Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
Radiation Protection and Nuclear Materials
Open Session

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Tuesday, September 22, 2015

Work Order No.: NRC-1912 Pages 1-439

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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RADIATION PROTECTION AND NUCLEAR MATERIALS

SUBCOMMITTEE - OPEN SESSION

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TUESDAY

SEPTEMBER 22, 2015

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:35 a.m., Dennis C. Bley, Chairman, presiding.

COMMITTEE MEMBERS:

DENNIS C. BLEY, Chairman of the Subcommittee

DANA A. POWERS, Member

HAROLD B. RAY, Member

JOY REMPE, Member*

STEPHEN P. SCHULTZ, Member

GORDON R. SKILLMAN, Member

ACRS CONSULTANT:

KORD SMITH

DESIGNATED FEDERAL OFFICIAL:

MAITRI BANERJEE

ALSO PRESENT:

ALEXANDER ADAMS, JR., NRR

JEFFERY BARTELME, SHINE

GREG CHAPMAN, NMSS

JIM COSTEDIO, SHINE

THOMAS ESSIG, Chesapeake Nuclear Services for NRR

MIRELA GAVRILAS, NRR

JAMES HAMMELMAN, NMSS

BILL HENNESSY, SHINE

CATHERINE KOLB, SHINE

MIKE LAUNI, Sargent & Lundy for SHINE

STEVEN LYNCH, NRR

JIM McILVAINE, Chesapeake Nuclear Services for NRR

JIM McINTYRE, Sargent & Lundy for SHINE

KEVIN MORRISSEY, NMSS

GREG PIEFER, SHINE

PAUL PRESCOTT, NRO

LOUIS RESTREPO, Atkins Nuclear Solutions US

OSIRIS SIURANO-PEREZ, NMSS

CHRISTOPHER TRIPP, NMSS

ERIC VAN ABEL, SHINE

ERNIE WRIGHT, Sargent & Lundy for SHINE

*Present via telephone

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P R O C E E D I N G S

(8:35 a.m.)

CHAIRMAN BLEY: This meeting will now come to order, please. This is a meeting of the ACRS Subcommittee on Radiation Protection and Nuclear Materials. I am Dennis Bley, Chairman of this meeting for SHINE Construction Permit Review.

ACRS members in attendance are Dana Powers, Ron Ballinger, Steve Schultz, and Rick Skillman. Joy Rempe will be joining us via the

telephone this afternoon. Our Consultant, Kord Smith is also here.

Ms. Maitri Banerjee is the Designated Federal Official for this meeting.

We have members of the SHINE Medical Technologies Team to brief the subcommittee regarding the construction permit applications for radioisotope production facility in the city of Janesville, Wisconsin for producing molybdenum-99.

We also expect to hear from NRR staff members regarding their review of this application. Several chapters of the SHINE application, chapters 6b, 11, 12 for the QA program, and 13b are scheduled for discussion today as noted in the agenda posted on the NRC meeting website.

The meeting will be an open and closed meeting, meaning parts of the meeting will be closed the public to protect proprietary information. We have designated a 45-minute session that may require us to be closed to the public toward the end of the meeting, as shown in the agenda.

We did not receive any requests from the public for time to make a statement.

We have two bridge lines established,

one for the public to hear the deliberations and an unpublished line to allow certain SHINE and NRC staff personnel to participate in the meeting remotely. The bridge number and the password for the first line were published in the agenda on the NRC website. To minimize disturbance, the public line will be kept in the listen-in only mode.

Before closing the meeting to the public, we will open the public bridge line to provide an opportunity for members of the public attending this meeting in person or through the bridge line to make a statement or provide comments.

Before we go into a closed session, I will ask the NRR staff and SHINE to confirm that only people with clearance and need to know are in the room. At that time, technicians in the booth will disconnect the public telephone bridge line.

For those of you on the private line who are participating in the meetings, keep your phones muted by dialing star, 6 and you can open yours up again by dialing star, 6 to minimize noise in the meeting.

For all participants, I will remind you that we now keep our personal microphones turned

off at the table, unless you are talking. You turn them on by touching right at the bottom of the microphone.

Dr. Corradini has a conflict of interest because of his work with the University of Wisconsin supporting the SHINE application and he recuses himself.

I now invite Dr. Mirela Gavrilas of NRR to introduce the presenters and start the meeting. Welcome.

MS. GAVRILAS: Good morning. You are going to hear a lot of continues in my brief remarks this morning because this is our third meeting on this topic, hopefully our last with the subcommittee. We continue to be on a very aggressive schedule to get this action completed. We now have a hearing date. As I communicated to you last time, the Commission decided to actually hold the hearing for the SHINE construction permit. We now have the date. It is going to be held on 12/15.

So, our objectives today are to continue with our overview of NUREG 1577, the particular chapters that have not yet been addressed. We are going to continue to introduce

new aspects of SHINE Technology and we are going to present to you what the objectives of the staff review have been.

Like in the previous two meetings, we would like your feedback on the staff's review of the construction permit and we would like to continue to flag issues that will require particular attention as the staff will review the licensing -- the operating license.

And with that, I will pass it on to SHINE, who will introduce themselves. Thank you.

MR. PIEFER: So, I'm the new face here but my name is Greg Piefer and I am the founder and CEO of SHINE Medical Technologies. I am actually just thrilled to be here. I got my Ph.D. in nuclear engineering from the University of Wisconsin and I remember hearing about things like this, never imagining I would be a part of them. So, it is very, very exciting to be here and to see the progress that both our team and the NRC team have made towards getting this every important project completed.

We have been working hard and they have been working very hard and I look forward to a productive meeting with you guys. Hopefully I will

be able to answer any of your questions and looking forward to your input.

I say to our staff all the time, and I really do mean it, it is a genuine feeling that I sleep very well at night, knowing that we have an independent regulator looking over our activities. And so I appreciate all your service and time as we go through the construction permit process.

So, thank you. I think you have met much of the rest of our team. We have a few consultants that will probably introduce themselves as we go through the hearing. Maybe just at the table we can go through a very brief who you are.

MR. COSTEDIO: Jim Costedio, I'm the licensing manager.

MR. HENNESSY: Bill Hennessy, the engineering manager.

MR. VAN ABEL: Eric Van Abel, nuclear engineer for SHINE.

MR. PIEFER: And the rest will announce themselves as they come up. So, thank you again. I look forward to a productive meeting.

MS. BANERJEE: Thank you. Folks out there on the bridge line, please mute your phones by dialing star, 6. We are hearing a lot of noise.

We are going to try to keep the line open, if we can.

MR. VAN ABEL: I'm Eric Van Abel and I am going to be discussing Chapter 6b of the SHINE PSAR, which covers radioisotope production facility engineered safety features and criticality control features in the RPF.

The ESFs in the SHINE facility are passive or active features. They are designed to mitigate the consequences of postulated accidents and ensure that radiological and chemical exposures are within acceptable limits. So, these are systems that act to mitigate the release and minimize consequences.

The criticality safety control system of Chapter 6b covers NCS controls for the RPF and SSCs, where uranium could be present and, therefore, where an accidental criticality is possible.

The IF is not within the scope of this facility -- of this section of the PSAR. The TSV and reactivity control in the TSV was covered in Chapter 4 of the PSAR.

These controls that are discussed in this section ensure that nuclear processes

throughout the RPF remain subcritical during normal and credible abnormal conditions.

First off, the engineered safety features in the RPF. As you will hear me go through, you will see the ESFs in the RPF are very similar to the ESFs that we discussed before with the irradiation facility. The same concept is applied.

There are five design basis accidents categories are addressed for the RPF. Three of those five, plus the MHA require ESFs to mitigate consequences. The categories are critical equipment malfunction, RPF Fire, and accidents with hazardous chemicals.

The confinement system at SHINE is a low-leakage boundary that surrounds radioactive materials or hazardous chemicals produced from licensed materials that could be released during an accident.

Similar to the IF, the confinement boundaries provided by the structure, the cells themselves, and principally, the ventilation ductwork servicing those cells in the isolation functions of that ventilation system.

ESF functions for the RPF are provided

by the barriers, the active valves and dampers, and in this case, the Radiological Integrated Control System is the actuating I&C system.

MEMBER SKILLMAN: Eric, I would like to ask this question. Since this is a construction permit activity that we are involved here in today, it is essential that before concrete is poured or when it is being poured, or when components are being procured, that that concrete is poured properly, that the components are procured properly.

So, you don't have a containment. You have a confinement. And the confinement is the concrete but it is also your system's RCA Vent Zone 1 and 2 and that is what is being depended upon to prevent a release.

Here is my question. At this early stage, what is your vision of what is nuclear safety related?

MR. VAN ABEL: The safety related, every piece of that boundary, the confinement boundary that is required to mitigate the accident, the walls, the concrete walls, themselves, the ventilation ductwork and the dampers themselves would be safety related.

MEMBER SKILLMAN: Okay, so we have got walls. We have got ductwork. We have got dampers. What about supports and restraints?

MR. VAN ABEL: Where necessary to ensure that those components maintain their integrity, supports and restraints that would be --

MEMBER SKILLMAN: What about bolts, fasteners, mollys that go into the concrete to hold the ductwork in place, devices that are in tension, such as steel rods that are threaded?

What I want to get to here is something that we will talk about in the QA program in a couple of hours that really ties into what you are crediting here for safety in Chapter 3 -- excuse me -- in Chapter 6. What you described in Chapter 3 is your components and how they are assured to be what they are supposed to be in Chapter 12c, which is your QA program description.

But what I am asking you is what is the extent of the safety devices, please?

MR. VAN ABEL: Any component that is necessary for that safety SSC to perform a safety function. So, if the restraints are necessary to support it during an earthquake or other postulated event, then they would have to be safety related as

well.

MEMBER SKILLMAN: Are the fasteners safety related?

MR. VAN ABEL: They have to perform a safety function, yes.

MEMBER SKILLMAN: Is the bulkhead to which these attach safety related?

MR. VAN ABEL: Yes, if it is required to maintain its integrity during the event, yes.

MEMBER SKILLMAN: Thank you.

MR. VAN ABEL: Next slide, slide 5 in the packet. Specifically, the confinement functions are provided by the principle components listed here. That is the reinforced concrete shielding of the hot cells, tank vaults and pipe trenches that is not only very thick radiological shielding approximately four feet thick in most cases, and there are penetration shields to those confinement boundaries will also be present to minimize leakage. Isolation valves on piping systems, RCA Ventilation Zone 1 and Zone 2 components, the isolation dampers themselves on both Zone 1 and Zone 2, and RVZ1 ductwork and filters that support that. And of course any supporting components that support that as well.

And then in this case, the RICS, Radiologically Integrated Control System, is a system that would sense the high radiation levels and actually the isolation of a particular cell.

The SHINE facility protects public health and safety via a confinement system. It is important to note that the inventory that SHINE is confining in each confinement area is approximately 10,000 times less than a typical power reactor with confinement.

And also the SHINE processes are generally low temperature low pressure processes. There is not a lot of stored energy in there where there is low dispersion forces to drive releases.

Next slide. On actuation of engineered safety features, normally when the cells are operating, the hot cells are maintained at a negative pressure, resulting in leakage there. So, if there is any contamination normally present in the cells, there would be an in-leakage of air to reduce the contamination being spread outside of the Zone 1 boundaries.

In the event of a DBA that releases radioactive material into the cell, that would be transported into the ventilation system and the

high radiation detected in the ventilation system would initiate confinement isolation of that cell.

The confinement isolation signal closes the isolation dampers on the inlet and outlet ports of the cell. We have redundant isolation dampers on inlets and outlets. So, either one of those two valves closing would be sufficient.

Dampers are also fail-safe. They close on loss of power. And the specific ESF actuation threshold will be determined during detailed design and they will be set low enough to ensure that we are below 10 CFR 20 limits with sufficient margins during detailed confinement.

The SSCs that perform ESF functions are safety-related and will meet the single-failure criterion. So, we apply the single-failure criterion to these systems so that any single act of failure does not compromise the capability of the system from performing its safety functions.

Duct and housing leak rate tests are performed in accordance with ASME N511 and the specific leak rates that are acceptable for each confinement area will be determined based on the final safety analysis.

The bubble-tight isolation dampers will

maintain their functional integrity during normal operations and accident conditions. They will maintain acceptable leak-tightness during the Design Basis Earthquake. They will maintain their structural integrity under plant shut-off pressure and they also provide damper position indications for the operators. The operators know what position they are in.

Low leakage seals are provided on each penetration and the overall leakage rates will be measured and tested of those cells prior to and during operations.

MR. VAN ABEL: Eric, let me ask this. You have just described several very important SSCs that are nuclear safety-related. You also mentioned that those dampers will be tested. Is the testing -- do you anticipate that the testing of the damper will be a safety-related activity?

Where do you describe the safety-related activities? You make a very strong case for your SSCs. There is no definition of safety-related activity either in your QA program or in Chapter 3, where you laid out your SSCs. What is a safety-related activity?

And if that was on your construction

activities, because if that definition isn't square with what you are doing, you can conceivably have construction activities that should have been a safety-related activity and was not.

What is a safety-related activity? The term is used throughout your documentation.

MR. COSTEDIO: We'll have operating procedures when the operators manipulate systems, safety-related systems. That would be a safety-related activity. Surveillance testing, that would be safety-related activities.

MEMBER SKILLMAN: But we are at the construction stage. Would pouring concrete be a safety-related activity?

MR. COSTEDIO: Sure. Yes, it is a safety-related component. Every activity you are doing before it where we have to apply the QAPD, then it would be a safety-related activity.

MEMBER SKILLMAN: Do you anticipate changing your QA program to identify what are safety-related activities?

MR. COSTEDIO: We are probably implementing procedures that would do that.

MEMBER SKILLMAN: Okay, thank you.

MR. VAN ABEL: Next slide. As we were

discussing, periodic testing of ESFs will be performed to be sure they can perform their safety functions. Penetration seals, isolation valves, isolation dampers, gloveboxes, and other components that are relied upon for confinement will be tested prior to and during operations. And the specific testing that will be performed and the testing intervals will be specified in the technical specifications in the OL application.

There are no emergency cooling systems for the RPF processes because none are required. Following loss of our process chilled water systems requiring cooling are shut down until cooling can be restored.

