NWMI
NORTHWEST MEDICAL ISOTOPES

U.S. Nuclear Regulatory Commission
Pre Construction Permit Application Meeting

Public Presentation

September 30, 2014
9:00am – 4:00pm

Agenda

September 30, 2014

Public Session

9:00am	*Opening Remarks*	*NRC*
9:05am	*NWMI Introductions and Status (10 min)*	*NWMI*
	Radioisotope Production Facility Licensing Approach (10 minutes)	
	Application of 10 CFR 50 and 10 CFR 70; 20 minutes	
	Integrated Safety Analysis/Criticality/Shielding Approach and Methodology (90 minutes)	
	Transportation Strategy (20 minutes)	
	Public Question and Answer Period (30 minutes)	
	NRC	

Non-Public Session

1:00pm	*Radioisotope Production Facility Construction and Operations*	*NWMI*
2:00pm	*Meeting Adjourn*	

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U.S. Nuclear Regulatory Commission
Pre Construction Permit Application Meeting

Public Portion

September 30, 2014
9:00am – 12:00pm

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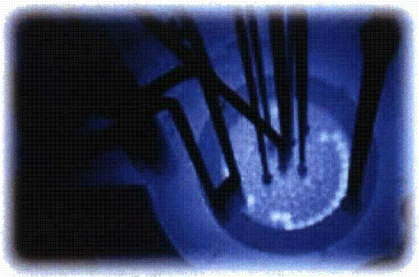
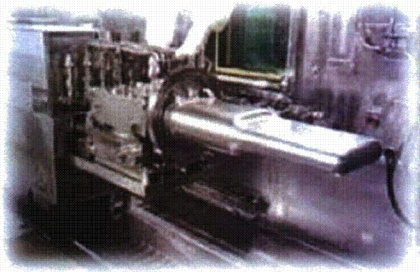
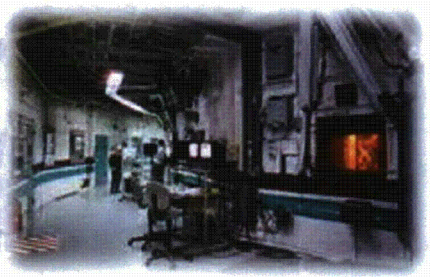
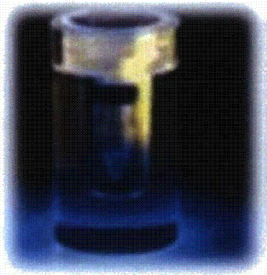
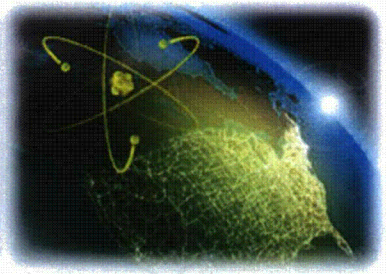
NWMI OVERVIEW AND STATUS

September 30, 2014

Organization

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Nick Fowler, President
Carolyn Haass, Vice President
Steve Reese, Irradiations Logistics Manager
Marcus Voth, Technical Adviser

Irradiation Services – University Reactors

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Radioisotope Production Facility Partners

Engineering Design

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Criticality, Shielding, and Safety Analysis

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Proven • Effective • Solutions

Transportation

STS
Secured Transportation Services

Licensing and Environmental Permitting

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Technology Demonstration Partners

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National Laboratory

OSU
Oregon State
UNIVERSITY

MCI INC
Materials and Chemistry Laboratory, Inc.
"Linking Technology to Solutions"

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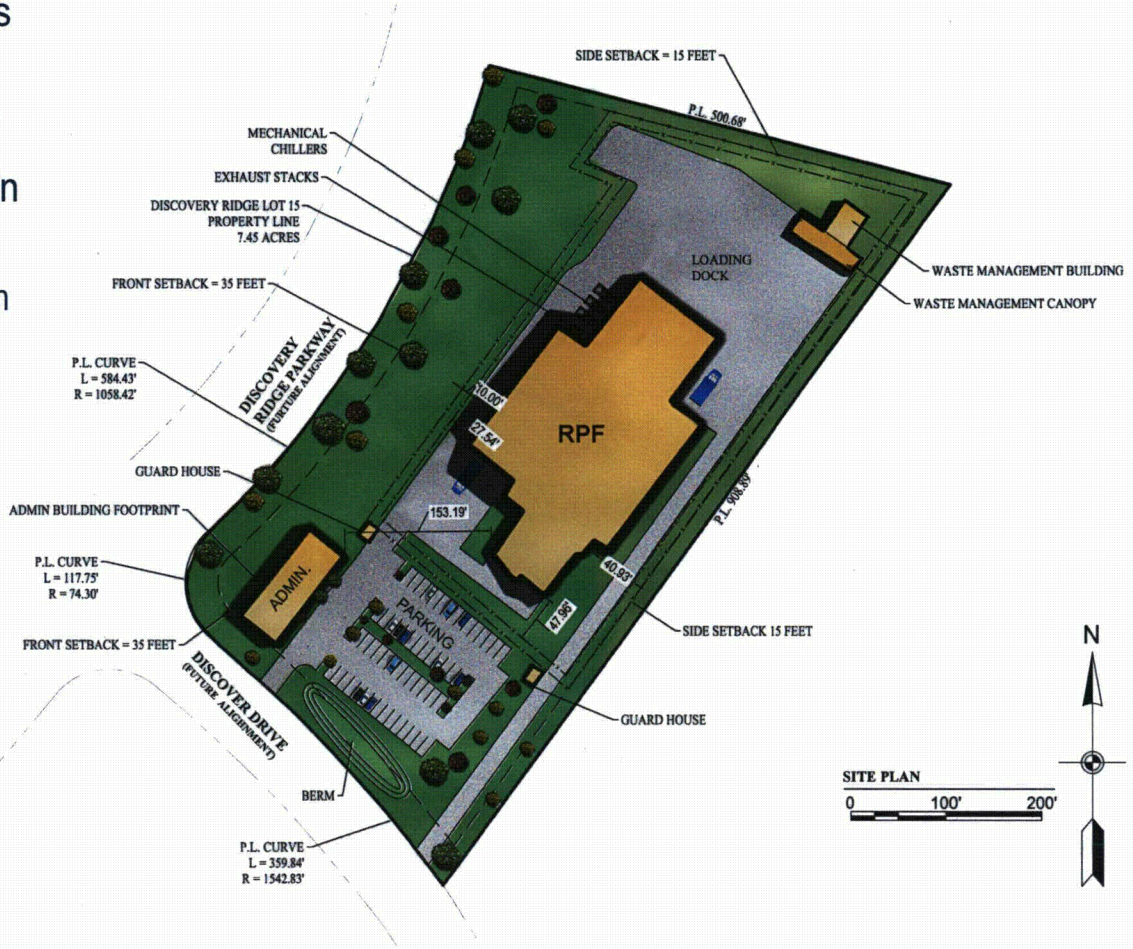
NWMI Project Status

