket Weight and Center of Gravity Calculation," dated August 18, 2010

Category: Transportation **Topic:** Closure Lid Metallic O-Ring

Reference: CoC 9225 Condition 10/SAR Sect 8.2, Table 8.2-1

Requirement: The closure lid has two O-ring seals, a Teflon seal and a metallic stainless steel seal.

Replace the closure lid metallic O-ring seal prior to each loaded transport.

Finding: NAC Procedure 315-P-02, Section 7.1.2.11 required replacement of the closure lid metallic O-ring seal prior to each loaded shipment. This requirement was also stated in Procedure DDP-1.15, as a note to Section 9.1.1.8. The metallic o-ring was replaced prior to shipment of the first NAC cask.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14 (b) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transport from GA to INL," Issue A

Category: Transportation **Topic:** Impact Limiter Lug Annual Inspection

Reference: NAC-LWT SAR Section 8.2.1.1

Requirement: During the annual NAC-LWT package maintenance, the impact limiter attachment lugs and the welds sealing the impact limiter shell to the lugs are visually examined with acceptance criteria in accordance with ANSI/AWS Code D1:2, Paragraph 8.8.1.

Finding: The required annual lug inspection for the three casks used for the GA fuel shipments was completed within the calendar year. Procedure 315-P-02, Section 6.14.3 required a visual inspection of the impact limiters lifting lugs. Section 6.14.4 required a visual check of all shell welds. Section 6.14.5 required a visual examination of the impact limiter attachment lugs and the welds sealing the impact limiter shell to the lugs. Inspections performed by NAC looked for cracking or other injurious defects using the acceptance criteria in ANSI/AWS Code D1.2, Paragraph 8.8.1. NAC-LWT Annual Maintenance Data Package for casks #2, #7, and #8 were reviewed during this inspection. The above sections were signed off as completed for each cask. Cask #2 had five discrepancies identified: three ¼" cracks in the weld connecting the gusset to the end skin and two gouges along a seam weld. Cask #8 had a 4" crack discovered on the inside of the limiter. All identified cracks were entered into the NAC corrective action system and were repaired with approved welding procedures.

Documents Reviewed: (a) NAC Procedure 315-P-03 "NAC-LWT Annual Maintenance Procedure," Rev 21 (b) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #2, dated October 2009 (c) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #7, dated May 2010 (d) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #8, dated May 2010

Category: Transportation **Topic:** Maximum Weight per Package

Reference: CoC 9225 Cond 5(b)(2)/SAR Drawing 315-40-079 (#3)

Requirement: The maximum quantity of material per package shall not exceed 4,000 pounds, including contents and fuel assembly basket or other internal support structures.

Finding: The maximum weight of the fuel, basket and internal support structures was less than 4,000 lbs. Each NAC-LWT cask contained five baskets of fuel. The bottom and top basket were physically different from the identical three intermediate baskets (middle). The maximum weight of the five combined baskets was 3,095 lbs. This included the weight of the fuel elements and other supporting internal structures. The bottom basket's maximum weight was 594 lbs, the three intermediate baskets maximum weights were 560 lbs each, and the top basket maximum weight was 821 lbs.

Documents Reviewed: (a) Required Shipper Data (RSD) Form 434.28, "Fuel and Packaging Required Shipper's Data Form," Rev 4

Category: Transportation **Topic:** Neutron Shield Tank Solution - 1% Boron

Reference: CoC 9225, Condition 5(a)(2), 11/SAR Sect 1.1

Requirement: When shipping TRIGA fuel elements, the neutron shield tank must be filled with an ethylene glycol/water solution that is at least 1% boron by weight.