The only process component that needs to continue to function in the RPF is the PVVS blower and that continues to operate on the UPS, uninterruptible electrical power supply.

The PVVS blower is small and it is not expected to require any chilled water or other forced cooling applied to it.

The fission product decay heat removal requirements of the batches of target solution that are in the RPF on a loss of offsite power or loss of chilled water, those decay heat requirements are

minimal and the tank would cool simply through natural convection.

Next slide. The criticality safety in the RPF is also covered in Chapter 6b of the SHINE PSAR. The design of the RPF and NCS program will ensure that inadvertent nuclear criticality is highly unlikely.

The NCS program in the facility will be designed to the listed ANSI/ANS standards there as modified by Regulatory Guide 3.71.

Next slide. The program will contain a number of element that includes the policy statement, which is the high-level requirements for the program; the V&V requirements for software that is used for criticality safety analysis; NCSE requirements, the nuclear criticality safety evaluations, will be performed during detailed design and the requirements for those NCSEs will be specified in the program; training and qualifications for staff at the facility, overall staff at the facility, as well as the nuclear criticality safety engineers; implementation of critical safety controls and limits; and the configuration control requirements, audits and inspections necessary; non-compliance processes for

criticality safety; guidelines for firefighting; the EP plan and response procedures manual requirements relevant to criticality safety -- we are trying to have one overall EP plan but the requirements are specified for criticality safety in the criticality safety program; the criticality detection and alarm system requirements; testing and calibration of active controls; and the overall criticality safety controls program.

MEMBER SKILLMAN: Eric, should we consider that this list on your slide 10 is a listing of safety-related activities?

MR. VAN ABEL: There are some aspects there that would support, as Jim was saying before.

MEMBER SKILLMAN: So how would a construction reviewer know what is your Q1 and what is not Q1?

MR. COSTEDIO: Those procedures that the staff -- they will be audit how we are implementing our procedures, the sign-offs, the check-offs, our records, our configuration.

MEMBER SKILLMAN: Okay, thank you.

MR. VAN ABEL: All right, then slide 11. The design of the RPF will adhere to the double contingency principle, DCP. And as I

mentioned before, the NCSE will be performed in detailed design and those will go through the DCP and the various systems.

The NCS training program will be developed and implemented for personnel in the SHINE facility. There will be a task analysis for worker job functions to give appropriate level of training for specific functions. And the NCS staff, themselves, will be trained and qualified in accordance with ANSI/ANS-8.26.

The CAAS, Criticality Accident Alarm System, will provide for detection and annunciation of criticality accidents in the SHINE facility.

As we discussed in Chapter 7, the CAAS that provides coverage for each area requiring coverage with at least two detectors. CAAS is safety-related and powered from UPS, the uninterruptible electrical power supply.

Personnel will be trained to identify the unique criticality accident alarm quickly and evacuate safely through the most direct zone.

Prior to implementing changes that could involve special nuclear material in the SHINE facility, it must be determined that the processes will remain subcritical, given the approved margins

of subcriticality for both normal and credible abnormal conditions.

SHINE's configuration management program will include criticality safety controls. Criticality safety controls will not be changed without the review of qualified criticality safety engineers.

And the NCS controls will be specified in operating procedures and equipment drawings explicitly marked as such, to ensure that they are not inadvertently changed without review.

10 CFR 50.59, of course, will be applied and used to be determined if a license amendment request is required also for changes.

SHINE will follow the technical practices for each controlled parameter as described in the ISG augmenting NUREG-1537, Part 1, Section 6b.3. For any of those controlled parameters that are not controlled for a particular process, SHINE will apply the most reactive conditions in the analysis for those parameters.

NCSEs and analyses and supporting calculations will be used to identify which particular parameters within a system are required control sand what those controls are.

SSCs that are identified as safety controls will be safety-related in the SHINE facility. Safety-related SSCs receive the full measure of the SHINE QAPD and the administrative controls for criticality safety will be implemented through our facility procedures as described in our operating license application.

And per the Technical Specifications, SHINE has written procedures to implement and maintain the criticality safety program.

The preferred means in the majority of the process in the SHINE facility are maintained subcritical by passive engineered controls, especially geometry of tanks, piping, and vessels.

We use subcritical by design vessels and piping as much as possible in the RPF processes. Each of the RPF process tanks, with the exception of the liquid processing tanks, are criticality safe by geometry.

CHAIRMAN BLEY: For any temperature --

MR. VAN ABEL: Yes.

CHAIRMAN BLEY: -- and weather conditions?

MR. VAN ABEL: The tanks are --

CHAIRMAN BLEY: What kind of controls

do you envision having to make sure that you won't ever get the solutions into other than these criticality safe tanks?

If you read through the criticality accidents at facilities, most of them have these kind of things. Somewhere along the line they end up putting stuff in a different tank than they were supposed to. Have you thought much about how you are going to deal with that?

MR. VAN ABEL: Yes, in the preliminary process, we went through a HAZOPS, Hazards and Operability Study, and looked at potential ways that the solutions could be moved into an unexpected location in the processes and looked at inadvertent criticalities. That way, also with that HAZOPS and the preliminary hazards analysis we perform, we looked at spills of tanks and vessels -- - leakage from tanks and vessels, and where that solution could go in those cases.

CHAIRMAN BLEY: Do you intend to redo the HAZOPS once you have the --

MR. VAN ABEL: Detailed design.

CHAIRMAN BLEY: How about once it is built? I mean the way you usually do a HAZOP is to walk through a facility and look. And things might

not end up the way you envision them when you are looking at drawings. Is that on the plan?

MR. COSTEDIO: It will be on the plan. I mean yes.

CHAIRMAN BLEY: That's something we will think about when we get closer to the time with meeting those plans. But we will be interested from that.

MR. VAN ABEL: For the waste tanks, we measure the uranium concentration and independently verify that uranium concentration before that material transfers to the waste processing tanks, to verify there is no appreciable quantities of fissile materials.

The pipe runs throughout the facility are single-parameter criticality-safe by geometry. They are smaller than the subcritical cylinder diameter. And the criticality-safe tank vaults are connected to a criticality-safe sump catch tank. So, if there is a leak in any of the process tanks containing such raw materials as uranium to another tank, that is criticality-safe by geometry.

And that's all I have for the criticality safety portion.

MR. SMITH: I had one general question

on criticality. When you fill your tank, you are getting a cold condition that is, at least I saw reference, you are getting very close critical. [Redacted] I think what I saw as the maximum in cold condition. And I didn't see anything other than monitoring the 1/M curve to tell you how close you are.

I didn't see anything how fast the reactivity comes down in the open dump-outs versus the reactivity rate which you are adding. Is that waiting for the final design?

MR. VAN ABEL: I think you are referencing the TSV design and TSV K-effective, which is, the actual K-effective is proprietary, just for future reference there.

But the rate of reactivity decreased from opening the dump valves in the TSV will be greater than the rate at which it possibly could add reactivity to the system. The actual K-effectives for these tanks in the RPF will be below the upper subcritical limits calculated for the upper subcritical limits. And the actual K-effectives will be less than 0.94, as I think the staff had talked about in a little bit.

MEMBER SKILLMAN: Eric, let me ask this

question. Are the actions to produce those calculations that you just explained to Dr. Kord safety-related activities?

MR. HENNESSY: Yes, those actions are performed in accordance with our quality procedures, including review and approval. The activities required to assure that these calculations meet our highest quality levels are all present.

MEMBER SKILLMAN: So, the answer is yes, those are safety-related activities by your definition.

MR. HENNESSY: Right.

MS. BANERJEE: I will just mention I do have some comments I received from members who weren't with us. Keep in mind that other areas of the document, the PSAR, that they apply here as well. There is a lot of places where there are assumptions but when we get moving toward an actual operating license, we will need to see the basis for all of those.

MR. LYNCH: All right, thanks for having us back. We're ready to talk. I will let everyone introduce themselves. We are here with NRR and NMSS to talk about engineering safety

features and nuclear criticality safety at the SHINE facility.

MR. TRIPP: I'm Christopher Tripp. I am the criticality safety reviewer. I actually work for NMSS not for Sergeant & Lundy or SHINE, as it said in the agenda.

And first we are going to talk about the engineered safety features in the confinement. I am going to let Osiris Siurano discuss that first.

CHAIRMAN BLEY: Is your mike on?

MR. SIURANO-PEREZ: Good morning. May name is Osiris Siurano. I am kind of the backup project manager for SHINE with NMSS. This chair is here for Mary Adams who is actual project manager for the project. She is out on personal business today. So, I will be replacing her for this presentation.

In Chapter 6b of the preliminary -- sorry.

MR. ADAMS: Al Adams from Research and Test Reactors.

MR. SIURANO-PEREZ: Sorry. In Chapter 6b of the preliminary safety analysis report, SHINE provided a description of the confinement

engineered safety features for the radioisotope production facility. SHINE described the structures, systems or components which constitute the confinement engineered safety features in the radioisotope production facility design and summarized the possibility of accidents, consequences to be unacceptable without mitigation.

Engineered safety features are mitigative, not preventative. Specific postulated accident scenarios indicate the need for the confinement engineered safety features.

SHINE identified three design basis accidents that required mitigative engineered safety features. These design basis accidents or DBAs encompass loss of offsite power and operator errors. The confinement engineered safety features, structures, systems or components provide active and passive protection against the potential release of radioactive materials or chemicals to the environment would be a design basis accident.

The confinement engineered safety features, structures, systems, or components provide for active isolation of piping and heating, ventilation, and air conditioning systems that penetrate confinement boundaries in certain post-

accident conditions.

The confinement engineered safety features consist of these five structures, systems, or components: Radiologically Controlled Area Ventilation Zone 1; Radiologically Controlled Area Ventilation Zone 2 isolation dampers, ductwork, up to filters, and filters; Radiologically Integrated Control System provides confinement isolation signal; isolation valves on piping systems; and hot cells, tanks, tank vaults, and pipe trenches.

The radioisotope production facility confinement areas include hot cell enclosures for process operations and trench and vault enclosures for process tanks and piping.

Confinement is achieved through Radiologically Controlled Ventilation systems, the Radiological Integrated Control System and biological shielding provided by the steel and concrete structures comprising the walls, roofs, and penetrations of the hot cells.

Shielding of the hot cells is described in detail in the Preliminary Safety Analysis Report subsection 4B.2.

In the event of a design basis accident that results in a release inside the hot cells,

radioactive material will be confined by the biological shield and physical walls of the cell, itself. Each line that connects directly to the hot cell atmosphere and penetrates the hot cell is provided with redundant isolation valves to prevent releases of gaseous or other airborne radioactive material.

To mitigate the consequences of an uncontrolled release occurring within a hot cell, as well as the outside consequences of releasing fission products through the ventilation system, the confinement barrier utilizes an active component in the form of bubble-tight isolation dampers, which are safety-related on the inlet and outlet ventilation ports of each hot cell.

These dampers close automatically upon loss of power or receipt of an confinement isolation signal generated by the Radiological Integrated Control System.

Fire initiating event of the Radiological Integrated Control System isolates the hot cells.

Upper performance assurance of the IT confinement component is achieved through factory testing and in-place testing. Duct and housing

leak tests will be performed in accordance with American Society of Mechanical Engineers, ASME Standard N511 with minimum acceptance criteria as applied in ASME Standard AG-1.

The design of safety-related systems including protection systems would be consistent with Institute of Electrical and Electronic Engineers Standard 379-2000 and NRC Regulatory Guide 1.53 in the application of the single-failure criterion.

Bubble-tight isolation dampers will be designed, constructed and tested in accordance with ASME Standard AG-1, Section DA: Dampers and Louvers. Dampers will be butterfly type, blade on frame, fabricated of heavy gauge stainless steel. Total leakage based on bubble solution test as outlined in the ASME Standard AG-1-2009 Section DA.5141.

Ventilation duct and ductwork support materials will meet the requirements of American Society of Mechanical Standard AG-1, Article SA-3000 materials.

Supports are designed and fabricated in accordance with the requirements of American Society of Mechanical Engineers Standard AG-1,

Section SA: Ductwork

The engineered safety features would be tested to ensure that engineered safety feature components would well maintain operability and can provide adequate confidence that a system will perform satisfactorily in service during postulated events.

The NRC staff examined the accidents in areas analyzed in Chapter 13b that could lead to significant radiological or chemical exposures or releases and verified that consequences can be sufficiently mitigated by the confinement engineered safety features.

The NRC staff determined that Section 13b of the Preliminary Safety Analysis Report contains sufficient information to conclude that scenarios for potential accidents of a radioisotope processing facility with consequences greater than the design basis have been analyzed by the applicant. Mitigation of consequences by a confinement system have been proposed in the Preliminary Safety Analysis Report analysis for any accident that could lead to potential unacceptable radiological or chemical exposures to the public, the facility staff, or the environment. The NRC

staff reviewed the designs and functional descriptions of the confinement engineer safety features. The designs and functional descriptions reasonably ensure that accident consequences will be limited to the levels found acceptable in the accident analysis of Chapter 13b.

The NRC concluded that the designs and functional descriptions of the confinement engineered safety features reasonably ensure that control of radiological and chemical exposures or releases during normal operation will not be degraded by the engineered safety features.

The NRC staff determined that the radiological consequences from accidents to the public, the environment, and the facility staff will be reduced by the confinement engineered safety features to values that do not exceed the applicable limits of 10 CFR Part 20 and the chemical exposure criteria specified in Section 3.5(b) of the Preliminary Safety Analysis Report.

With regards to containment, Section 6b.2 of the Preliminary Safety Analysis Report states that the SHINE radioisotope production facility does not employ a containment feature due to a low temperature and power level of facility

operations. The safety analysis demonstrates that confinement features are adequate to mitigate potential accidents. NRC staff agrees that containment is not required for normal operation or accident mitigation.

The safety analysis in Chapter 13b of the Preliminary Safety Analysis Report show that confinement provides sufficient mitigation and containment is not necessary.

There is no emergency cooling system associated with a radioisotope production facility. So, based upon review of the accident analysis provided in Chapter 13b of the Preliminary Safety Analysis Report, the NRC staff agrees that no emergency cooling system is needed for the radioisotope production facility.