- Construction Permit Application
  - Received Exemption to Submit Construction Permit Application in 2 Parts (Oct 2013)
  - Part 1 complete; submission is imminent
  - Part 2 Submission will be submitted within 180 days of Part 1 submission
    - Part 2 of Construction Permit Application and Preliminary Design in progress
- Proof of concept tests have and are being performed in cooperation with MU, OSU and ORNL
- Prototypic target production initiated
- Siting Decision Taken; Option formalized
- Strategic Partnerships and Major Subcontractor development complete
- Facility financing in progress



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RPF LICENSING APPROACH

September 30, 2014

Licensing Overview

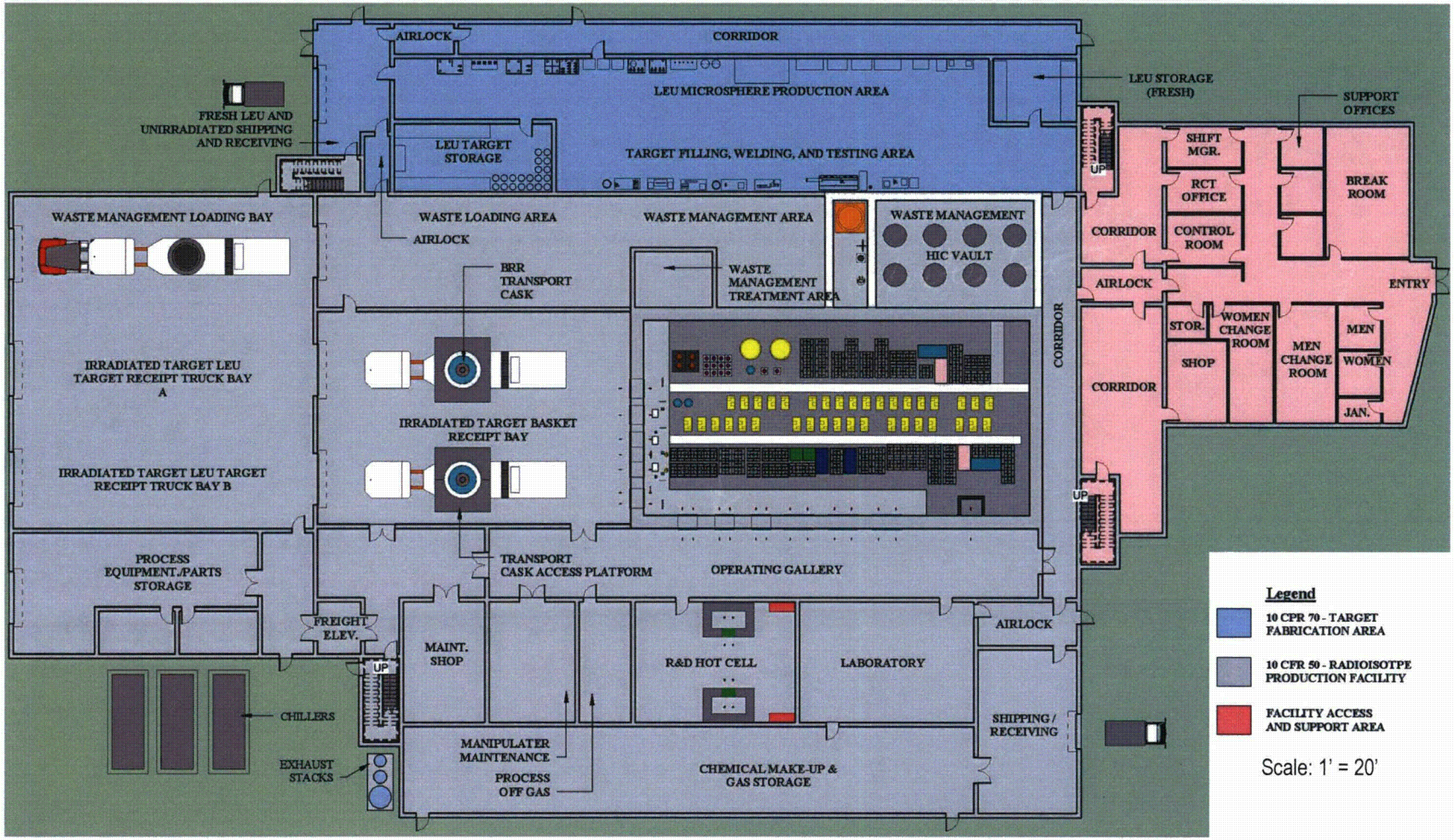
- 10 CFR 50 format, content, and review guidance for non-power reactors given in:
  - NUREG-1537, Part 1, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content*
  - NUREG-1537, Part 2, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria*
- *Final Interim Staff Guidance Augmenting NUREG-1537 (Oct 2012), "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Parts 1 and 2*
  - ISG provides explicit criteria for application of review guidance in NUREG-1537 for non-power reactors and RPFs
- 10 CFR 70 applies directly to the LEU target manufacturing portion of the RPF
  - NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility – Final Report*, provides review guidance for this portion of the license
- ISG NUREG-1537 allows many review elements for NUREG-1537 to be satisfied by NUREG-1520
- For example:
  - NWMI will follow an ISA process consistent with licenses submitted under 10 CFR 70
    - ISA will use approach, methodologies, and terminology provided in NUREG-1520 and regulatory requirements found in 10 CFR 70 to meet acceptance criteria specified in ISG NUREG-1537
    - On completion of development of ISA baseline documents, a set of items relied on for safety (IROFS) will be incorporated into technical specifications for 10 CFR 50 facility operations and into PSAR
    - 10 CFR 70 facility operations will use remaining IROFS and will be incorporated into an ISA report/summary
    - Formats for technical specifications and IROFS will be contained in one template of the IROFS boundary definition packages that cover requirements of both 10 CFR 50 and 10 CFR 70

Licensing Approach

- NWMI will combine several license activities and submit one application that covers all applicable regulations for construction and operation of a Commercial Radioisotope Production Facility (RPF) under 10 CFR 50
  - Process ^{99}Mo and recycle LEU under 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*
  - Target fabrication (ability to receive, possess, use, and transfer of SNM) under 10 CFR 70, *Domestic Licensing of SNM*
  - Ability to handle by-product material under 10 CFR 40, *Rules of General Applicability to Domestic Licensing of Byproduct Material*
- NWMI's understanding
  - NRC will approve and issue one license under 10 CFR 50; Activities under 10 CFR 70 and 10 CFR 30 will be part of 10 CFR 50 license (10 CFR 50.31, *Combining Applications*)
  - NRC will complete a single review process (10 CFR 50.32, *Elimination of Repetition*)
  - Only interact with one group within NRC (e.g., administrative, license reviews and approvals, inspections) (10 CFR 50.32, *Elimination of Repetition*)
  - Fees will only be assessed under 10 CFR 50
- Other Integrated License Activities to support NWMI's RPF
  - University Reactor will amend it's current operating license's to support production of ^{99}Mo
  - Required transportation casks will be amended, if necessary