Finding: The neutron shield boron concentration had been verified for all three casks used by GA for the spent fuel shipment during the annual maintenance checks for each cask. Procedure 315-P-03 established the annual maintenance requirements for the NAC-LWT casks. Sections 6.5 and 6.7 of Procedure 315-P-03 required the shield tank level and boron concentration to be checked and corrected, if necessary. If the level was determined to be low, Section 6.5.5 required the tank to be replenished with the neutron shield tank fluid mixture specified in Section 5.2, which provided the number of gallons of ethylene glycol and de-mineralized water to mix with 70-75 lbs of potassium tetraborate to obtain the desired concentrations for the neutron shield tank solution. As part of the annual maintenance checks of the casks, Section 6.7 required removing a sample of 2 cups from each cask to be sent for chemical analysis for assurance of 1% boron concentration. If the boron concentration was low, Section 6.7.5.1 provided the necessary instructions for adding additional potassium tetraborate to the shield tank solution to obtain the required boron concentration. After the solution was added to the neutron shield tank, Section 6.7.5.1.c required the tank concentration to be retested for boron concentration level. The NAC-LWT Annual Maintenance Data Package for casks #2, #7, and #8 documented that all three casks had been checked for shield water level and boron concentration. The boron concentration lab sample results were 1.16%, 1.09%, 1.05% wt boron for casks #2, #7, and #8, respectively.

Documents Reviewed: (a) NAC Procedure 315-P-03 "NAC-LWT Annual Maintenance Procedure," Rev 21 (b) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #2, dated October 2009 (c) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #7, dated May 2010 (d) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #8, dated May 2010 (e) Galbraith Laboratory Report, PO# 715574, dated October 28, 2009 (f) Galbraith Laboratory Report, PO#784526, dated May 19, 2010 (g) Galbraith Laboratory Report, PO# 784525, dated May 19, 2010

Category: Transportation **Topic:** Personnel Barrier Used for Shipping

Reference: CoC 9225 Condition 12

Requirement: For shipments of TRIGA fuel elements, an ISO container or a personnel barrier must be used.

Finding: An ISO intermodal container was used for the first shipment of GA spent fuel as a barrier around the NAC-LWT cask. Procedure 315-P-02, Sections 7.2.2.19 through 7.2.2.25 described the process for installing and locking the ISO container around the cask.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14 (b) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transport from GA to INL," Issue A

Category: Transportation **Topic:** Placarding

Reference: 49CFR172.504(a) and (e)

Requirement: Each transport vehicle carrying Radioactive Yellow III material must be placarded as Radioactive on each side and each end.

Finding: Radioactive placards were placed on all four sides of the ISO container.

Documents Reviewed: None

Category: Transportation **Topic:** Route Approval

Reference: 10 CFR 73.26 (i)

Requirement: For shipments by road, a detail route plan shall be prepared.

Finding: On February 19, 2010 Secured Transportation Services, on behalf of GA, submitted to the NRC a route approval request for the shipment of the GA spent fuel to the DOE facility near Idaho Falls, Idaho. On April 29, 2010 the NRC responded to Secured Transportation Services approving the requested route.

Documents Reviewed: (a) Letter from Secured Transportation Services to the NRC requesting approval of the route for the GA spent fuel shipment to Idaho Falls, Idaho (safeguards document not publically available), dated February 19, 2010 (b) Letter from NRC to Secured Transportation Services entitled "Approval of Highway Route to Transport Spent Nuclear Fuel from General Atomics, San Diego, California to the Idaho National Laboratories, located near Idaho Falls, Idaho, Docket #70-07011 (NRC Route 207)" dated April 29, 2010 (ML101120328).

Category: Transportation **Topic:** Torque Requirements for Lid Bolts

Reference: CoC 9225 Condition 8/SAR Table 7.3-1, Table 8.2-1

Requirement: Visually inspect the cask lid bolts for damage and replace as required. The closure lid bolts should be torqued to 260 +/- 20 in-lbs. All bolts shall be lightly lubricated using Nuclear Grade Pure Nickel NEVER-SEEZ or equivalent.

Finding: The cask lid bolts were properly torqued prior to shipment of the first NAC-LWT cask. NAC Procedure 315-P-02, Sections 7.1.4.7 and 7.2.1.2 required the closure lid bolts to be lubricated with NEVER-SEEZ, or equivalent, and torqued to 260 ± 20 in-lbs. Procedure DDP-1.15, Section 9.1.3.1.3 listed the same torque requirements. The lid bolts were lubricated and torqued to 260 in-lbs with calibrated torque wrench Serial # 1197500099. Calibration due date was April 13, 2011.