Thank you very much.

MEMBER SKILLMAN: I would like to ask this question please to the staff. When you review very carefully the drawings that are part of the PSAR, particularly the drawings that are part of Chapter 6 that show the ventilation systems that you just described as adequate to protect the public, there is no indication on those drawings of where the breaks are between Q1 and Q2.

So, my question is how did you conclude that this system, these systems are adequate?

MR. CHAPMAN: I think we looked primarily at the design criterion that they are going to use. And we are going to be that way or putting Quality Level 1 to that.

So, we didn't really look at specifics as to what is designated Quality Level 1 but we know that their intention is to do that.

MEMBER SKILLMAN: Well, let's talk about that a little. We are at the construction stage, construction permit stage. Isn't it imperative that we know now how they are going to construct their ventilation systems? I mean once the concrete is poured, it is poured. Their opportunity to change the structure has just ended, unless they end to chip out a lot of concrete.

CHAIRMAN BLEY: Before you respond, we need to get your name on the record.

MR. CHAPMAN: Greg Chapman, NMSS.

CHAIRMAN BLEY: Thanks.

MR. CHAPMAN: We noticed that there are a lot of detail design that is not present in these drawings. So, we had to make our conclusions based on all the written tests.

MEMBER SKILLMAN: So do we. So do we, ACRS. We have got to make our decision based on those, too.

So, what I am asking is. This is a construction permit. Both of us are involved in here. And unless this is constructed properly, the SHINE team is going to face major changes. And so unless this has been constructed properly with the design breaks where they need to be, with the quality breaks where they need to be, then we can be complicit in proving something that really isn't what we wish it to be.

MR. CHAPMAN: Well, certainly can put a condition in there, if we need to, to review the drawings before they are constructed.

MEMBER SKILLMAN: I made my point. Thank you.

CHAIRMAN BLEY: Just a quick question. You referred a lot to the accident analysis here. When we get to the accident analysis, we will have more details at that point? Here, it was mostly just referring to it.

MR. SIURANO-PEREZ: Yes, yes.

CHAIRMAN BLEY: Okay, it'll wait until then. Thank you.

MR. TRIPP: Now, we are going to talk about the review of the criticality safety for the radiation protection facility. This doesn't pertain to the Target Solution Vessel but everything in the Radiation Production Facility will maintained subcritical with a minimum margin of 0.05, as we will talk about when we get to the validation shortly.

We based that, although this is an R Part 70 facility, we basically applied the same criteria we would apply based on the ISG to NUREG-1537, which was based off of our NUREG-1520 for fuel production facilities. And it consists of commitments to the NCS Program, which will completely design and come up with the safety controls that are going to need to be constructed into the facility and that includes commitments to the principal design criteria and design bases that we refer to in the ISG as technical practices.

And the base requirements, which are similar to those that you would have for a Part 70 fuel facility are that they must maintain subcriticality under normal and credible abnormal conditions, including an approved margin that we will talk about shortly; compliance with double

contingency principle; and a provision for a criticality alarm system.

MEMBER POWERS: Can I ask a question about the subcritical under normal and credible abnormal conditions? Do credible abnormal conditions include protracted lay-up of the facility?

MR. TRIPP: Protracted what?

MEMBER POWERS: Lay-up. To get a bad finding, they have got to spend some time. They have to go talk to the Commission about why they ought to continue to operate less for say six months. Does that include that when you think about criticality safety?

MR. TRIPP: I'm sorry, I can't hear.

MR. LYNCH: The facility is shut down for an extended period of time. Is that included?

MR. TRIPP: In shutdown?

MR. LYNCH: Yes.

MR. TRIPP: I would think it would apply to all modes, both operating mode when they are in maintenance and also shutdown. Yes, certainly for any operating mode of the facility, which I would assume would be covered by tech specs.

CHAIRMAN BLEY: Let me pursue that just a little bit. This is a little bit like what Dick was talking about before and it was brought up at a previous meeting.

If you don't build into the system before it is constructed this kind of capability to have a problem develop or a regulatory problem develop that causes you to interrupt operations and have the ability to move fluids into a stable place for the long-term, can't pull things up to restart, it could be a major problem later.

I, personally, know of at least two facilities, one of which operated for a very short time and then was shut down because they hadn't thought this through ahead of time, and another one that was completely broke and finally never operated because they couldn't handle this kind of an interruption. And if you don't do it before you pour the concrete, it is really hard to do it later. So, I don't think we have heard much from either the applicant or from the staff thinking about this kind of situation. Maybe it is, in this case, there could be safety aspects as well but there are certainly operational aspects that are crucial to this thing to be working in long-term.

MR. TRIPP: I will say the same thing that was just said recently and probably applies to a bunch of other areas. We don't have a lot of details about the detail design. So, what we are looking at here is the program that will do the safety analysis and the design criteria they are going to use.

I would presume that we would look at the detailed design prior to reviewing the operating license. And we actually have the proposed condition that will get into that.

MEMBER POWERS: The difficulty we are having or I am having is exactly what Dick was talking about. Once you pour the concrete, you are kind of stuck. And I am not seeing attention to protracted lay-ups of the facility the way I would have expected. That is, how to get everything cleaned out so I don't have liquids precipitating obscure and unanticipated solids in places that I didn't ordinarily expect them to be and they would not be under normal operations for short periods of time but they appear at long periods of time. And then it gets me into trouble possibly during the lay-up but more likely it is an accumulation that causes me headaches when I restart.

That usually entails some feature of the design to assure you get the facility completely clean if you are going into a protracted lay-up. I am just not seeing attention to that question. And it seems like it is a question that because it can involve tanks, and pipes, and things like that, that you would want to pay attention to before you poured some concrete.

MEMBER SKILLMAN: Let me build on Dr. Powers' question, please. In your safety analysis that you have reviewed as 6b31, it is on page 6b-16 of the PSAR, 6b-16, the statement if the CAAS -- excuse me -- the CAAS is not a control from the perspective of criticality control, however, the CAAS is considered safety-related. I would think it should be.

Is the geometry of the detector or the detectors for the CAAS safety-related? Is the design of the layout of the detectors to the source being detected a safety-related activity? And in your review, did you look at that? And is there concrete that could interfere if it is not poured properly?

MR. TRIPP: Well, the answer is we did not look at that. We don't have a detailed

description of the Criticality Alarm System at this point. And I would expect it is possible concrete or other shielding could interfere with that.

And so you will see that we have a proposed condition that we get to look at that prior to the installation of the Criticality Alarm System. That is the reason for having that condition that is being proposed by the staff, exactly what you are saying.

MEMBER SKILLMAN: Fair enough. Thank you.

MR. TRIPP: So, I'll describe exactly what the staff looked at. Like I said, we primarily did a programmatic review of commitments for the criticality safety program. We reviewed Section 6b.3 of the PSAR, as supplemented by various responses to RAIs. And there were a series of other supplemental information that was reviewed, criticality validation report, an NCS manual, which is basically a more detailed description of the criticality safety program.

And we did look at some preliminary safety evaluations, although we didn't look at them in any great amount of depth because we didn't receive them until the end of July.

We looked at principal design criteria, design bases to provide reasonable assurance of subcriticality combines with double contingency. So, both like we just talked a minute ago.

The application does have a commitment to Criticality Alarm System, to a program. There are certain elements to the program that are applicable to the design and construction, basically, those that involve development of criticality safety evaluations, which will be the basis for limits and controls, which we have not looked at this point; management measures that are applicable to design and construction, such as configuration control; making sure that we have proper staff to perform the evaluations and so forth.

And there are a number of ANS standards that apply to this that were alluded to previously.

Commitments to design criteria, some of the principle design criteria are compliance with double contingency principle; determination of upper safety limit for maximum K-effective to be calculated by the code because it is the codes that are going to be used to derive the limits and controls for the facility, such as dimensions and

equipment and so forth; and also, ensuring that all credible criticality events are rendered highly unlikely. Although, they are not required to follow the performance requirements out of 70.61, we still had to have some kind of a standard that would be applied during the detailed accident analysis.

Now I am going to describe a few of the key issues that arose during the review, the principle one of which concerns this issue about having a validated computer code to do the analysis. That is the first thing you have to have in place before you can do any of the safety analysis to demonstrate your subcritical.

And safety limits are based on controlled parameters, using the well-established computer code MCNP and it has to be validated using critical benchmark experiments. And those are drawn from the International Handbook for Evaluating Criticality Safety Benchmark Experiments, which is a large program managed out of Idaho National Laboratory.

The difficulty arises because now we are operating a facility or contemplating a facility at 20 percent enrichment, using the uranyl

sulfate-based solution process, which is something we don't see in the rest of the fuel cycle industry. There are very few benchmarks that have those properties, that chemical form and very few that are up around 20 percent enrichments.

The closest experiments to those under-predicted K-effective by about 2.9 percent. And after evaluating them, SHINE had originally discarded them as outliers. The difficulty that we faced for that was that outliers, as our guidance in Reg Guide 3.71, which endorses in the ANSI standard 8.24 on criticality code validation, says that outliers should only be rejected based on an inconsistency of the data with known physical behavior. And the only reason -- there were some suspicion there was some difficulties with the experiments but they are closest experiments to the actual conditions that will be encountered and they all under-predicted K-effective.

So, I'm going to slip ahead to the next slide for a minute. And we will see a graph of all of the critical benchmarks that we used for SHINE as a function of enrichment. Shown here, SHINE is just barely just under 20 percent enrichment. And before experiments that under-predict that were in

question and were discarded as outliers are those that are indicated in red.

The result is, if we calculate the -- if you include those four benchmarks, then the upper safety limit goes from the red dashed curve to the blue dashed curve. The dashed and the solid curve are two different ways of calculating the upper safety limit that are commonly used in the industry.

So, it does produce an effect where if you include these benchmarks, it could reduce the upper safety limit by 0.8 percent, which could impact on the criticality limits like allowable masses, dimensions, and so forth that will be used throughout the facility.

Since those are the closest experiments to the actual SHINE conditions, it was very important that we get to the root of that particular issue. There has been some research. They have gone back to the benchmark evaluators and determined that there are some critical volumes that appear to be inconsistent between the model and the description in the benchmark evaluation report by about three percent and they account for the low K-effective. This is still subject to

ongoing investigation.

CHAIRMAN BLEY: What was the basis that SHINE used for rejecting these outliers, they were just outliers?

MR. TRIPP: Well, I think the fact that -- I mean it is pretty obvious here but the basis was they went back and determined there appeared to be an error in the evaluation. But the benchmark evaluators have -- it is going to take a fair amount of time to go through and research that and find out and confirm that there is an actual error. There appeared to be an inconsistency in the description.

What SHINE recently indicated in their most RAI response, which was from September 15th was that they believe they have identified the problem and that the benchmark is fine but there was an error made in the model. If they correct the model, they believe it will bring them more in line with the other experiments, although it will still under-predict it.

CHAIRMAN BLEY: When you say in the model, that is in the calculation?

MR. TRIPP: In the calculation, right. The description of the geometry of the calculation.

And they have gone back and reevaluated that and believe that now it will produce a K-effective that is under-predicting by about one percent, but still under-predicting but much more in line with what is normally seen for these sorts of things.

So, until they --

MS. BANERJEE: Can I interject, please? Maitri Banerjee, DFO. This is a new RAI response that you don't have, Steve?

MR. LYNCH: I guess we will provide it to you if it hasn't but it did just come in within the last week or so.

MS. BANERJEE: Thank you.

MR. SMITH: On slide 10, of all those points of data that you have, are all of those uranyl sulfate critical?

MR. TRIPP: No, the only ones that are uranyl sulfate are these four benchmarks and there are a few up around 90 percent enrichment. But the vast majority of them are not. There are very few uranyl sulfate benchmarks out there.

MR. SMITH: So that makes it very important not to throw away the important points.

MEMBER BALLINGER: I was going to say, doesn't that argue in the other direction, that you

really ought to go after that and see if that is a real sort of something unique about uranyl sulfate in these conditions?

MR. TRIPP: Right, that was our concern with them throwing the outliers out. They may be throwing out good information that might be trying to tell us something.

Yes, so what we had originally proposed as a license condition that we have since dropped, based on this latest response, this response also committed that they would increase the margins of criticality to 0.06. As you can see, the potential effect on the upper safety limit, if we increase the subcritical margin, it would be sufficient to bound that.

So, they may revisit that in the future but I believe we have determined that would require an amendment.

MR. SMITH: But is that the proper way to add four points that are the only relevant points? Wouldn't the upper safety limit change a long more than that small amount you are showing there?

MR. TRIPP: Well, they are the most relevant points. They are not the only relevant

points. We have computer codes that can go through and look at the cross section of data, compare that between the model and the experiment and figure out how much of the bias is due to different effects, using the tsunami code, which is developed by Oak Ridge.

So, it shows that all of these experiments have some applicability. But certainly the uranyl sulfate ones at 20 percent are, of course, the closest to SHINE and a big concern if they are all under-predicting. So, they have agreed to bound that by increasing the subcritical margin until this can be investigated further.

MEMBER SCHULTZ: Can we go back to 9 for a moment? The bullet that talks about the preliminary research, is that preliminary research that has been performed by SHINE or by the staff?

MR. TRIPP: Neither. I believe that SHINE has gone back to the Idaho National Lab, the group that maintains this benchmark handbook. And it is a large group of individuals throughout the criticality safety community that are involved in developing that and they have done some preliminary investigation. They have had to go back to the experimenters. The experiments were done in Russia

back in the 1970s. So, there is some difficulty in reconstructing exactly what conditions applied at that time.

MEMBER SCHULTZ: Did that investigation begin as the result of this controversy related to the deletion of the data?

MR. TRIPP: Yes, it did.

MEMBER SCHULTZ: So, what is the time frame? I am trying to understand how fast we are rushing toward a conclusion on this important issue.

MR. TRIPP: Well, they said it would take up to a year to do a thorough investigation of this. The experimenters that were involved, they are dead. And this is a Russian facility and a lot of documentation is difficult to reconstruct.

MEMBER SCHULTZ: I guess I am picking at the words but it says preliminary research shows that they are underestimated by and may account for. We need to be very cautious about how we move forward with this, I think.