Facility Licensing Layout

- 10 CFR 50 Activities
  - Irradiated Target receipt
  - Irradiated target disassembly
  - Target dissolution
  - ^{99}Mo separations, purification and packaging
  - LEU reclamation and purification
  - Waste management
  - Associated laboratory and support
- 10 CFR 70 Activities
  - Receipt of LEU (from DOE)
  - Production of LEU microspheres
  - Target fabrication and testing
  - Shipping/loading of fabricated targets
  - Associated laboratory and support areas
- 10 CFR 30 Activities
  - Handling of byproduct material



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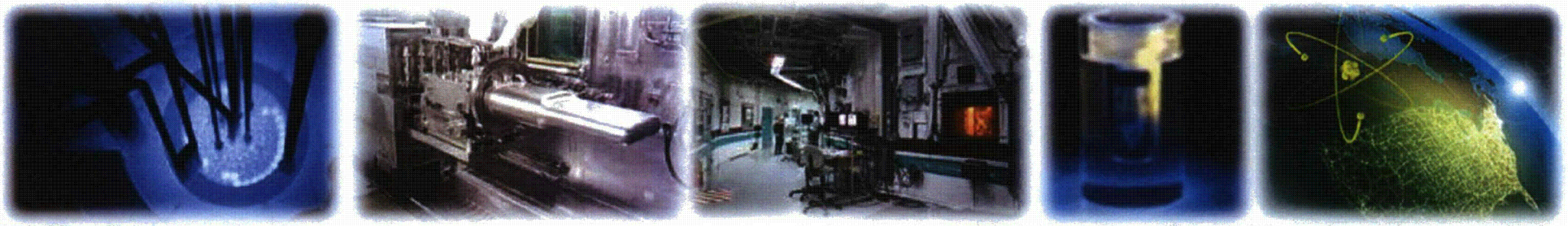
INTEGRATED SAFETY ANALYSIS

September 30, 2014



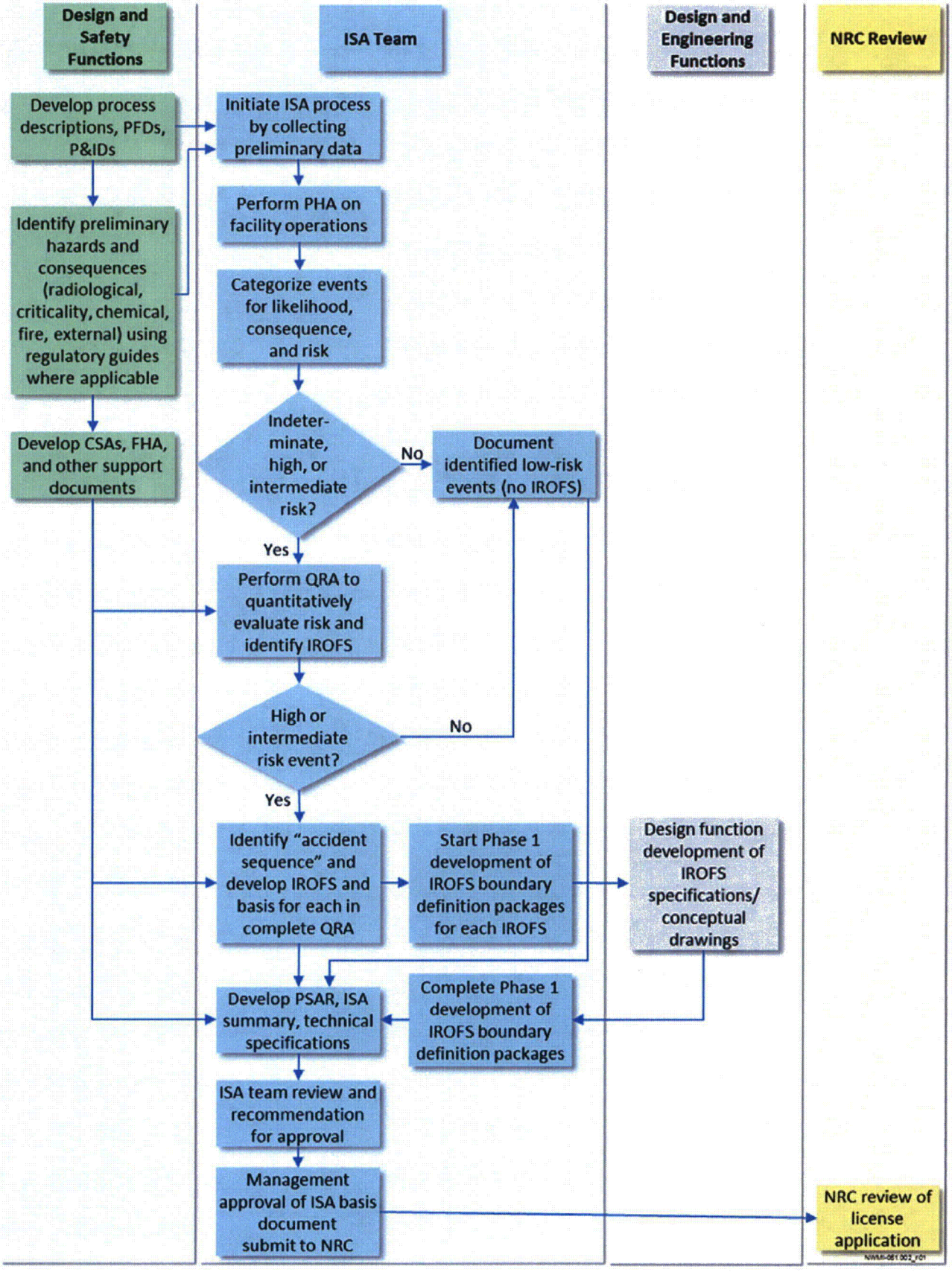
Licensing Regulations

- 10 CFR 50
  - Requires a PSAR and Technical Specifications
  - NUREG-1537
    - Provides NRC format, content, and review guidance for 10 CFR 50 non-power reactor licenses
    - Interim Staff Guidance (ISG) – Modifies NUREG-1537 to apply to Production Facilities
- 10 CFR 70
  - Requires a License Application (describing program and commitments) and an ISA Summary (describing the Safety Basis)
  - NUREG-1520
    - Provides NRC format, content, and review guidance to 10 CFR 70