Documents Reviewed: (a) GA Procedure DDP-1.15 "Packaging of TRIGS SNF for Transport from GA to INL," Issue A (b) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14

Category: Transportation **Topic:** Torque Requirements for Port Covers

Reference: CoC 9225 Condition 8/SAR Section 7.1, Table 7.3-1

Requirement: The alternate port cover bolts are torqued to 100 +/- 10 inch-pounds. All bolts shall be lightly lubricated using Nuclear Grade Pure Nickel NEVER-SEEZ or equivalent.

Finding: The alternate port cover bolts were properly torqued prior to shipment of the first NAC-LWT cask. NAC Procedure 315-P-02, Sections 7.2.1.6 and 7.2.1.23 required the alternate port cover bolts to be lubricated with NEVER-SEEZ and to be torqued to 100 ± 10 in-lbs. Procedure DDP-1.15, Section 9.1.7.2 also specified the 100 ± 10 in-lb torque requirement. The port covers were properly torqued using a calibrated torque wrench, NAC ID# HT-205-NAC, Serial No. 1097302502.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14 (b) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transport from GA to INL," Issue A (c) Instrument Calibration Solutions Certificate of Calibration #12802 for SNAP ON torque wrench Model QC2R200 (serial # 1097302502) dated April 13, 2010

Category: Transportation **Topic:** Vacuum Drying Criteria/Helium Backfill

Reference: CoC 9225 Condition 9 & 10/NAC-LWT SAR Sect 7.1

Requirement: The cask cavity is vacuum dried by attaching a vacuum pump to the vent and/or drain port and evacuating the cavity to a pressure of less than 10 torr (13 mbar) and continuing to vacuum pump for an additional 15 minutes. If the cavity pressure rise is less than 5 torr (6.7 mbar) during a 10 minute isolation and hold period, there is no free water in the cavity and the cask cavity is verified as dry. If the pressure rise is > 5 torr (6.7 mbar), the vacuum drying will continue until the dryness verification criteria is met. Backfill of the cavity by helium is required in order to conduct the lid helium leak test required by CoC 9225 Condition 9.

Finding: The first cask was successfully dried to less than the 10 torr requirement for 15 minutes and backfilled with helium. The cask was vacuum dried to 1.8 torr, then isolated. At the end of 15 minutes, the cask pressure was 5.1 torr. The cask was then filled with helium to a pressure of 0 psig. NAC Procedure 315-P-02 had incorporated the required instructions for vacuum drying and helium backfill into Sections 7.2.1.6 thru 7.2.1.18 consistent with SAR Section 7.1 and SAR Sections 7.1.6.25 thru 7.1.6.29. Attachment 1 "Procedure Traveler" to Procedure 315-P-02 provided the documentation of the vacuum pressures and required QC sign-off upon completion of the test. The GA Procedure DDP-1.15, Sections 9.1.5 and 9.1.6 also included the drying and helium backfilling requirements and a table recording the results. The gauge used for the dryness test was serial # NAC-PV20-2 with a calibration due date of April 28, 2011. The helium used for backfilling the cask was 99.995% minimum purity.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14 (b) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transport from GA to INL," Issue A (c) Praxair Certificate of Conformance for Helium 4.5 Grade for Cylinder #5620858 dated April 13, 2010

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Category: <u>Transportation</u> **Topic:** <u>Valve Port Cover Pre-Shipment Inspection</u>

Reference: NAC-LWT SAR Table 8.2-1

Requirement: Perform the following inspections of the valve port covers prior to each loaded transport: a) visual inspection, and b) air pressure drop test of the alternate port covers at 15 +1/-0 psig. Replace seals as necessary. If the valve port cover seals are replaced, a helium leak test in accordance with SAR Section 8.1.3 is required.