MR. LYNCH: What I think this is, SHINE is actively updating the validation report with what they believe are the correct figures and they are working on getting that updated validation

report to us as soon as they can. And that is on a magnitude of months.

MEMBER SCHULTZ: Well, I understand that but they are rapidly moving toward what I see to be a conclusion that is in their favor, if you will, that this underestimation is real and, therefore, they have reevaluated instead of three percent it is one percent. So, that is a good result for them. So, they are moving rapidly to integrate that and yet it will take a year to validate what Christopher just said what the situation really is with the International Benchmark Team.

So, I think we need to understand more about what is happening here before we agree with the conclusions.

MR. LYNCH: Understood. I think SHINE might be able to clarify real quick exactly what they are --

MR. VAN ABEL: Yes, this is Eric Van Abel with SHINE. I just want to clarify the process and where everything is at.

The original thought that we had was that there was an error in the benchmark, the volumes weren't lining up. And we communicated

with the benchmark personnel at Idaho and they agreed with our conclusions. But since then, we determined that the model itself had the error. We don't believe, currently, that there is an error in the benchmark. So, we fixed the model and we have updated our model of their benchmark and we are getting K-effectives that are very close to one indicating they will fall in line with all the other experiments currently. So, we have agreed to this larger margin of subcriticality for now, just because our modeling is not complete.

MEMBER SCHULTZ: Okay. Just to nail down the timing, is what you have just described in your RAI response?

MR. VAN ABEL: What I have said, yes. And the actual completion of the new validation report is not done yet and that is in progress. Current modeling shows it much closer to one.

MEMBER SCHULTZ: Very good, thank you.

MEMBER SKILLMAN: Let me ask this. Is there a backup plan such that if it is recognized that the K-effective is actually higher than predicted, that you have a different TSV, a different vessel, that does not have as much material, so that it is indeed less reactive?

That is really a question for SHINE. Eric or SHINE Team?

MR. VAN ABEL: Could you repeat that a little bit?

MEMBER SKILLMAN: Yes. Is there is a backup plan that is a hardware fix? Almost everything you have done on this facility is taking credit for geometrically safe or passive safe. In this particular case, you have got a TSV that has a certain amount of 20 weight percent uranium-335. If these reactivity estimates turn out to be higher than you anticipated because of code or error or process, safety-related activities process, do you have a TSV, target solution vessel that is smaller, that holds less materials, so that you do not have the reactivity issue to deal with?

MR. VAN ABEL: Our plan is that we are not going to predict the reactivity in the TSV perfectly with these codes. We will adjust the concentration of the uranium slightly, a few grams per liter, to account for these reactivity effects. So, we will adjust the concentration during our startup commissioning to find the right concentration that masters that TSV geometry correctly.

MEMBER SKILLMAN: Okay, thank you.

MEMBER BALLINGER: But that is a process adjustment, not a physical adjustment.

MR. VAN ABEL: Correct.

MEMBER BALLINGER: A physical adjustment is always better than process adjustments.

MR. VAN ABEL: I mean these codes aren't perfect. So, we are not going to know the exact concentrations to the tenth of a gram before we start up the TSV. We are going to have to, in our startup commissioning determine what is the right concentration that corrects for the biases in the codes.

MEMBER SCHULTZ: Eric, I am still trying to sort out the third to the last and second to the last bullets there. They are saying two different things compared to what you described.

Is the preliminary research correct that the benchmark evaluation was underestimated by three percent? Go ahead.

MR. VAN ABEL: The second to the last bullet is correct. We do not know of a current problem with the benchmarks.

MEMBER SCHULTZ: Therefore, you used

the original data in the benchmark and demonstrated that --

MR. VAN ABEL: The K-effectives are better than one percent.

MEMBER SCHULTZ: -- with a revised model that you are now using.

MR. VAN ABEL: Yes, with the corrected model.

MEMBER SCHULTZ: Okay. And if the underestimation is there, then it will provide a better result? You are not including the underestimation in your evaluation at this point?

MR. VAN ABEL: We are not including those cases in the current validation report that proves one because we haven't finished that. When we include the new cases, it might drop the USL slightly but by much less than one percent. So, we feel that one percent is bounded.

MEMBER SCHULTZ: Okay, thank you.

CHAIRMAN BLEY: I would just like to take a minute, before we go on, as a reminder to the committee and everyone else, we will have a closed session later today if you need it. So, if any of us bring up issues that deal with particular parameters of this design, be careful. And perhaps

if you can address this or if you should hold it off until the closed session.

MR. TRIPP: Okay, so the next issue and the next bigger issue we had was there were these expectations in the guidance, the ISG that indicate that the applicant should commit to certain criteria for the use of controlled parameters, mass, geometry, moderation, and so forth.

And some of these criteria specify what measures they should use when they are controlling that parameter. That is further down the road. But a lot of them deal with how they are addressed in the models. For instance, if they are not controlled in reflection, they should assume full water reflection, which is a foot thick of water and so forth; minimum spherical mass, where they are allowing a mass control, that sort of thing.

And those technical practices provide conservative margins that we consider part of the assurance of subcriticality. So, a certain amount of conservatism is assured by that that adds to the margin of subcriticality and provides us confidence that it would be subcritical under normal and credible abnormal conditions.

The PSAR did not contain those

commitments. Again, in the latest RAI response, I believe it was the latest, they had committed to now they are going to follow those commitments and bring those in.

So, and the third issue I want to talk about is that originally the PSAR had said that criticality would be rendered not credible. Now, that exceeds the standard we usually apply for these kinds of fuel cycle facilities. On the Part 70 side, we would require them to show that credible scenarios remain highly unlikely because criticality would be a high consequence event. But here they originally say that criticality would be not credible.

So, we did question that and had some RAIs on that. They have since come back and agreed to say that they will render criticality to be highly unlikely, which is consistent with what we have for a fuel cycle facility integrated safety analysis.

There are some criteria for defining what non-credible is. And this has been a consistent issue on the fuel side because if it is not credible, it is not generally listed in the scenarios in the integrated safety analysis. So,

we are interested in what they screen out and what the technical basis for that is. Because in the assumptions they are making to screen those scenarios out, then they have to assure that there is an assumption that is carried forward but those are generally not considered controls.

So, they did have criteria for what is considered not credible and those include things such as an external event with a frequency of less than ten to the minus six per year, many unlikely human errors or actions and so forth. But they don't allow reliance on any control features because if it is not critical, there is really nothing that you should need to have to control. If you have to control it, then it is credible.

So, they had had those criteria but they did not have their prohibition against using facility engineered or administrative control features as the basis for saying that it wasn't credible.

So, they have now committed to make that highly unlikely and part of that includes there is a preference that they will consider engineered over administrative and passive over active engineered controls. And in using that

control hierarchy, they will, along with a conservative margin that they have now committed to should be sufficient to ensure that they will be able to meet highly unlikely. So, that has been resolved.

All these issues have been resolved, except for the validation issue, which is ongoing.

So, I quickly want to just summarize the proposed permit conditions that we have. And some of these conditions have been updated because we had recent discussion in a few days, both based on the recent RAI responses, as well as internal discussions.

So, the wording here is slightly different than what you have in the version of the SCR that you have. But the first condition relates to this issue about the design of the Criticality Alarm System. And we propose requiring SHINE to provide a technical basis for the design of the CAAS, including a method for determining detector placement. To answer your previous question, yes, we do consider that to be safety related. And that is to be provided prior to installation.

The second condition is that SHINE should provide the basis for determining if they

determine that criticality is not credible of a particular scenario, a particular area, they should provide the basis to us for that, so that we can review it prior to installing the equipment. Again, we want to avoid a situation where they pour concrete and then we find it not to be acceptable.

And as I said before, this has been an issue, this credibility/non-credibility issue has been an issue in fuel facilities, which is the reason for calling that out specifically.

We then have a more general condition that says prior to installing equipment, they should submit summaries of the criticality safety evaluations. Now, we understand they may not have a final signed off criticality safety evaluation. They currently have some preliminary safety evaluations that we have looked at somewhat but we are asking them to provide a summary of the criticality safety analysis prior to installing the equipment that defines what the hazards are; what parameters they are relying on; describes the normal conditions and the upset conditions that they considered as part of the analysis; their approach to meeting the double contingency principle; and lists of any engineered controls and

assumptions built into the facility. Administrative controls are things that can be done after they pour the concrete but engineered controls have to be built in.

So, we are proposing that they be required to do that prior to installing the equipment to give us an opportunity to review it.

We are also proposing a condition that they have a -- they will account for the production of fissile isotopes other than uranium-235, such as plutonium, in the irradiated solution that is then brought to and refurbished from the target solution vessel.

We have done some confirmatory analysis that indicates that there is a slight reactivity effect due to the ingrowth of those fissile isotopes. The condition in that safety evaluation report also mentions deuterium but we have determined that deuterium is not going to be an issue. And none of their current evaluations that we looked at address that. So, we would want that to be addressed as part of the design criteria.

CHAIRMAN BLEY: I have a question for you. So, fuel cycle facilities, you referenced NUREG-1520, I think, for this discussion you had

about credible/incredible, highly unlikely and unlikely. Is that correct or is it somewhere else?

MR. TRIPP: No, that is correct. It is in Chapter 3 of NUREG-1520, Revision 1.

CHAIRMAN BLEY: Okay. And you define those terms very precisely.

MR. TRIPP: Well, they allow licensees to define the likelihood terms.

CHAIRMAN BLEY: That's what it looked like to me when I looked at it so, I am a little confused about your discussion that they didn't meet the criteria for one or the other, since they define it.

MR. TRIPP: Well it has more specific guidance related to what is considered not credible. There is a series of three bullets, basically, in the guidance that pretty much the whole fuel cycle industry has adopted. Unlikely and highly unlikely, that is defined in various ways throughout the industry.

CHAIRMAN BLEY: But you did say the incredible is defined by three bullets in Chapter 3.

MR. TRIPP: Correct.

CHAIRMAN BLEY: Okay, I looked and had

a little trouble finding that but I will look some more.

Go over it again for me because you mentioned ten to the minus six for external events, multiple unlikely human errors, and what was the third one?

MR. TRIPP: The third one is just basically prohibited by physical arguments, physical laws.

CHAIRMAN BLEY: So, really impossible.

MR. TRIPP: Right, but of course the difficulty is a lot of the times those kinds of arguments are based on some mode of the process, the equipment that you have and so forth. So, we want to be very careful that they don't base that off of something that really should be declared as a control.

CHAIRMAN BLEY: But in NMSS, that is what everybody has pretty much accepted they mean by these terms -- that one term, incredible. The other terms, they kind of define themselves.

MR. TRIPP: That is correct, although we have had a lot of issues when it comes to implementing those, which was the reason for proposing this condition.

CHAIRMAN BLEY: Okay, thanks. I was a little confused because I had looked at that and couldn't quite see it as clear to me as you made it sound but I think I get it now.

MR. TRIPP: Okay.

CHAIRMAN BLEY: Thank you. I believe this is a good time for a break. We will break and come back by 10:15, please. We are in recess.

(Whereupon, the above-entitled matter went off the record at 9:58 a.m. and resumed at 10:16 a.m.)

CHAIRMAN BLEY: We're back in session.

MR. HENNESSY: Next, we will proceed with Chapter 11, Radiation Protection and Waste Management. we have Mike Launi from Sargent & Lundy supporting this and Ernest Wright also from Sargent & Lundy.

MR. LAUNI: I'm Mike Launi. I'm the nuclear technologist and regulation manager at Sargent --

CHAIRMAN BLEY: Could you pull the mike a little further or speak a little louder?

MR. LAUNI: I'm Mike Launi. I'm the nuclear technologies and regulation manager at Sargent & Lundy. I will start off with radiation

protection.

The radiation source terms and the parameters that are used to evaluate them are provided in Tables 11.1-1, which is the parameters, and then the source terms are provided in Tables 11.1-2 and 11.1-3. Next slide.

The SHINE facility have airborne, liquid and solid radioactive sources. The airborne sources are gases produced from the Mo-99 production.

Liquid sources are present at a number of locations in the facility. There are no radioactive liquid discharges from the facility normally. And there are some solid sources that exist in several locations in the facility.

The activities are designed such that the estimated annual doses to the maximally exposed individual at the site boundary or at the nearest resident are below the dose requirements of 10 CFR 20.1101(d). They were calculated initially using an ICRP 30 for the PSAR. And SHINE will incorporate age-dependence per ICRP 72 for the FSAR.

The maximally exposed individual at the site boundary was determined to by 9 millirems per

year and at the nearest resident, which is about a third of a mile away, as 0.6 millirems per year.

I will now go into the radiation protection program for SHINE will meet the requirements of 10 CFR 20, Subpart B and use the guidance contained in Regulatory Guide 8.2, which is the administrative practices and radiation surveys in Chapter 3.

The objectives of the program will be to prevent acute radiation injuries and to limit the potential risk of probabilistic effects, the stochastic effects, to acceptable levels.

SHINE has developed some preliminary administrative limits, which are shown in this table, which are on-tenth the 10 CFR 20 dose limits.

The radiation protection program organization is shown here. The staff is consistent with the guidance provided in Regulatory Guides 8.2 and 8.10. A Radiation Safety Committee is established and meets periodically at least annually to review a status of products, measure performance, and look for trends and to review radiation safety aspects of the facility operations in accordance with 10 CFR 20.1101(c).

The plant manager is responsible for the operation of the facility, including the protection of personnel from radiation exposure resulting from facility operations and materials and for compliance with applicable NRC regulations and the facility license.

The environment, safety and health manager reports to the CHOOF and has the responsibility for directing the activities that ensure the facility maintains compliance with appropriate rules, regulations and codes. This includes ES&H activities associated with the nuclear safety, radiation protection, chemical safety, environmental protection, and industrial safety in establishing and maintaining a radiological environmental monitoring program.

And then the radiation protection supervisor reports to the environment, safety and health manager.

The operations manager is responsible for operating the facility safely and in accordance with the procedures so that effluence released to the environment and exposure to the public and on-site personnel meet the limits specified in the applicable regulations, procedures, and guidance

documents.

MEMBER SKILLMAN: Mike, before proceeding, may I ask you please to go back to slide 17?