ISA Process

- Conduct PHA of a system using a combination of written process descriptions, PFDs, P&IDs, and supporting calculations to identify events that could lead to adverse consequences
- Adverse consequences are evaluated qualitatively to identify their likelihood and severity using guidance on event frequencies and consequence categories consistent with regulatory guidelines
- Each event with an adverse consequence that involves licensed material or its byproducts is evaluated for risk using a risk matrix that enables user(s) to identify unacceptable intermediate- and high-consequence risks
  - For these risk events, IROFS will be developed to prevent or mitigate consequences of events and event tree analysis will be used to demonstrate that risks can be reduced to acceptable frequencies through preventative or mitigative IROFS
- Fault trees and failure mode and effects analysis can be used to:
  - Provide quantitative failure analysis data (failure frequencies) for use in event tree analysis of IROFS
  - Quantitatively analyze an event from its basic initiators to demonstrate that quantitative failure frequencies are already highly unlikely under normal standard industrial conditions, thus not needing application of IROFS
- Once IROFS are developed, management measures will be identified to ensure that IROFS failure frequency used in analysis is preserved and are able to perform their intended function when needed
- IROFS will be converted to Technical Specifications in support of 10 CFR 50 activities
- Will utilize computer codes that have been accepted by the NRC, or will provide V&V

ISA Process Flow Diagram



Preliminary HAZOPs

- Preliminary HAZOPS has been completed using conceptual design information
- Identified RPF Hazards
  - Criticality
  - Release of radioactive off-gas or radioactive products
  - Radioactive waste
  - Explosion – resulting from production of hydrogen by radiolytic decomposition of irradiated fissile solution
  - Tank and equipment failure leading to a release of radiological or chemical materials
  - Release during receiving of hazardous chemicals outside RPF that impact licensed material areas

Consequence Categories Per 10 CFR 10.61

- ISG NUREG-1537 allows performance requirements of 10 CFR 70 to be used as a basis for identifying hazards criteria for license application; hazards will be divided into three consequence categories
  - Radiological dose limits
  - Toxicological limits
  - Chemical hazard limits
- Primary NWMI RPF hazards include:
  - Radiological dose hazards from irradiated target material (chronic and acute dose) and accidental nuclear criticality (acute dose)
  - Toxicity from uranium and other fission fragments
  - Chemical hazards from the nitric acid used in some of processing

Category Description	Consequence Category	Workers	Offsite Public	Environment
High consequence	3	Radiological dose[a] > 1 Sv (100 rem) Airborne, radiologically contaminated nitric acid > 170 ppm nitric acid (AEGL-3, 10-min exposure limit) Unshielded nuclear criticality	Radiological dose > 0.25 Sv (25 rem) \ toxic intake > 30 mg soluble U Airborne, contaminated nitric acid > 24 ppm nitric acid (AEGL-2, 60-min exposure limit)	
Intermediate consequence	2	Radiological dose between 0.25 Sv (25 rem) and 1 Sv (100 rem) Airborne, radiologically contaminated nitric acid > 43 ppm nitric acid (AEGL-2, 10-min exposure limit)	Radiological dose between 0.05 Sv (5 rem) and 0.25 Sv (25 rem) Airborne, contaminated nitric acid > 0.16 ppm nitric acid (AEGL-1, 60-min exposure limit)	24-hr radioactive release > 5,000 × Table 2 of 10 CFR 20,[b] Appendix B
Low consequence	1	Accidents with lower radiological, chemical, and/or toxicological exposures than those above from licensed material and byproducts of licensed material	Accidents with lower radiological, chemical, and/or toxicological exposures than those above from licensed material and byproducts of licensed material	Radiological releases producing lower effects than those listed above from licensed material

Likelihood Categories

- In evaluating the risk associated with an adverse event, likelihood categories are qualitatively determined as part of PHA process
- "Unlikely" accident is a sequence of events that occurs with a frequency of 10^{-3} and 10^{-5} events per year
  - Frequency range is consistent with the application of IROFS failure frequencies
  - Lowest level of IROFS [e.g., administrative controls (AC) and augmented administrative controls (AAC) with sufficient management measures to enhance robustness] can be credited with up to 10^{-3} failures per year
  - Robust passive engineered controls can be credited with 10^{-4} failures per year
  - Active engineering controls can be credited somewhere in between, depending on complexity and robustness
  - Not all IROFS will have a credited failure frequency of less than 10^{-3} failures per year
    - Some human error failures will require multiple levels of protection and more than two IROFS to satisfy performance criteria of an accident sequence
- "Highly unlikely" accident sequences are those sequences of events that occur with a frequency of less than 10^{-5} events per year
- Following definitions will be used to define an event as not credible ➔ Thus, not being evaluated further
  - An external event has a frequency of occurrence that can conservatively be estimated as less than 10^{-6} event/ year
  - A process deviation consists of a sequence of many unlikely events or errors for which there is no reason or motive
  - A convincing argument exists that, given physical laws, process deviations are not possible, or are extremely unlikely

	Likelihood Category	Event Frequency Limit
Not Unlikely	3	More than 10^{-3} events/year
Unlikely	2	Between 10^{-3} and 10^{-5} events/year
Highly Unlikely	1	Less than 10^{-5} per events per year

Quantitative Risk Assessment

- As part of PHA process, NWMI will identify a likelihood category for each event and consequence category for the adverse consequence
- Using these categories, a risk index will be calculated and event sequence identified and be determined to either be an acceptable or unacceptable risk
- Events with adverse consequences/events that need further quantitative development are identified, a QRA is performed which involves:
  - Compiling all initiators that are identified in PHA process to create a bounding accident sequence description
  - Identifying consequence of accidents through reference to the consequence analysis
  - Evaluating likelihood of the accident using event tree analysis, fault tree analysis, or FMEA analysis techniques
  - Developing IROFS to prevent or mitigate hazards and initiating development of IROFS boundary definition packages
  - Demonstrating IROFS prevent or mitigate the accident using event tree analysis
- During QRA process, accident sequences will be defined as needed to identify IROFS; these can be initiated by human error, systems failure, IROFS failure, or an external event

Severity of Consequences	Likelihood of Occurrence		
	Highly Unlikely (Likelihood Category 1)	Unlikely (Likelihood Category 2)	Not Unlikely (Likelihood Category 3)
High Consequence (Consequence Category 3)	Risk Index = 3 Acceptable Risk	Risk Index = 6 Unacceptable Risk	Risk Index = 9 Unacceptable Risk
Intermediate Consequence (Consequence Category 2)	Risk Index = 2 Acceptable Risk	Risk Index = 4 Acceptable Risk	Risk Index = 6 Unacceptable Risk
Low Consequence (Consequence Category 1)	Risk Index = 1 Acceptable Risk	Risk Index = 2 Acceptable Risk	Risk Index = 3 Acceptable Risk

Maximum Hypothetical Accident

- Maximum Hypothetical Accident (MHA) is used to bound consequence values for all credible potential accidents
  - MHA is a "Non-Credible/Unmitigated Accident"
  - MHA will provide bounding consequence values for all credible potential accidents
  - MHA and ISA will serve to satisfy Accident Analysis Requirements of 10 CFR 50.34(a)
- Assumed:
  - Certain facility, process, and procedural quantities will not be fully developed
  - Conservatism will be used to ensure a broad safety envelope to be defined