Finding: The valve port covers were inspected on the first cask and successfully air pressure drop tested. NAC Procedure 315-P-02 incorporated the requirements from Table 8.2-1 of the SAR for the valve port cover pre-shipment inspection. Section 7.2.1.25 connected a gas supply to the test fixture and performed a gas pressure drop leakage test by slowly applying 15 psi +1,-0 psi to the port cover seal annulus and then isolated the gas supply. Section 7.2.1.26 required observation of the test pressure gage for a minimum of 10 minutes to ensure that there was no pressure drop observed. GA Procedure DDP-1.15 also included the air drop test criteria in Sections 9.1.7.3 thru 9.1.7.7. The test was considered successful if there was no pressure drop observed (i.e. less than the minimum gauge readability of 0.25 psi) during the 10 minute hold period. Section 7.2.1.27 required a helium leak test if the Viton O-ring was replaced in Section 7.2.1.22.

The Ashcroft test pressure gauge used to measure the air pressure drop had a range 0-40 psi with a minimum readability of 0.1 psi. Calibration due date was February 19, 2011. For the first cask loaded with GA fuel, the valve port covers were visually inspected and an air drop test was performed. Both the vent port cover and drain port cover had a pressure drop less than the 0.25 psi reading on the gauge. The vent port cover tested 0.16 psi air pressure drop. The drain port cover tested 0 psi.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14 (b) GA Procedure DDP-1.15 "Packaging of TRIGA SNF for Transport from GA to INL," Issue A (c) Instrument Calibration Solutions Certificate of Calibration #12047 for pressure gauge Serial # A4A-135413 [NAC Serial # 1354413]

Category: <u>Transportation</u> **Topic:** <u>Water Jacket Pre-Shipment Inspection</u>

Reference: NAC-LWT SAR Table 8.2-1

Requirement: Visually inspect the water jacket and expansion tank fill, drain and inspection port plugs for leakage prior to each loaded transport.

Finding: The water jackets and expansion tank were inspected for the first cask prior to shipment and no leaks were found. NAC Procedure 315-P-02, Section 7.1.2.2 required a visual inspection of the neutron shield tank fill, drain and level inspection plugs for signs of neutron shield fluid leakage. If leakage was detected or suspected, the shield tank fluid level was to be verified and corrected, as required. There were no water jacket leaks that occurred during the time the NAC-LWT cask was positioned for receipt of the baskets, during the movement of the NAC-LWT cask into the ISO container, or during the time vacuum drying was occurring. All this took several days with no leaks detected.

Documents Reviewed: (a) NAC Procedure 315-P-02 "NAC-LWT Cask Generic Operating Procedure," Rev 14

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Category: Transportation	**Topic:** Water Jacket Relief Valve Pressure Test

Reference: NAC-LWT SAR Section 8.1.4.1, Table 8.2-1

Requirement: Overpressurization protection is afforded the neutron shield tank (water jacket) in the form of a relief valve that is designed to open at 165 psig +/- 10%, and reseat. The relief valve is removed from the cask and hydraulically pressure tested using a calibrated system to verify relief valve opening and closing pressures. Failure to operate within tolerance is cause for rejection. Rejected valves are rebuilt or replaced and retested prior to use. Annually the water jacket relief valve (neutron shield tank) is replaced with a new pre-set valve or the opening and resetting pressure verified.

Finding: The water jacket relief valves for Casks # 2, #7 and #8 were successfully tested during their latest annual maintenance operation. The over pressurization protection relief valve was required to open at 165 psig +/- 10% (16.5 psig) per SAR Section 8.1.4.1. The annual LWT maintenance Procedure 315-P-03, Section 6.8 replaced the relief valves and conservatively tested the actuation at 165 psig +/- 5 psig. The replaced relief valve serial numbers and signature of completion for the pressure tests for Casks #2, #7, and #8 were included in the NAC-LWT Annual Maintenance Data Package for each cask.

Documents Reviewed: (a) NAC Procedure 315-P-03 "NAC-LWT Annual Maintenance Procedure", Rev. 21 (b) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #2, dated October 2009 (c) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #7, dated May 2010 (d) NAC-LWT Annual Maintenance Data Package for NAC-LWT Cask #8, dated May 2010