MR. LYNCH: Sure.

MEMBER SKILLMAN: The bullets that you show here, particularly at the bottom of the page are the anticipated exposures.

MR. LYNCH: Yes.

MEMBER SKILLMAN: Are calculations from which those numbers came the product of Sargent & Lundy or the product of the SHINE Radiological Protection Team?

MR. LYNCH: These were done, these calculations, and correct me if I am wrong, were done by -- we had done the initial calculations and then another contractor did the follow-up revision to the calculation.

MEMBER SKILLMAN: So who owns the calc?

MR. HENNESSY: Excuse me. SHINE owns the calculations.

MEMBER SKILLMAN: Do you own the results?

MR. HENNESSY: Yes.

MEMBER SKILLMAN: Are the processes

that brought you these calcs processes that are safety-related activities?

MR. HENNESSY: Yes, the calculations were performed in accordance with quality procedures, safety related quality procedures by the contractors who did them.

MEMBER SKILLMAN: Thank you.

CHAIRMAN BLEY: Excuse me, before you go ahead Mike. Folks on the bridge line, please mute your phones using star, 6. You are making a lot of noise for us here.

MEMBER SCHULTZ: One more question in background. On 19, you have listed the requirements for the protection program as preliminary limits. Can you describe why you have chosen to call them preliminary administrative limits?

MR. LYNCH: I think at the moment because that is what they are thinking they are going to have those limits set at but that could potentially change when we get to final design.

CHAIRMAN BLEY: That is something that is set before the OL.

MR. HENNESSY: Yes, the admin limits will be determined before the OL.

MEMBER SCHULTZ: But how are these being treated, I guess is my question? Are they being treated as expectations or goals or what? We are going into a number of different areas. The next session is on ALARA. What are the principles that you are working to abide by?

You have chosen these on the basis of the 10 CFR 20 limits and divided by ten. It seems like a good practice to establish something. I just want to know what, in your mind, these are. Is it something you would like to achieve, you are desiring to achieve?

MR. HENNESSY: The administrative limits would be limits that would be objectives for maximum dose for individuals working at the plant and if someone were to approach a limit, we would have to look at ways of restricting their work to make sure that didn't happen. If there was no other way to do that, if it was unavoidable, we would have to have processes in place for approval. They aren't something that we would just exceed without consideration.

MEMBER SCHULTZ: Okay, thank you.

MEMBER BALLINGER: So what you are saying is, is that is limited -- I just heard what

you said. There is another way to look at this and that is to say this is a limit and we are going to strive never to exceed it. In other words, adjust your procedures and things like that to drive the dose down, as opposed to having this limit here and if somebody approaches it, do something.

MR. HENNESSY: That's correct.

MR. LYNCH: Okay, we are on slide 21.

The radiation protection program will the written radiation protection procedures. The procedures are prepared, reviewed and approved. Work in any radiologically controlled areas is performed in accordance with the radiation work permits and the procedure we use is the guidance contained Regulatory Guide 8.10.

There is commitment to the radiation protection training. The training will use the guidance contained in Regulatory Guides 8.10, 8.13, and 8.29, and ASTM E1168-95.

Personnel entering restricted or controlled areas are trained or are provided escorts who are trained in radiation protection. And they will be retraining for that at least annually.

Moving on to the ALARA program. SHINE

IS committed to an operating philosophy that maintains occupational exposures as low as reasonably achievable, such as installing temporary and permanent shielding for radioactive material; use of time and distance to minimize exposure.

In addition, design considerations take into account ALARA such that the radioactive material, to the greatest extent practical is remote handled and isolated from on-site personnel by shielded compartments and hot cells; for reliability and maintainability, thereby reducing maintenance requirements on radioactive components; to reduce radiation fields and control streaming, thereby reducing radiation exposure to individuals during operation, maintenance, and inspection activities, and to reduce access, repair, and removal times, thereby reducing the time spent in radiation fields during operation, maintenance, and inspections.

Additional ALARA considerations during the design are provided in PSAR Section 11.1.3.2. It is about four pages of things that will be filed as we are going through final detail design.

Moving on to radiation monitoring and surveying. Personnel entering radiologically

restricted areas will wear personnel monitoring devices. There will be continuous air monitors in the facility; continuous tritium detectors; stack release monitoring that will be continuous for noble gases, aerosols, iodine, and there will be tritium effluent monitoring also. There will be radiation area monitors; obviously, control point monitoring for exiting radiological controlled areas. We will have portal monitors, friskers, hand and foot monitors, and small article monitors; and criticality monitoring, the CAAS system.

In addition, radiation surveys will be conducted to ascertain the radiation levels and concentrations of radioactive materials and potential radiological hazards and to detect the release of radioactive material from facility equipment during operations.

These will comply with 10 CFR 20, Monitoring and Surveying, including subparts F on surveys and monitoring, subpart C on occupational dose limits, subpart L on the records, and subpart M on the reports.

And we use the guidance contained in Regulatory Guides 8.2, 8.7, 8.9, 8.24, 8.34, and the standard ANSI N323-1978 on radiation

protection, instrumentation, testing and calibration.

In addition, there will be radiation exposure control and dosimetry. SHINE will use the following radiation area designations as defined in 10 CFR 20, including consideration of neutron and gamma dose rates.

Unrestricted areas means an area to which access is either limited or controlled by SHINE. This would be the area beyond the site boundary. Radiation areas are both accessible areas where the dose rates is greater than five millirem per hour, 5 mrem per hour and 30 centimeters from the radiation source or from a surface that the radiation penetrates. High radiation areas are those accessible areas where the dose rate is more than 100 mrem per hour at 30 centimeters from the radiation source or from a surface that the radiation penetrates. Very high radiation areas are those accessible areas where the dose rate is greater than 500 rads per hour at one meter from the radiation source. An airborne radioactivity area includes a room, enclosure, or area in which the airborne radioactive materials composed wholly or partly of licensed material

existing in concentrations 1) in excess of the provided air concentrations specified in Appendix B in CFR 20.1001 to 20.2401; or 2) to the extent -- to such a degree that if you are present in the area without respiratory protection equipment could exceed, during hours an individual is present in a week, an intake of 0.6 percent of the annual limits on intake or 12 DAC-hours an individual is present -- excuse me -- 12 DAC-hours. Excuse me.

The contaminated area is an area which SHINE defines as an area where removal contamination levels are above 0.33 becquerel per 100 square centimeters of alpha activity or 16.7 becquerels per 100 square meters of beta-gamma activity.

An area which also could result in an individual receiving a dose equivalent in excess of 0.5 millisieverts in one hour at 37 meters from the radiation source or from a surface that penetrates or is designated as a radiation area as defined by 10 CFR 20.1003.

MEMBER SKILLMAN: Mike, let me ask this question. Are there any areas that you would anticipated would be posted lethal?

MR. LAUNI: I would think -- the only

thing I can think of that would be high enough would be inside the radiation units.

MEMBER SKILLMAN: So, I see that you identified the several postings.

MR. LAUNI: We would have to look at the final.

MEMBER SKILLMAN: And you didn't post legal. So, I am just curious how come.

MR. LAUNI: That is a potential but right now I would think that the irradiation units would be very high radiation areas we posted that but we will see in the final calculations with that.

MEMBER SKILLMAN: So you are saying there could be. If we find it, we will post it that way but right now we don't anticipate that there will be.

MR. LAUNI: Not at this moment.

MEMBER SKILLMAN: Got it. Thank you.

MR. LAUNI: Okay, moving on to contamination control equipment and facility layout.

In general, the equipment and facility layout design considerations are to prevent the spread of contamination to the facility and the

environment and to facilitate eventual decommissioning in accordance with 10 CFR 20.1406.

Process equipment containing irradiated materials located within shielded compartments or hot cells.

Access and egress to the restricted areas is strictly controlled via administrative procedures, as we mentioned earlier, radiation work permits, and passive confinement structure design.

The use of embedded pipes is minimized and shielded pipe trenches are provided for liquid and airborne confinement and leakage detection.

Next is environmental monitoring. Radiological environment monitoring is in accordance with 10 CFR 20.1302 and we also considered guidance for Regulatory Guide 4.1 and NUREG-1301.

The effluent monitoring program includes 24 locations for direct radiation exposure; there is five continuous air samplers for airborne exposure; there is groundwater test site wells; and ingestion exposure from milk for at least the first five years of operation, depending on the results, whether it continues beyond that. And in addition, there will be preoperational

baseline monitoring beginning one year before operation.

In addition, there is a respiratory protection program. Process and engineering controls incorporated into the design of heating -- go into the design of heating in the HVAC systems, as the primary means of controlling the concentration of radioactive material in the air. Respirators may also be used to maintain doses ALARA. The respiratory protection program meets the 10 CFR 20, Subpart H. And fume hood and glovebox operation and maintenance involving uranium-235 processing uses the guidance contained in Regulatory Guide 8.24.

That is the extent of the radiation protection program presentation. And I will turn it over to my colleague Ernie Wright.

MR. WRIGHT: Thank you, Mike. Well good morning. My name is Ernie Wright from Sargent & Lundy Engineers. This morning I shall discuss solid, liquid, and gaseous rad waste expected to be produced at the proposed SHINE facility. I shall address the generation, collection, processing, solidification, storage, and disposal of the various waste streams.

But before we review the waste streams, allow me to note that SHINE will comply with federal regulations related to radioactive waste. These include Standards for Protection Against Radiation, Licensing Requirements for Land Disposal of Radioactive Waste, Packaging and Transportation of Radioactive Material, Radiation Protection Programs, Solid Wastes, and Hazardous Materials Regulations.

Okay, this slide presents the overall facility process summary, showing major sub-processes in block diagram form. This process flow has been previously discussed during the Chapter 9 PSAR presentation last month. I repeat it here because the radwaste streams will emanate from the sub-process blocks shown.

For the SHINE facility, all the wastes are classified as low level waste. Based on preliminary design and conservative assumptions, waste streams in classifications A, B, C and greater than Class C are expected. The waste streams will be in the form of liquids, solids, and gases. Examples of each are as follows.

For solids, neutron generators and their components, extraction columns, resins, Off-

gas Systems, zeolite beds, miscellaneous equipment, glassware and trash.

For aqueous liquids that will be solidified via an in-drum system using a Portland cement-based scrub. These include extraction and purification wastes, URE3X raffinate, spent caustic scrubber solution, and decontamination wastes.

Regarding organic liquids that will not be solidified and processed offsite, there is a very small quantity of tributyl phosphate that came from the UREX process.

Regarding gaseous wastes, we have off-gas from the Target Solution Vessel, which is held for decay, 40 day minimum, along with off-gases from the processing vessels and tanks. Those two combine into a stream that is processed through a caustic scrubber and then passed through charcoal and HEPA filters and released from the facility vent stack.

MEMBER SKILLMAN: Ernie, before you proceed, on that slide number 32, where are you accounting for protective clothing and hoods, hoodies, shoe covers, tape, gloves, that sort of thing?

MR. WRIGHT: Under solids as --

MEMBER SKILLMAN: Miscellaneous equipment?

MR. WRIGHT: Miscellaneous equipment and trash.

MEMBER SKILLMAN: Thank you.

MR. WRIGHT: It is traditionally DAW, dry active waste. Okay, next slide, please.

Regarding radioactive waste controls, the key features of the pollution prevention and waste minimization program include incorporation of design features that will minimize radioactive waste -- that is very key; employee training and education on general environmental activities, as well as waste minimization requirements, goals and accomplishments with emphasis on waste minimization; individual responsibilities for pollution prevention and waste minimization; and requirements for employees to consider pollution prevention and waste minimization in their day-to-day activities and in engineering activities.

Regarding release of radioactive materials from the facility, radioactive wastes are processed and packaged as required to meet the waste acceptance criteria of licensed disposal facilities. The SHINE facility will not discharge

any material from the RCA to the sanitary source system. Gaseous wastes are treated on-site prior to release, completely analogous to the processes used in pressurized water reactors, i.e., gaseous waste is collected, compressed, and stored in decay tanks until they meet release criteria.

The following two slides present a table of waste stream summary from the PSAR Chapter 11. These two slides are based on our preliminary design and conservative assumptions.

This slide presents primarily a solid waste. I would like to draw your attention to the first line, which has a large volume. Primarily the volume is due to neutron generator components.

I would also like to draw your attention to the last two lines that address greater than Class C waste. Note that the volumes are very small and in final design, we may be able to eliminate greater than Class C waste.

The next slide enumerates all the liquid waste streams that will be cement-solidified in-drum.

And then from there, finally containers of rad waste will be placed in a separate storage area to allow decay and await shipment. The

storage area is within the security fence but separate from the irradiation and processing cells.

And that concludes my presentation on radioactive waste.

MEMBER SCHULTZ: Ernie, the gaseous wastes that are treated and then released offsite, they go to the site boundary dose, is that the major component?

MR. WRIGHT: Yes, that is the major component. Yes, it is.

MEMBER SCHULTZ: And what kind of thinking and process -- what kind of thinking about the process has been done to assure that there is, if you will, an offsite ALARA focus; that the system that is being put in place to reduce the dose offsite is appropriate and suitable? You compare it to PWR and that is nice but in terms of your facility and what is being designed here, how have you convinced yourself that you have done enough?

MR. WRIGHT: Well, there is a couple of reasons. We have continuous stack monitors. And then if you go farther into the plant, they have release criteria for the decay tanks. Forty-day decay is minimum but procedurally, you can adjust

that to increase your decay.

MEMBER SCHULTZ: Okay.

MR. WRIGHT: And would you like to add anything to that, Catherine?

MEMBER SCHULTZ: And then you have equipment that you use to process as well, in terms of the scrubbers and the filters and so forth.

MS. KOLB: Yes, I guess I would just add, I want to reiterate that we will have HEPA filters before the exits. We will have charcoal filters after the processing, after the scrubber solution, as you mentioned.

We are looking at ways to reduce it in the final design. This is our preliminary design.

MEMBER SCHULTZ: Okay, thank you. Approximately how many drums per year do you expect to ship offsite?

MR. WRIGHT: I'll let Catherine handle that.

MS. KOLB: In the PSAR we have it listed as approximately 1,150 drums.