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CRITICALITY APPROACH

September 30, 2014

ANSI/ANS 8 Standards

- NWMI's RPF presents a number of differing challenges to nuclear criticality safety (NCS):
  - Enriched uranium
  - Uranium metal storage and handling
  - Numerous uranium solution processes
  - Uranium-bearing reactor target fabrication, handling, and staging
  - Waste handling and staging for disposal
- NWMI has considered and accounted for criticality safety in function of the RPF design
  - Has developed a criticality safety strategy based on NCS "first principles," recognized industry standards, and guidance handbooks at the early conceptual design stages
    - ANSI/ANS-8 series
    - TID-7016 / TID-7028; LA-12808 / LA-10860-MS (etc.)
  - Uses "first principles" and guides as bases for conceptual equipment design and process area layouts including:
    - Geometry constraints (e.g. pencil tank diameters)
    - Tank array spacing (conservative)
    - Consideration of transition from "safe-geometry" process equipment to less-restricted waste staging and processing equipment

Criticality Safety Program Overview

- Administrative Components (ANSI/ANS-8.1; ANSI/ANS-8.19) [*in progress*]
  - Program/policy procedure (defines responsibilities and program structure)
  - Analysis procedure (CSE)
  - Qualification requirements for NCS staff
- Interfaces will be developed with other safety disciplines, as appropriate
- Management measures will be developed and implemented to support NCS program (10 CFR 70.62) (e.g., configuration management/change control, audits/assessments, training, procedures)
- Analysis (ANSI/ANS-8.1; ANSI/ANS-8.19)
  - Establish analysis code(s) and appropriate code validation against accepted critical experiment benchmarks (ANSI/ANS-8.24; NUREG/CR-6698) [*in progress*]
    - Determine appropriate project Upper Subcritical Limit(s) for the facility-specific "areas of applicability" (i.e. the facility-specific process materials, structures, and parameters)
    - Incorporate code bias and bias uncertainty, and establish an appropriate administrative margin
  - Establish agreement for use of verified and QA'd computing platform [*in progress*]
  - Criticality Safety Evaluations [*in progress*]
    - Every fissile material handling process in the facility will be considered in CSEs
    - Graded approach based on complexity of unit process and type(s)/quantities of fissile material involved

NWMI Criticality Safety Program Overview (con't)

- Engineered Features and Administrative Controls [*TBD in CSEs*]
  - CSEs will identify physical features and process controls to be implemented in the facility design and (later) procedure development to prevent criticality
  - Control scheme, in order of preference (NUREG-1520 and ANSI/ANS-8.1)
    - Physical/engineered features (e.g. tank geometry, spacing)
    - Active engineered features (e.g. interlocks, fail-safe valves)
    - Administrative controls – Augmented and/or simple (e.g. mass limit on manual handling, # of containers on a cart)
- Integrated Safety Analysis
  - NCS will be a fundamental aspect of ISA
  - NCS analysts will participate in/contribute to the ISA development
  - CSEs will incorporate upset conditions identified in the ISA development (e.g. HAZOPs) and will identify NCS controls for consideration as IROFS in the ISA
- Regulatory and Standards considerations:
  - 10 CFR 70: -.24, -.61, -.62, -.64, -.65, -.72, and Appendix A
  - NUREG 1520
  - Regulatory Guide 3.71 (endorsement of ANSI/ANS-8 series)
  - ANSI/ANS-8 series of industry standards (as applicable to facility design and unit processes)

Criticality Safety Evaluation Process

- Will establish criticality safety limits & controls for fissionable material systems/operations
- Will document the affected and controlled NCS parameters for each process in RPF (e.g. mass, geometry, moderation, etc.)
- Will be developed with input from management, operations, project/process engineering, and other applicable disciplines
- Will be iterative and essentially "living documents" throughout design/construction/operations
- Analysis may drive changes to equipment and process designs
- CSEs will be revisited and revised as necessary as overall design matures
- Each operation involving or with potential to affect fissile material will be evaluated and determined to remain subcritical for all normal and credible abnormal conditions (ANSI/ANS-8.1)
  - Evaluations will provide documented compliance with the Double Contingency Principle and as necessary 10CFR70.61 requirements
  - Double Contingency Principle of ANSI/ANS-8.1 states:

    "Process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible"
- NWMI will conduct an independent (ANSI/ANS-8.19)

MCNP Validation (ANSI/ANS 8.24 Requirement)

- NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology
  - Define operation/process to identify range of parameters to be validated
  - Select critical experiment data
    - U metal
    - U compounds
    - U solutions
  - Model and run experiments
  - Analyze data
    - Determine bias and bias uncertainty
    - Identify trends in data
    - Test for normal or other distribution and select statistical method for data treatment (NUREG/CR-4604, Statistical Methods for Nuclear Material Management)
    - Identify and support sub-critical margin
    - Calculate Upper Sub-critical Limit (USL)
  - Define Area of Applicability (AOA) of the validation

Example – Pencil Tanks

- Used "first principles" and guides as bases for conceptual equipment design and process area layouts including geometry constraints (e.g. pencil tank diameters), provided conservative tank array spacing
- Passive design considerations
  - Use favorable geometry diameter tanks
  - Tank material is compatible with SNM contained
  - Spacing conservatively set
  - Seismic qualified skid mounting
- Active engineered considerations
  - None
- Administrative actions considered
  - None

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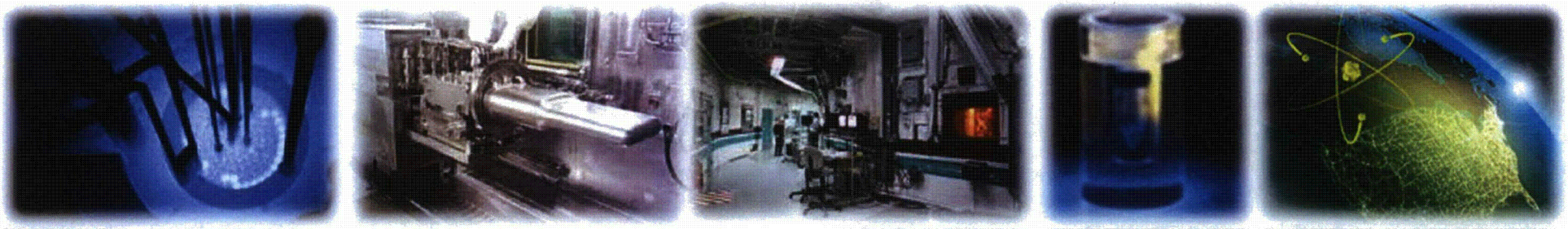
SHIELDING APPROACH