MR. WRIGHT: Fifty-five-gallon drums.

MEMBER SCHULTZ: Okay, thank you.

CHAIRMAN BLEY: Okay, thank you. Anything more?

MR. WRIGHT: Okay, thank you.

CHAIRMAN BLEY: Steve, before you guys go forward, do you have people here, if we go ahead before lunch with the QA program?

MR. LYNCH: I don't see -- oh, yes, we do have people here.

CHAIRMAN BLEY: Okay, good because if we can get some of that done before lunch, that would be good. That will give us more room on the safety analysis.

MR. LYNCH: Okay.

MR. ESSIG: My name is Tom Essig. I'm with Chesapeake Nuclear Services. And I will be presenting this morning, Chapter 11.1 and 11.3 of the Staff's review.

And while my focus is mostly on the radiation processing, the radiation units, the collection thereof, I have with me Greg Chapman from NMSS who was involved in the review on the radioisotope production facility items.

So, -- but the Radiation Protection Program is a site wide program. So I'll present it. And then as needed, if there are questions of Greg, then he's here available for that purpose.

So the Radiation Protection Program

that we have reviewed includes the elements shown on this slide. Basically starting with the nature and magnitude of the radiation sources.

How they're shielded and controlled via the ventilation system, the ALARA considerations that were included, radiation monitoring surveillance and dosimetry, how contamination is controlled, environmental monitoring, and then lastly respiratory protective equipment.

MEMBER SKILLMAN: Tom, let me ask you this question.

MR. ESSIG: Go ahead.

MEMBER SKILLMAN: As I see your title sheets there, you're from Chesapeake Nuclear Services.

MR. ESSIG: Yes.

MEMBER SKILLMAN: So are you contracted by the NRC? Or contracted by SHINE?

MR. ESSIG: Yes. We are contracted by NRC through a company called ISL. We are a subcontractor to ISL, who has the contract with NRC for performing this review.

MEMBER SKILLMAN: Thank you.

MR. ESSIG: Sorry.

MEMBER SKILLMAN: Yes, sir.

MR. ESSIG: I guess I should have applied that at the beginning.

MEMBER SKILLMAN: Thank you.

MR. ESSIG: Okay. The regulatory requirements that are applicable here are of course, sections of Part 50. And embedded in that, although we're not showing the slide here, Part 20 is certainly applicable to our review.

Acceptance criteria. We basically have used NUREG 1537 and the interim staff guidance.

Now, the areas of review. We first -- these are in the order that we did the review of Chapter 11.1. The shielding and ventilation system for the radiation facility and the radioisotope processing facility.

Monitoring dosimetry, occupational public doses, ALARA considerations, contamination controls and effluent environmental monitoring. And then lastly, the uses of respiratory protective equipment.

And next I will describe the review procedures and the technical evaluation that we did. Starting with Section 11.1, which is the radiation protection section.

We did a subsection by subsection

analysis and evaluation of the Section 11.1. So we did 11.1.1 up through 11.1.7. And then we had the basic PSAR, but supplemented by several RAIs that we found necessary to issue to clarify the information that was in the PSAR.

And all of which was aimed at -- to assessing the adequacy of SHINE's radiation protection design features for both the radiation facility and the radioisotope processing facility in support of the issuance of a construction permit.

In a similar manner we reviewed Section 11.3, the respiratory protective -- proposed Respiratory Protective Program, and compared it to the relevant sections of 10 CFR 20.

So, under 11.1.1, radiation sources, the key aspects there where we assessed the shielding around the radiation unit and the RPF. And showed -- it showed that the dose rate of less than 1 millirem per hour, that that value was in the PSAR. And we found it to be reasonable.

And too further understand that, we prepared an RAI that was actually 4a2, that chapter. Because that's where the shielding was actually discussed.

But we considered it to be as part of the radiation source. So I've included on this slide here.

We looked at the -- for airborne activity. We looked at the zoning designations that noted that Zone 1 intended for routine occupancy, could have airborne activity.

SHINE had defined that Zone as being .01 to 1.0 DAC. And we had requested some additional information on that definition.

And while we feel that Zone 1 because it's routinely occupied, meets the Part 20 limits, we are looking to SHINE to provide assurance that indeed having concentrations up to 1.0 DAC on a routine are ALARA. And so, we will be looking forward to having that assurance in the future.

The staff's review of the effluent release source term, we found that it is sufficiently complete. An environmental pathway dose assessment had not been performed.

And specifically the inclusion of the environmental pathway was not fully in the initial evaluation. We issued RAI 11.1-9. We received a response from SHINE.

But, as a result of that response, we

feel that a further evaluation was needed. Although the response does show that the parameters that are in Part 20 for ALARA releases, 10 millirem per year, are met.

We have some residual questions about the manner in which those parameters were defined. And so that will be the subject of a future review as the design becomes more final.

MEMBER SKILLMAN: Tom, on that slide, occupied area up to 1.0, derived air concentration, --

MR. ESSIG: Yes.

MEMBER SKILLMAN: Is that because the occupied areas are co-located to higher concentrations and cannot be made to be independent? Or is that because of duct work leakage?

What drives that DAC?

MR. ESSIG: I believe it's just the design of the facility. And the -- Zone 1 is certainly separate from Zone 2 and 3. But the design criteria that they used for Zone 1 was that they would permit up to 1.0 DAC.

Which is certainly, it's consistent with Part 20. You know, it does meet the Appendix

B values. But, what we were challenging is, is it ALARA?

I mean, have you done all that you could to reduce concentrations below that value? And that will be coming in the future.

Now, what could be used? Well, possibly if that ALARA could be demonstrated by -- administratively by limiting occupancy to the area.

If the design itself doesn't achieve -- if I can't be shown that the design itself is ALARA, then the other ways that you can show ALARA are using administrative means such as access control and that type of thing.

MEMBER SCHULTZ: Are these bounds --

MEMBER SKILLMAN: Thank you.

MEMBER SCHULTZ: Are those bounds from .01 to 1.0 DAC, are those provided just to present a range? Given that 1.0 DAC is acceptable under Part 20.

Or are there calculations that demonstrate that they're going to achieve, that they're going to have those concentrations?

MR. ESSIG: Yes. The PSAR does include --

MEMBER SCHULTZ: Calculations that show

those concentrations will be present?

MR. ESSIG: Well, it's the -- basically the design criteria for Zone 1, --

MEMBER SCHULTZ: Well, design criteria and results are different.

MR. ESSIG: Yes.

MEMBER SCHULTZ: I'm trying to understand the criteria. But I'm trying to understand whether they're -- the same way you are, I believe, trying to determine what is the overall approach to radiological controls in those areas.

MR. ESSIG: Well, as I understand it, the source term that the -- the radionuclide production source term is of course contained in the radiation facility. And there's going to be a certain amount of leakage that does occur via the ventilation system.

MEMBER SCHULTZ? That could occur?

MR. ESSIG: That could occur.

MEMBER SCHULTZ: Okay.

MR. ESSIG: I'm sorry, yes, correct. Could occur. And it -- when the system is operating properly, then based on that leakage of the radionuclide source term into the occupied area, then it would have those concentrations.

But I -- the detail regarding the individual radionuclide inventory and how that's -- how that leakage manifests itself or could manifest itself into an unoccupied area, the details are not yet developed.

MEMBER SCHULTZ: It's worth pursuing. Thank you.

MR. ESSIG: Yes. Okay, so there were supplemental issues after finalization of the design that we felt we needed to see.

Just to kind of summarize these, we touched on them already. But the source terms for final shielding design, for liquid waste and the molybdenum extraction and purification system, the actual design of the shielding is -- the finalization of it has been put off beyond the construction permit issuance date.

And -- or time frame. And the -- again, what we were just discussing the confirmation of Zone 1 up to 1.0 DAC air concentrations, the confirmation that those are ALARA.

And then the environmental exposure pathway dose assessment in compliance with Part 20 public dose limits and particularly the ALARA

provisions, the 10 millirem per year for air activity release that will be confirmed at a later date.

With regard to the Section 11.1.2, we particularly looked at the proposed facility organization and the lines of authority for the RPM to make sure that the radiation protection manager will have the necessary independence. And will be able to raise issues separately from the plant operations component.

You saw the organization chart on a previous presentation. We feel that that was an appropriate separation. And that the radiation protection manager will have access to facility executive management.

The training to provide -- be provided a staff and visitors, we determined that it would meet the requirements of 10 CFR 19. And consistent with the Reg Guides listed here.

The Radiation Safety Committee responsibilities, we review the charter of the Committee as proposed. And based on that charter, it should -- we believe it should be able to provide a relevant oversight role.

The use of radiation work permits, we

found that there was a commitment to use them. And that the proposed nature of the RWP as described in the PSAR should be effective in the management of radiation exposures by the SHINE staff.

And we had no RAIs for this subsection of Chapter 11.

The ALARA Program, 11.1.3, the overall considerations we noted that SHINE had a proposed update and modify traffic control security and access control and HB procedures as design layout experience is gained. As an ALARA measure that would be an administrative processes that would be applied.

Program design considerations, we determined that the access to equipment requiring maintenance was provided. Equipment with high level exposure rates is compartmentalized. And adequate provisions exist for storage and use of mobile shielding.

And the PSAR was missing in management commitment to develop and implement an ALARA Program. And we requested that in an RAI. And SHINE has subsequently provided that commitment.

The purpose there was to determine at

the highest levels of facility management, whether or not they were committed to maintain exposures ALARA. And they have done so.

MEMBER SKILLMAN: Tom, could that -- did that ring a bell in the staff's mind? I mean, SHINE is all about irradiating 20 percent uranium 235.

MR. ESSIG: Yes.

MEMBER SKILLMAN: And it would seem that there would have been a fundamental orientation instantly out of the blocks for the SHINE team to have an ALARA Program. So, what happened?

MR. ESSIG: I don't know that I can answer what happened. I do know when we looked for this, I mean, we saw the -- and reviewed the ALARA Program description.

But what seemed to have been missing from it was a commitment by the top management of the facility to have an ALARA Program. Even though it's required by regulations.

There is a -- we always look for a commitment from the highest levels of management to make sure that indeed we as the high level management support having an ALARA Program.

MEMBER SKILLMAN: Let me ask one more.

MR. ESSIG: Okay.

MR. COSTEDIO: This is Jim Costedio from SHINE. We view the PSAR as a commitment. So, we put all the requirements for ALARA in the PSAR. But we did not call it a commitment specifically in the PSAR.

So at the RAI, we just added the word. But all the requirements are there. We think it's very important. That's what we think.

CHAIRMAN BLEY: Okay, thanks. Go ahead and speak on that one.

MEMBER SKILLMAN: I'll hold. Thank you.

MR. ESSIG: Okay. Under radiation monitoring and surveys, we identified, or the PSAR identified several types of sampling and monitoring equipment located within the radiological controlled areas at the exits and at the plant stack.

Continuous air monitors will be used in controlled and restricted areas. However, we did not have the locations specified in the PSAR.

But there seemed to be a large number of continuous air monitors. And so we're willing

to let that be settled at a future date.

Believing that words generally there that were in the PSAR would have them in the proper locations. Just that we didn't have a specific drawing showing the location of a continuous air monitor or where they were.

Control point monitoring, portal monitors, friskers, hand and shoe monitors, tool monitors at the exit, the RCA, that seemed fairly straightforward.

Written surveillance program procedures are to be developed. And we found the level of specificity acceptable at the PSAR stage. And that we did not need any RAIs for this particular aspect of the program.

Under radiation exposure control and dosimetry, we looked at both external and internal dosimetry. We found that they're going to be wearing beta-gamma dosimeters. They're changed quarterly.

And exposures greater than 25 percent of the administrative limits will be -- which we found earlier were 10 percent of the Part 20 limits.

So, they'll be investigating, or they

proposed investigating any time an exposure greater than 25 percent of that 10 percent, will be investigated and reported to the radiation protection manager.

The internal dosimetry states that SHINE will use a combination of in vivo measurement. Typically whole body counting. And excreta measurements and air concentrations which are completely in line with the approach that is required by Part 20.

Support facilities, there's a radiologically controlled area entry and exit. Personnel decontamination, protective clothing put in storage provided.

We had a couple of RAIs here. One of them regarding the radiation area designations. And whether very high radiation areas will be included in the facility design.

Because in going through the design, we did not see mention of very high radiation area. What you saw on the previous slide are greater than 500 rad per hour to meter.

And these require a very high level of control. And the reason we're asking it at this stage is because if there is something that needs

to be included in the plant design for a very high radiation area, maybe a shield plug or something of that sort that we wanted to raise it fairly early on.

Whereas the other radiation area controls, the high radiation area and radiation area don't require such controls as are required by very high radiation area.

They don't have to have a very high radiation area. But, in the event that they found it necessary, it would be prudent to address that at a fairly early stage.

MEMBER SKILLMAN: And I'd like to go on record, this is why I asked the question about the lethal posting. It's the exact same issue.

Is there an area that is of such high level that it needs that very special attention that you would have for a VHRA? Lethal would be the same way.

MR. ESSIG: The NRC posting requirements did not address an area that would be considered lethal. But however, the very high radiation area described as posing grave danger.

And those are the words that are used in the regulation.

MEMBER SKILLMAN: Right. Yes. Yes.

MR. ESSIG: And so I think that because there are very strict controls over access to very high radiation areas that the manner in which they would be designed and administered, if when done in accordance with the regulations, should prevent the situation like you were describing according to lethal.

MEMBER SKILLMAN: I meant grave. And I thank you for the correction. Very good. Thanks.

MR. ESSIG: Okay. And design considerations for decommissioning. Basically the question here is, have draining, flushing and decontaminating equipment minimizing the buildup of radiation material in equipment with an eye toward decommissioning.

And meeting the requirements of 20.1406. And we didn't find any RAIs necessary here. And we felt that sufficient design considerations had been included.

Environmental monitoring, the PSAR identified direct exposure monitoring plus sampling of air, groundwater and foodstuffs. Here we found necessary to issue RAIs to clarify.

The first one was a simple matter of

the PSAR referenced air monitors. In the definition of air monitors commonly used, it is an instrument that not only collects the sample, but provides a real time readout at the location.

I didn't feel that's what they had intended here. And they clarified that no, what we really meant was air sampler for the environmental location.

Which is typically the situation. It involves an air sampler and not a real time monitor.

The other RAI was the -- they had not indicated that they would regularly sample milk from nearby dairies. Only when certain effluent trigger points were exceeded.

We felt that because they had cow and goat dairies in the vicinity that it was necessary to request a commitment from them to monitor that pathway. And they did so.

And then lastly, the third RAI was the -- we requested them to reassess their position on a number of direct exposure monitoring locations using TLD. They did increase the number of locations to the satisfaction of staff.

And then briefly on the Respiratory

Protection Program. We had one small issue there. It was clarification where they -- SHINE had said that their fit factor for respiratory protective equipment would be at least 500 times the assigned protection factor for the device.

And where were had devices where the fit factor was 1,000, so if you're looking at a protection factor of just doing the math here, 500 times 1,000, you'd have a protection factor of 500,000. Which we feel would be beyond the capability of most quantitative fit testing methods.

So we would just ask that that be clarified in the FSAR. Which is the case.

We had no RAIs for this section of the program. We found generally consistent with regulatory requirements.

The findings, we found that the radiation exposure limits based on a shield and ventilation system as supplemented by the final design information, supercell and liquid waste storage tank shielding, administrative controls such as posting access controls, and the Zone 1 ventilation system controls are ALARA, that we found that that would be acceptable.

And that commitments made for the organization in the manner that they intend to be operated regarding facility staff, environment and public are protecting them from unacceptable exposure radiation was appropriate.

The basis for ALARA procedures was found to give us reasonable assurance. That doses of occupational workers were publically maintained at ALARA levels.

And that the general effects and of monitoring and surveillance meet the regulatory guides. And would be consistent with Part 20.

And the program for posting and access control of restricted areas, controlled areas, and so forth, would be consistent with Part 20.

And that the description and level of pertaining to the plant design features intended to contain leakage for an eye toward decommissioning the facility, we found that those to be sufficient at the construction permit stage.

And the respiratory protective equipment is generally consistent with the regulatory requirements.

And that concludes my presentation on Level 1 and Level 3.

CHAIRMAN BLEY: I just have one little question.

MR. ESSIG: Yes?

CHAIRMAN BLEY: There are a number of places where you indicted that more clarification needs to come in the future with respect to their responses to some of the RAIs.

MR. ESSIG: Yes.

CHAIRMAN BLEY: Is it the staff position that the responses are adequate for the construction permit? Or do they need those clarifications before the construction permit is completed?

MR. ESSIG: Our positions are at or before the construction permits --

CHAIRMAN BLEY: They are adequate? So there are no --

MR. ESSIG: There are no conditions in the construction permit that they would have to provide the information soon after the construction permit is issued. I don't have a time frame on that.

CHAIRMAN BLEY: Okay.

MR. LYNCH: Yes, and how we're tracking that too. So in the cases of RAIs that were asked,

when we got the information we needed, oftentimes SHINE would make a commitment that they, you know, that the more information would be provided.

And what we've done and what will be concluded is that included as an appendix to the FSAR, it's always to regulatory commitments that SHINE has made that we will verify that they have done for the FSAR.

CHAIRMAN BLEY: Thank you, Steve.

MR. LYNCH: We still have one more presentation for Chapter 11 to finish up on radioactive waste management.

CHAIRMAN BLEY: You need to pop this up to fill the screen there.

MR. LYNCH: Yes. I don't have the right one.

CHAIRMAN BLEY: It's the little thing in the corner, right?

MR. McILVAINE: Madam, gentlemen, my name is Jim McIlvaine. I'm with Chesapeake Nuclear Services also. That's subcontracted through ISO to perform the review of 11.1, the radioactive waste management systems.

Again, the same regulatory requirements exist. And radioactive waste management for the

SHINE facility is -- it's significantly different then the power reactors. Because they're generating a lot of waste over the entire spectrum of waste streams.

They intend to store much of it onsite for as long as it takes to decay to a level where they can ship it as Class A waste.

So, that would bring up a number of attendant radiation protection issues both within the facility and in the design of the waste staging and shipping building. Which is mention in the PSAR. But that design is not presented at all. It will be something available and will have to be reviewed at the FSAR stage.

Our staff review was as rigorous as could be performed with the amount of detail we had on the program. Again, I should have mentioned earlier, maybe it's in the next slide, 11.2 describes the radiation -- radioactive waste management program, the radiation controls and release of radioactives -- of radioactive material.

The actual systems that handle the radioactive waste are described in Chapter 9b5 and 9b7. So you have already seen what there is of those.

The key issues were with the way radioactive waste management program is that do they have or do they have the staff and the processes to assure that they can handle radioactive waste for a facility that currently doesn't exist in the U.S.

They do have defined program objectives, management and supervisory responsibilities, program elements such as the self-assessments, audits, training, record keeping and document control that presents a sufficient administrative structure that they will need to -- can meet the requirements and can continue to improve their processes and procedures as they figure out ways to minimize the waste generated.

Elements of the program that will need to be reviewed more closely at the FSAR stage, which were not presented in the PSAR are the development of the waste management procedures, the waste management charter and how these are all integrated into their overall conduct of operations. Which was identified in a different Chapter, but really not discussed.

Radioactive waste controls are presented in several tables at the end of 11.2

where they take each of the waste streams in particular, and define those things that will be done, or that must be done in order to control that waste.

To adequately characterize it both quantitatively and qualitatively. And then package and -- for a subsequent packaging for storage, transportation, and ultimately for disposal.

The control -- the tables identify sampling. They don't provide any details of how some of that sampling is going to be done.

Going back into the drawings that are provided for systems in Chapter 9, they do show again, the sampling points. So that the waste streams can be adequately characterized and quantified.

That is something that will have to be looked at in some detail at the FSAR stage to assure that that can be performed in an ALARA manner.

And again, the liquid waste chemical characteristics and the radioactive material content of the waste is significantly different then what we are used to in power reactors.

We are talking about UREX raffinate,

which is a nitric acid with a PH less than one in the table 90 something of the PSAR. So, that led to an RAI to SHINE regarding cement solidification.

They had made a -- they refer to EPRI NP-2900 as a basis for being able to accomplish that. The NP-2900 report focused primary on boric acid waste.

It does suggest that you can certainly solidify acidic waste. There will just have to be some modifications made to the waste in order to accomplish that.

They have agreed to a PC -- or they did commit to a pre-commissioning test of the solidification system to assure that the PCP can provide a homogenous solid for these waste streams.

And that is adequate at the PSAR stage. We will continue to review radioactive waste management operating procedures -- we will review the radioactive waste management procedures at the FSAR stage.

And the storage and handling in the storage and shipping building will be reviewed in detail at the FSAR stage also.

MEMBER POWERS: There have been some notable failures in concrete isolation of low level

waste. Particularly at Rocky Flats.

Have you looked at those to see that if we understand why they failed? And that we are not going to commit similar --

MR. McILVAINE: I found a -- I did find a NUREG CR-71.71, dated 2013 that deals with cement degradation in nuclear power plants.

Now, this focuses primarily on structural concrete in MPPs that is like reactor pressure vessel shielding. And it looks at the radiation damage to the material.

It has caused, at least according to the authors of that report, it is primarily both a thermal and a destruction of covalent bonds. My concern was, do we have enough thermal energy in the raffinate to cause a problem?

Or the high level of short term -- short lived fission products in the raffinate, is that a significant issue? A colleague of mine did some preliminary calculations using the micro shield program.

And again, these are, you know, my assumptions, not a SHINE calculation. But it looks like the total decay heat in any one particular 55 gallon drum is going to be watts or tens of watts.

Not anything that's going to pressurize a drum. And although the radiation levels are going to be large, they're not going to be 10 to the fifth grade that starts causing effective decay or degradation of concrete.

MEMBER POWERS: No, that was not the problem at --

MR. McILVAINE: At Rocky Flats.

MEMBER POWERS: Rocky Flats.

MR. McILVAINE: No.

MEMBER POWERS: At Rocky Flats the concrete just failed to set up. And they literally had the stuff falling apart on them.

It might be useful to look at that, the findings on that just to understand that what the limits are on using concrete to isolate some of these strange waste streams that come from reprocessing.

MR. McILVAINE: It's -- again, they have committed to the preliminary tests. There is a -- again, there's not --

MEMBER POWERS: Basically Rock Flats did all kinds of preliminary tests. And what happened was they had produced, what? Several tons of concrete cylinders.

And then found them falling apart on them. Just literally disintegrated. And it was not at all because of thermal load. There's almost no thermal load on these materials.

But the concrete was just failing to set up.

MR. McILVAINE: And again, it is a chemical issue. And it is something that SHINE will have to, I suggest, spend some time making sure that they have a process control program that can produce a homogenous mass that doesn't fall apart.

MEMBER POWERS: I suspect the real problem is going to be sulfate contamination. Because you combust the concrete.

MR. McILVAINE: Yes.

MEMBER POWERS: It's -- concrete, the setting up of concrete, I mean, I -- is a -- it's not -- it's the easiest thing in the world except when it doesn't happen.

MR. McILVAINE: Precisely. And it's a -- it is a complex chemical process. And chemistry nitric acid, sulfuric acid, a number of the different waste streams that go into the liquid waste storage tanks may have an impact.

Again, they have committed to pre-commissioning testing. And that should provide adequate assurance that they have found a solidification process that will work.

CHAIRMAN BLEY: The testing will I guess need to be with some kind of surrogates, right?

MR. McILVAINE: It will be done with surrogates. Not with the radioactive material is what they're committed to.

CHAIRMAN BLEY: How are we going to consider whether those surrogates are adequate? And I wasn't aware of this problem at Rocky Flats.

And I don't know if people around the world have had it on other facilities of this sort. But --

MEMBER POWERS: Well, I think in the end, the problem at Rocky Flats was they went to too lean of a concrete. The cement mixture.

But, I don't know that for a fact. But it is, I mean, we tend to think of oh, well, they just put it in concrete and everything's fine.

Well, here's a case where it was not fine. It doesn't seem to be even widely recognized. This is a problem.

And like I say, you can rest assured that Rocky Flats did all kinds of tests beforehand. But, if your guy's mixing the concrete actually put in too little cement, that's a problem that's hard to catch in testing.

MR. McILVAINE: And SHINE has identified that they intend to adjust the PH prior to evaporation. And then add additives to the poured in cement and I believe slag.

And they have not finalized certainly any kind of a ratio, but their 1,500 drums a year was based on a -- I think a .5 or .7 ratio of waste to cement.

So, they are certainly aware of the potential problems. And have proposed a, hopefully conservative solution.

MEMBER POWERS: You know, they may run into problems if they're adjusting PH with things like psyllium or potassium hydroxide, either one of them. You get those ratios a little too high and the concrete doesn't like it.

MR. LYNCH: We understand your concerns. And I think this is something we'll make sure we take a look at with the final designs.

But, it's always good to have examples

of precedents to look to, to make sure we can avoid mistakes from the past.

MEMBER BALLINGER: Will this issue be factored into the QA Program? In the sense that thousands of drums, all cemented in some way, is there -- will there be a way to make sure that the properties are -- remain the same throughout this whole disposal process?

MR. LYNCH: I think in terms of how the Quality Assurance Program will address it specifically, I can have one of our quality assurance reviewers here.

MEMBER BALLINGER: I mean, if it's going to be discussed in another Chapter, that's fine. But --

MR. McILVAINE: The solidification is normally handled through the process control program. Which is a requirement at power plants. It's part of the technical specifications.

I would expect to have a similar requirement for SHINE that gives you both the waste characteristics and the proper mixture. The process if you will.

And it would be covered under the QA Program as any other technical specification

program would be.

MR. WRIGHT: This is Ernie Wright from Sargent Lundy Engineers. May I augment with what was just said?

MR. McILVAINE: Certainly.

CHAIRMAN BLEY: Yes, please.

MR. WRIGHT: Okay. You mentioned EPRI NP-2900. And indeed it dwells on boric acid because it was concerned with pressurized water reactors.

It also adjusts the sulfate waste. And it provides regions of proper solidification for sulfate waste and boric acid waste based on concentrations and the type of cement you're using.

So there's a lot of guidance there. In addition, we pursued third party testing of crowed wastes. So that's valuable.

And on your slide, the PCP, that's a process control program that assures you have a monolithic cement waste form. So, that's really the key, the process control program and how you write it.

MR. McILVAINE: Yes.

MR. WRIGHT: Okay. That's all I wanted to add. But it was a very good discussion. I

appreciate it.

MR. McILVAINE: Okay. Thank you. Any other questions or discussions? I think I'm finished.

CHAIRMAN BLEY: Okay. I have no idea how long we're really going to take on QA Program. Why don't we go ahead and try at least to get the SHINE presentation.

And then we'll break for lunch and hear from the staff after lunch.

MR. McINTYRE: All right. I got to go off script right off the bat here. Instead of saying good afternoon, it's good morning still.

So you might throw me all off. My name's Jim McIntyre. I'm with Sargent Lundy. I'm here to talk about Chapter 12, the Quality Assurance Program description. Next slide please.

10 CFR 50.34 requires the description of the Quality Assurance Program to be applied to the design, fabrication, construction and testing of the structure, systems and components of the facility.

SHINE is required to implement the guidance contained in Parts 1 and 2 of NUREG -1537, guidelines for preparing and reviewing applications

for non-power Reactors and associated in our staff guidance to meet regulatory requirements.

NUREG-1537 states the following ANSI/ANS-15.8 provides an acceptable method of meeting the Program requirements of 10 CFR 50.34. SHINE has developed the QAPD in accordance with the ANSI/ANS-15.8, which is the Quality Assurance Program requirements for research reactors. That's the 1995 edition, reaffirmed in 2013.

The SHINE QA Program description describes the administrative and engineering controls for ensuring compliance with requirements. And applies to the design, construction and operation of the SHINE facility.

Within the Quality Program, SHINE will apply a graded approach to those items and activities that could impact the quality of safety related SSCs and other components not specifically designated as safety related.

MEMBER SKILLMAN: Jim, let me ask this.

MR. McINTYRE: Yes?

MEMBER SKILLMAN: And I asked this at the first meeting with SHINE, which was months ago. Why not Appendix B to 10 CFR 50 instead of the ANSI standard?