September 30, 2014



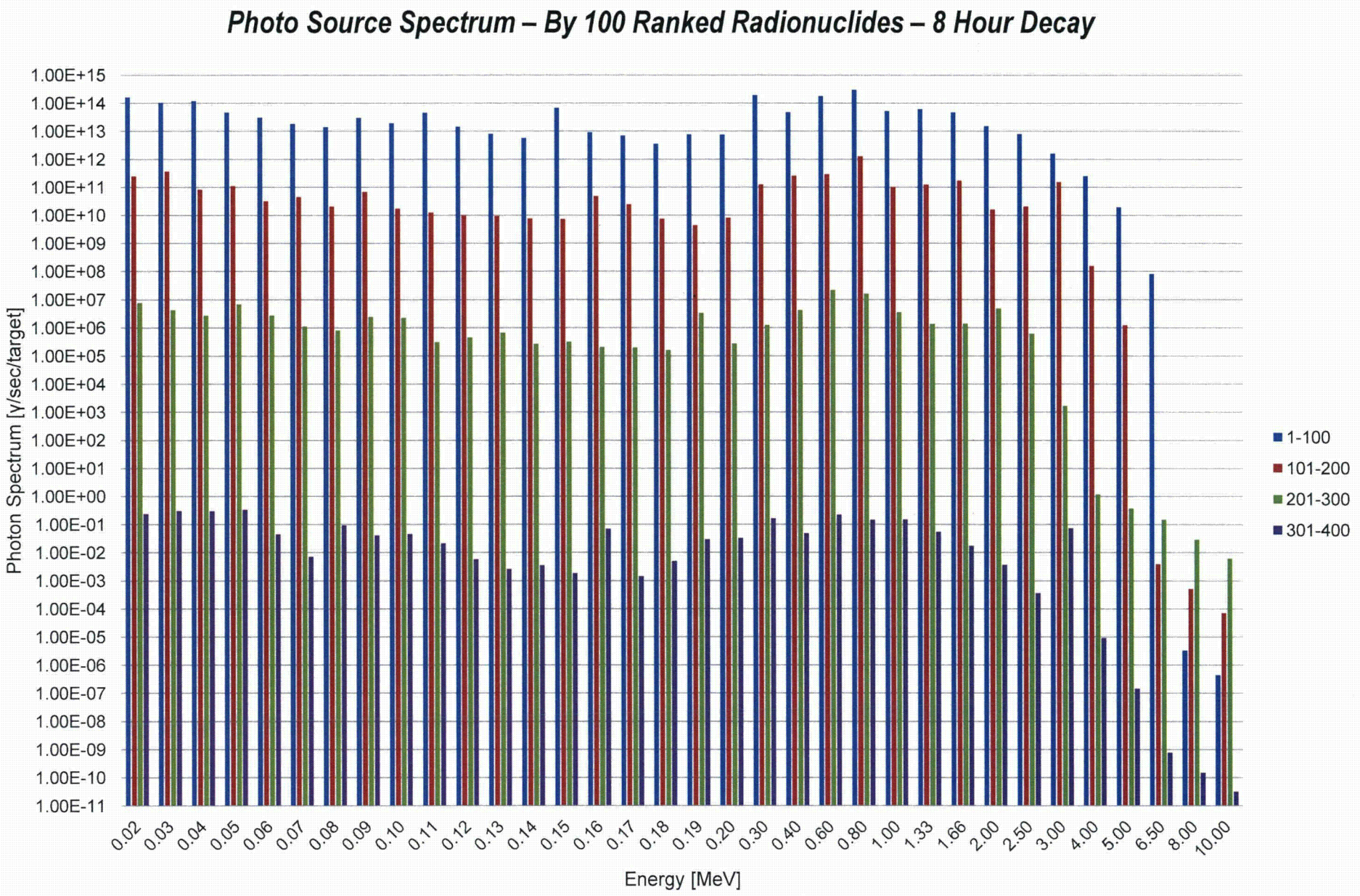
Shielding Design Analysis and Process

- Analysis
  - Supports ISA by providing dose rates for postulated radiological events
  - Verifies that dose rates will meet occupational dose rate limits (10 CFR 20.1201) and dose rate limits for members of public (10 CFR 20.1301)
- Process
  - MCNP model of process components
    - Materials, geometry, source term
    - Tallies, variance reduction
    - Calculation, post-process
    - 20-30 process models envisioned
  - Shield Wall Design
    - Difficult deep penetration problem requiring advanced variance reduction, elaborate source description
    - Hot cell penetrations will be analyzed

Source Term Generation

- Radionuclide inventory from detailed neutron transport and activation analysis
  - Provides radionuclide inventory at End-of-Bombardment (EOB)
- SCALE v6.1.3 ORIGEN-S decay calculation
  - Decay-only calculation provides inventory and gamma spectra vs decay time
    - Activation model is zero-dimensional and will not be used for activation analysis
    - Gamma group structure uses 34 energy groups
  - ENDF/B-VII.1 decay library
    - Corrects error in ENDF/B-VII.0 in ^{238}U decay chain (which skips ^{234m}Pa)
  - Bremsstrahlung photon libraries for betas produced in UO2 or water
    - Only concern in Target Fabrication Area
  - Neutron source will be calculated
    - Not expected to be important but will be quantified

Source Term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0.40
0.60
0.80
1.00
1.33
1.66
2.00
2.50
3.00
4.00
5.00
6.50
8.00
10.00
Energy [MeV]
1-100
101-200
201-300
301-400

Shielding Calculations

```
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 ab01

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( 0.000000, 0.000000, 1.000000)
origin:
(     0.00,     0.00,     0.00)
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Mesh tally dose rate [rem/hr]
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runtpe = ab01r
dump             11
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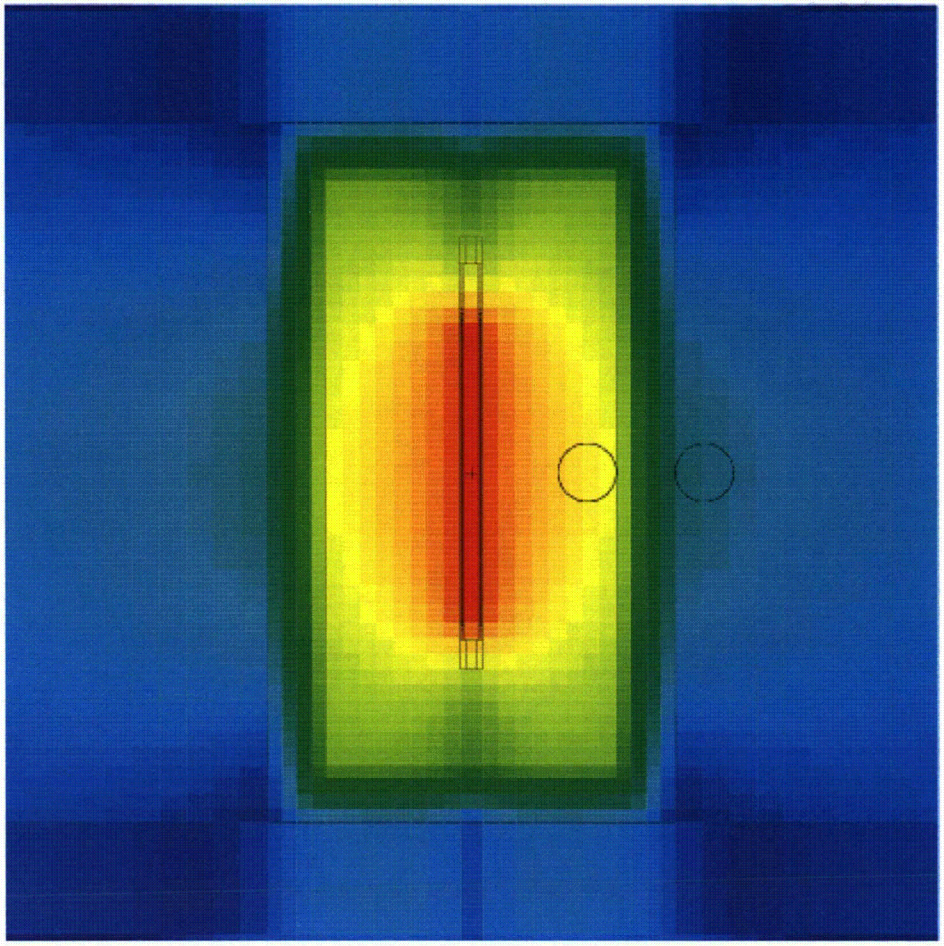

- MCNP6 v1.0 with mcplib84 photon library
- Detector, Track-Length, and Mesh Tallies

Shielding Related Analyses

- Inhalation Dose Analysis
  - Suitable air dispersion analysis code
- Dose from Criticality
  - 10 CFR 70 requirement for emergency response procedures
- Criticality Accident Detection and Alarm System (ANSI/ANS-8.1; ANSI/ANS-8.3)
  - Facility will be equipped with system to detect occurrence of a criticality accident and provide for prompt evacuation of personnel
  - Specialized analysis will be needed to determine accident detector type(s) and placement throughout the facility (will be included in final facility design)
  - Minimum Accident of Concern, Immediate Evacuation Zone established, Demonstrate effectiveness

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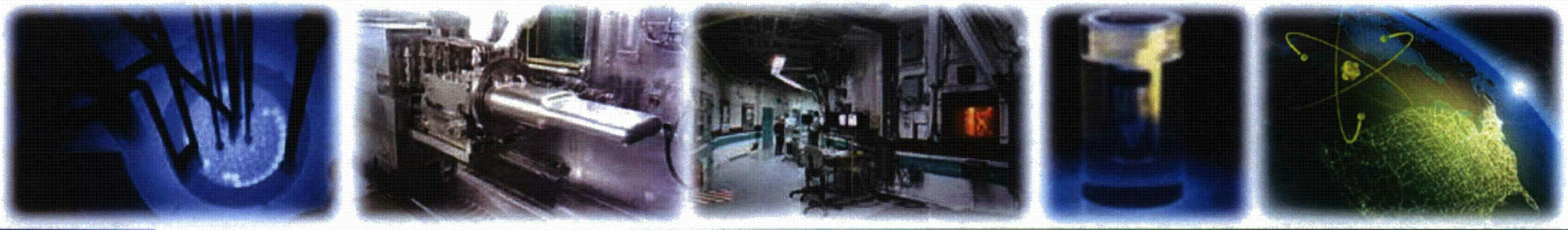
TRANSPORTATION

September 30, 2014

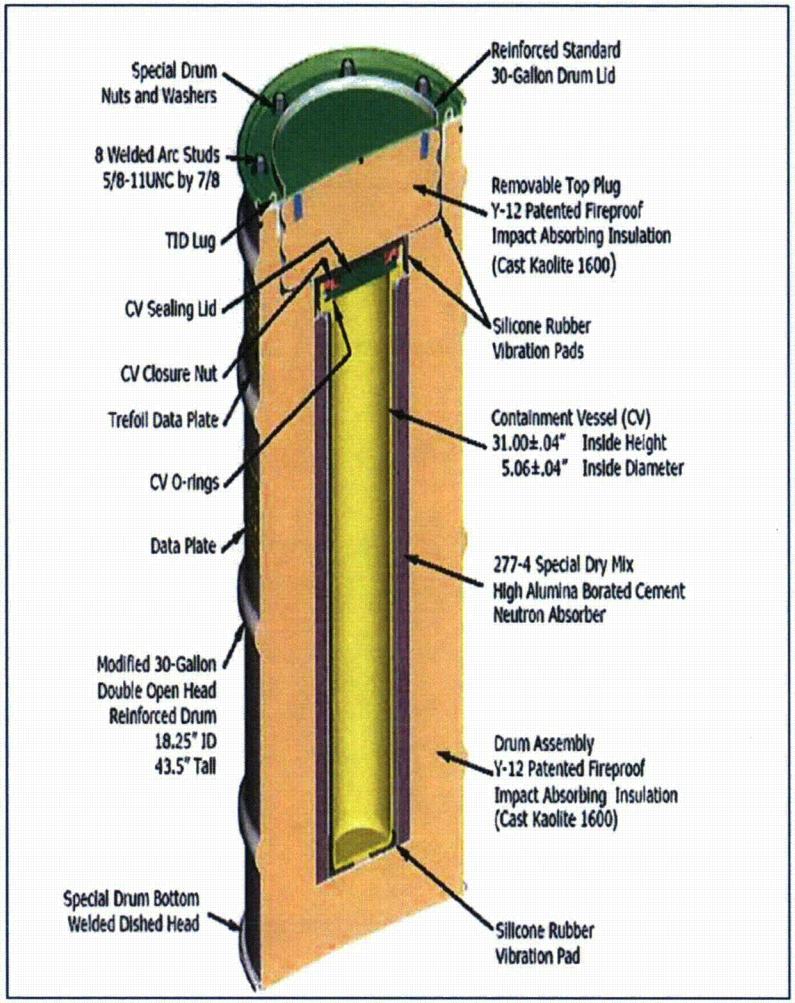


Radioactive Materials to be Transported

- Fresh LEU from DOE Y-12 Facility
- Unirradiated targets from RPF to University Network
- Irradiated targets from University Network to RPF
- ^{99}Mo product produced at RPF
- Spent LEU to DOE (location: TBD)
- Radioactive waste
  - All Waste types are expected to be Class C or Less

Fresh LEU Transport

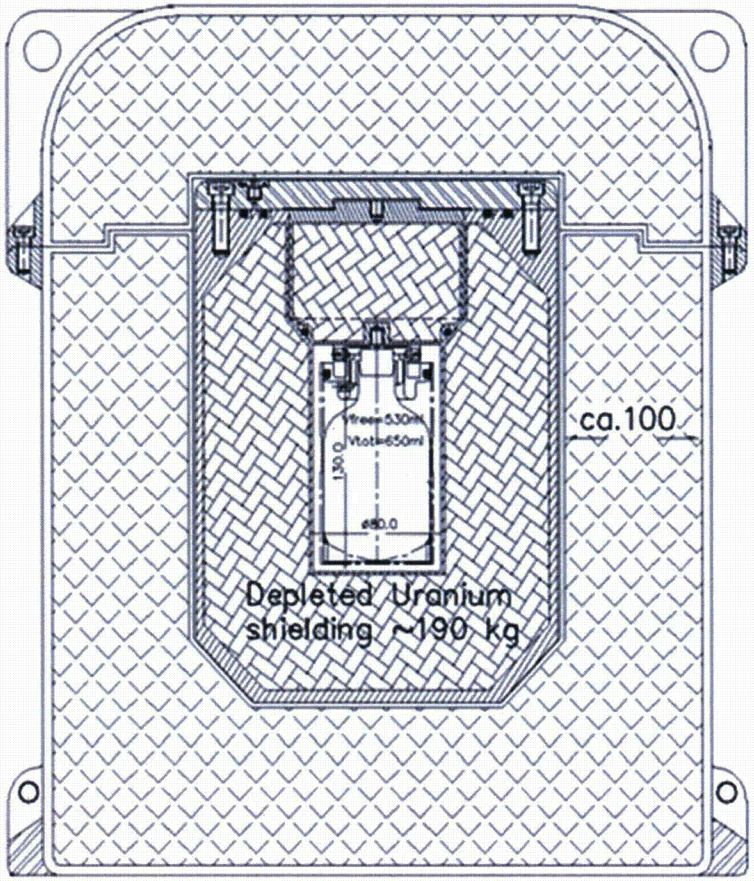
- Shipment Origin: USDOE Y-12 Facility
- Destination: Columbia, MO (Discovery Ridge) → ~590 Miles
- Transportation Mode: Ground Using Certified Cask
- Annual Shipments – 1
- Shipped via Panel Van as Exclusive Use
- Cask Recommendation – ES-3100
  - DOE Y-12 Program (Y-12) routinely uses ES-3100 and is currently licensed for LEU that will be used under the S.99 "Lease Take-Back Program"
  - Y-12 has many packages in inventory and maintains them in a ready-use state and maintains spare parts and has QA program for their use and maintenance
  - All procedures are in a mature state
- Spent LEU (e.g., recovered U3O8 from RPF) can use ES-3100 for returns to DOE



Schematic of ES-3100
Source: SER for Model ES-3100

^{99}Mo Product Transport

- ^{99}Mo Product
  - Shipment Origin: Discovery Ridge (Columbia, MO)
  - Transportation Mode: Air/Ground Using Certified Container
  - Destination:
    - Lantheus Medical Imaging - LMI (Billerica, MA)
      - ~8 Miles to Columbia Region Airport via ground transport
      - Air transport to Logan International Airport
      - ~31 Miles to LMI Campus via ground transport
    - Mallinckrodt (St. Louis, MO)
      - ~112 Miles via ground transport
  - Weekly Shipments – 2 each with 4-5 casks
- MIDAS Type B(U) Container
  - Currently used by current Mo-99 producers to both LMI and Mallinckrodt
  - Other potential containers include:
    - Nordian packages F-327 / F-245
    - DAHER TLI Versa-pack



Unirradiated LEU Targets Transport

- Shipment Origin: RPF in Columbia, MO
- Destination:
  - MURR (Columbia, MO) → ~6 miles
  - OSU (Corvallis, OR) → ~2063 miles
  - 3rd Reactor → TBD
- Transportation Mode: Ground using certified cask; bi-weekly shipment to Universities
- Recommendation
  - Several casks could be used for transport of unirradiated LEU ES-3100
    - Can be leased from Y-12 (minor modifications to current contents of CofC would be required)
    - Purchase casks from qualified vendor (need NRC approved QA Plan; Estimate 6-9 months obtain NRC approvals, develop procedures, train staff, etc.)
  - TRIGA Fuel Casks (owned by DOE) – TRIGA-1 and TRIGA-2
    - Currently only single DOE packages
    - Only two casks exists for both TRIGA-1 and TRIGA -2
    - Packages are difficult to manage and use
  - Cask identified for transport of irradiated targets
    - Issue – Potential contamination of unirradiated targets due to internal contamination levels in casks
    - Option can easily be bounded when license amendments are obtained for any option used for irradiated target transport
    - No additional time added to the project for licensing
    - Additional activities with cask would include preparation of the package for shipment loaded with fresh targets. (e.g., leak test, paperwork)

Irradiated LEU Target Transport

- Shipment Origin: University Reactor(s)
- Destination: Columbia, MO
  - MURR (Columbia, MO) → ~6 miles
  - OSU (Corvallis, OR) → ~2063 miles
  - 3rd Reactor → TBD
- Transportation Mode: ground using certified cask
- Weekly Shipments – 2+
- Cask Candidates
  - BEA Research Reactor Cask (BRR Cask)
  - NAC-LWT (full model)
  - Modified NAC-LWT Short Model
  - GE-2000
- Evaluations of existing transport package designs are being conducted to determine the most suitable and economically feasible candidate
- Currently there is no package licensed for the transport of NWMI's irradiated targets
- Minor modifications to existing licenses can accommodate irradiated targets with a 6-12 month licensing window (BRR Cask and NAC-LWT full model)
- NAC-LWT short model not yet licensed
- Each cask design being evaluated will require the use of a transfer system or upgrade to reactor facility infrastructure
- Additional basket loading strategies are being evaluated to streamline the loading and unloading process
- NWMI is currently in the process of determining which cask is most appropriate
  - Expect to have decision in 4th Q 2014

Radioactive Waste

- All shipments will originate from Columbia, MO
  - Waste Control Specialists (Andrews, TX) → 913 Miles
- Transportation of radioactive waste will be by ground transport in appropriate certified casks or shipping containers
- Solid waste/trash consolidated and shipped as LSA
- Higher activity solid waste (resins/zeolite) and liquid waste will be stored on site for decay, solidified and shipped as LLW
- ~12 shipments per year and will meet NRC/DOT requirements

Incident-Free Transportation Approach

- Incident-free radiological doses are determined for members of the public and workers that are involved in transportation of the LEU, irradiated and un-irradiated targets, ^{99}Mo product, and radioactive wastes (transportation workers and handling workers)
- Transport modeling code: RADCAT/RADTRAN
  - Calculates dose/shipment: members of public and transportation workers (drivers/handlers)
  - Key inputs: package dose rates, transport route and population density
- Highway route and distance traveled determined by use of route data from MapQuest and applicable GIS data available from ArcGIS software
- Census data files were used to derive the population density along the route (USCB, 2010)