MR. McINTYRE: Right. We are meeting the requirements of the ANSI standard. As part of the preparation for this, we did do a comparison with Appendix B.

And in all almost a word by word verbatim compliance and in all cases very few words, we were in compliance. For instance, I think it was Requirement 13 that talks about controls.

Some specific control wording wasn't specifically in the SHINE QAPD. And you know, we would put that in an implementing procedure.

So, those types of things. So we did do that comparison very close.

MR. COSTEDIO: But the real reason, I mean, we are meeting the requirements that are set forth based on our -- yes, I mean, based on our level of risk and what the NRC has put forth and said you will follow this. That's what we're following.

As we said in Chapter 6b, the radionuclide inventory in any one confinement area is approximately 10 thousand times less than a power reactor.

We believe, SHINE's position is that

ANSI 15.8 1995 provides a sufficient level of quality for our facility that's commensurate with the risk.

MEMBER SKILLMAN: I understood that from the original discussion that we had. But let me push it a little bit further.

MR. COSTEDIO: Yes.

MEMBER SKILLMAN: So, Appendix B to 10 CFR 50 is for power reactors and fuel facilities. This is not really a research reactor. It's not a power reactor either.

MR. COSTEDIO: This is not a reactor.

MEMBER SKILLMAN: Well, in a way it is.

MR. COSTEDIO: We don't produce spent nuclear fuel.

MEMBER SKILLMAN: No, you produce fission products. And you produce heat when you're creating your product. Okay?

MR. COSTEDIO: Okay.

MEMBER SKILLMAN: But, here's where I'm going, not so much to focus on the facility per se, but on the inspectors who will be doing the inspection, who really understand Appendix B in terms of construction.

And they could be your greatest allies

in making sure that the final product, your final facility is what it needs to be. So, I just wonder if perhaps that very important aspect has been overlooked.

Let me say it differently. You might say well, we're going to use the ANSI standards. It's plenty good enough for the risk that we have.

But it could be that the inspectors, and I'm presuming they're going to come from Region II, which is where the fuel facility inspectors come from, are deeply aware of how to inspect a fuel facility.

And they know the warts and wrinkles in the regulation. And they could give you the greatest value added in their inspection.

MR. COSTEDIO: I don't disagree with that. But I don't want to rely on an NRC inspector to be -- say, you know, to learn from that to say that, you know, our facility is safe.

We have to do that ourselves.

MEMBER SKILLMAN: Well, I concur with that.

MR. COSTEDIO: Our inspectors and us. That's our responsibility. And our QA Program needs to be able to have all those issues taken

care of so when the NRC inspectors come they don't find anything.

MEMBER SKILLMAN: Okay. But what I really hear you saying is, as far as the SHINE team is concerned, the ANSI standard is sufficient for the purpose that you intend.

MR. COSTEDIO: Yes.

MEMBER SKILLMAN: End of story.

MR. COSTEDIO: Yes.

MEMBER SKILLMAN: Thank you.

MR. ADAMS: Can I add something here just for clarification? As we discussed the last time we met that we are developing a construction inspection program.

And an aspect of that is the inspection of quality assurance. And the developers of that program, I think have looked at the differences and understand that yes, the inspectors are going to come out of Region II. And they're going to be, you know, experts in the quality assurance as -- you know, as described by Appendix B.

I see the biggest difference in Appendix B and the ANSI standard that I think the frameworks and sort of the water front they cover are the same. Appendix B has a lot more I think

detail in it then the ANSI standard does.

That was to allow the, you know, the flexibility of many different facilities that, you know, research reactor types. And you know, our extension, what we call non-power utilization production facilities now that the medical facilities are coming onboard.

MR. McINTYRE: Thank you, Al. Okay. Thank you. Next slide please.

Talked about the graded approach quality while implemented graded approach quality within the SHINE QAPD. Three levels are defined.

QL-1 for items, is items and activities shall implement the full measure of the QAPD. And shall be applied to safety related SSCs and to safety related activities.

QL-2 -- and an example of a QL-1 would be the target solution vessel for instance.

QL-2 is applied to selected SSCs and activities intended to support or protect the safety function of safety related equipment. QL-2 program elements are applied to an extent that is commensurate with the item's importance to safety.

Implementing documents will establish the program element applicability. Examples of a

QL-2 would be fire protection. It would be an activity that would fall under QL-2.

And lastly, all items that would not be QL-1 or QL-2 are QL-3. QL-3 is applied to non-safety related SSCs and activities. And does not support or protect the safety function of safety related SSCs or activities.

An example of a QL-3 would be the demin water system.

MEMBER SCHULTZ: Well, under QL-2, to an extent that is commiserate with the item's importance to safety. How is that being determined?

How is the item's importance to safety being determined? What type of listing? What type of valuation is done?

What type of categorization is done on a component by component, system by system basis to assign QL-2? And it sounds as if under QL-2 there's a variety of approaches that could be used under that category.

There is a QL matrix that I'll -- do you want to?

MR. HENNESSY: In the design of systems and components, the engineers will have to look at

the impact of that. Particular component and what it's failure could mean to a safety related system.

For example, how does it support a safety related system. There's not one answer for that. It's really a design is, we need to look at that for each component and determine.

So safety related is pretty clear cut. QL-2 is not so clear cut. So we'll have to review that and make sure that we understand it.

Some things are pretty, you know, straightforward like seismic tools are one. Fire protection, different things that require other codes and standards to be applied to that would be QL-2.

MEMBER SKILLMAN: Let me build onto Dr. Schultz' question. And as I ask these questions, I'm not trying to be a -- I'm not trying to harass you.

But we have experience around this table for years where we've watched. For example, a 50.59 program not being fully obeyed, allowing a major change, it enabled an inappropriate component to be installed.

So, my point is, clarity of what is quality class. And here's the example. On Table

351, you have your system classifications. That was the document that Jim just referred to.

And that is at a system level. At a system level there's an asterisks. And it says, this will be the highest qualification of the device in that system. And many could be lower. Or some could be lower.

And in that same Table, you identify your seismic classification with that same asterisks. Meaning some will be seismic one, but not all will be seismic one. Some may be seismic two.

And I grant you that is the designer's prerogative. But let me not be a designer for a minute. Let me be an inspector.

And I come in and say golly, I see that you've got your facility structure safety related category one. I guess that means the secretary's chair in the office where a visitor one enters is a seismic one chair.

And you're going to say, oh, we don't mean that. And I'm going to say, well, how do I know that?

So, where is your QCL? Where is your quality classification listings?

MR. COSTEDIO: We're going to have a master equipment list.

MEMBER SKILLMAN: You're going to have a master equipment list. But it's not available yet?

MR. COSTEDIO: No. But we don't have the complete design yet.

MEMBER SKILLMAN: But you're going to pour concrete. And I'm going to go back to my discussion earlier --

MR. COSTEDIO: Well, our schedule doesn't -- our schedule has us completing design and then --

MEMBER SKILLMAN: Before concrete is poured?

MR. COSTEDIO: Yes. And then builds it. Yes. That's the way it has it right now.

MEMBER SKILLMAN: I want to go back to a discussion we had perhaps two hours ago. And that is, you're going to be doing your ventilation system.

MR. COSTEDIO: Yes.

MEMBER SKILLMAN: And you're going to have vent duct work that is necessarily seismic one and quality one because it's in RCA Zones 1 and 2.

And that's going to attach to a framework. And that's going to attach to concrete.

MR. COSTEDIO: Yes.

MEMBER SKILLMAN: And involved there will be the fasteners, the devices and supports. And by my read, if I'm not a QA inspector, all of that is Q-1. Including the calculation that shows that that duct work is good for the seismic duty for that duct.

MR. COSTEDIO: That's the intent of it.

MEMBER SKILLMAN: That's the intent. Okay. So, you're saying you're going to have a Q list.

MR. COSTEDIO: Yes.

MEMBER SKILLMAN: Are you also going to have a safety related activities list? Because you do not have safety related identify -- safety related activities identified.

MR. COSTEDIO: You didn't have a safety related activity list at a power plant.

MEMBER SKILLMAN: Well, those of us who were in that world knew, when we did a calculation for seismic, when we did a calculation for flow rate for emergency core cooling, we knew that that calc was a Q-1 calc. And that we were on the hook

for Appendix B to 10 CFR 50 criterion three design control for that calc.

We knew that.

MR. COSTEDIO: We know that.

MR. SKILLMAN: Well, why not?

MR. COSTEDIO: We do know that.

MR. SKILLMAN: And where will that list be? So that an inspector can know that the calc for that duct, the foundation of that duct is a Q'ed calc?

MR. HENNESSY: Whether our calculation procedure identifies which calculations are safety related and which aren't, we already have that process in place.

MEMBER SKILLMAN: That's not well described in the documentation.

MR. HENNESSY: Well, it's an implementing procedure.

MEMBER SKILLMAN: So, an implementing procedure. Okay. Well, I'm going to start again. I think you've got a deficiency in your QA plans. Because you've identified what is an SSC, I think that's on page 25 or so of your QA plan.

But you do not have a definition of what is a safety related activity. And I believe

you need that.

MR. HENNESSY: Thank you.

MEMBER SKILLMAN: Thank you.

MR. McINTYRE: The SHINE QAPD for design construction modification is in accordance with the ANSI/ANS-15.8 as we stated a little bit earlier. And the SHINE QAPD will contain the following 18 requirements.

I won't read them. Up from organization down to assessment. Next slide.

Likewise, once we get to facility operations mode, we will also be in accordance with the ANSI 15.8. And the SHINE QAPD will contain the following 15 elements from organization all the way down to appropriate labeling. Next slide please. You got it.

Now that we've talked about the QAPD-1, I want to talk a little bit about the operational structure. First I want to point out, this has been revised based on the comments from the previous ACRS meeting.

These changes have been captured in the PSAR. Which has been revised to document some of the changes you see here. The plant manager changed to a Level 2. Ops manager changed to a

Level 3.

Chief operating officer now establishes the review and audit committees, holds approval authority for those activities and ensures that the appropriate technical expertise is available. And the ES&H manager now reports to the chief operating officer as you can see from the org chart here.

And all those have been captured. Next be sure on the slide about the previous slide with the organization with the organization under review and auditing activities. The review and audit committee can do these with the appropriate expertise and experience, established and numbers designated by the chief operating officer and provide an independent assessment of the operation.

Scope of the review function and the audit function are in accordance with Section 623 and 623 of ANSI/ANS-15-1 2007, the development of tech specs for research reactors.

Some key elements of that are upon the completion of the review, a written report of any findings and recommendations of the review committee shall be provided to SHINE executive management.

Deficiencies identified during an audit

will be entered into the correct action program. And deficiencies uncovered that effect nuclear safety shall immediately be reported to level one management.

That's all the procedures. Operating procedures provide appropriate direction to ensure that the facility is operated normally and within its design basis and in compliance with tech specs.

And in accordance with 15.1, SHINE shall prepare, review and approve written procedures for the following basic topics.

And that concludes my presentation.

CHAIRMAN BLEY: Okay. Anything more from the Committee?

(No response.)

CHAIRMAN BLEY: Well, thank you. And why don't we go talk -- is there any problem with waiting until after lunch? Okay. We'll recess until 1:00 for lunch. And we'll come back with the staff's presentation on QA.

(Whereupon, the above-entitled matter went off the record at 11:49 a.m. and resumed at 1:00 p.m.)

CHAIRMAN BLEY: We're back in session and we'll hear from the staff at this point on

Chapter 12, Quality Assurance.

MR. PRESCOTT: All right. Good afternoon, madam and gentlemen. My name is Paul Prescott. I'm with the NRO, the Quality and Vendor Branch, and was responsible for the Chapter 12 review of the SHINE PSAR.

Next slide. As you're well aware, it's been discussed a number of times today that 10 CFR 50.34 is what applies, specifically (a)(7). But also (b)(6)(ii) applies for the managerial administrative controls that are applied during the operations phase of the plant life. But as I'll tell you right now essentially they didn't submit a full program on operations. That's going to be submitted at a later date when the FSAR comes in, which is pretty much standard practice as we've seen from the reactor facilities in the past.

Next slide, please. Reg Guide 2.5 spells out what is acceptable to the staff for implementing the applicable regulatory requirements in 50.34. It addresses the acceptability of 15.8, specifically the 1995 version, for addressing the quality requirements needed for QA Programs for all phases of plant life for these types of facilities. It's similar to Appendix B that allows a graded

approach to a quality program. Unlike nuclear power plants in a few facilities the full scope of Appendix B is not directly applicable to facilities that fall under the scope of Reg Guide 2.5 as was discussed here today.

Next slide, please. NUREG-1537 contains the Guidelines for Preparing and Reviewing of Non-Power Reactors in Part 1 of 12.9, Quality Assurance. It outlines where the regulatory requirements exist in 10 CFR 50, which is under 50.34.

CHAIRMAN BLEY: Excuse me a minute.

MR. PRESCOTT: Yes, sir.

CHAIRMAN BLEY: I slipped up. We're not supposed to be running our meeting without our designated minder here.

MR. PRESCOTT: Oh.

CHAIRMAN BLEY: And I thought she was.

PARTICIPANT: She's here.

CHAIRMAN BLEY: Oh, there you are. Okay. I thought I had seen you and then I lost you. Okay.

My apologies. Please continue.

MR. PRESCOTT: That's okay.

CHAIRMAN BLEY: Are we all set, sir?

COURT REPORTER: Yes.

CHAIRMAN BLEY: Okay.

MR. PRESCOTT: Essentially what it says in 12.9 under NUREG-1537 is that the applicant should consider the use of 15.8 as a way to meet the quality requirements that are necessary to operate the facility. In Part 2 it has essentially a codicil talking about that for the QA reviewer that's going to perform that review he be notified that essentially this is a different type of review than a standard Appendix B review and that is for non-power reactors and that the guidance of 15.8 should be consulted in that review.

Next slide, please. So a summary of the application. Essentially SHINE provided sufficient information to make a determination that the QAPD is adequate for the conduct of design and construction activities. However, additional quality controls will need to be implemented for the operations and decommissioning phases of the facility's plant life. Section 3 of the SHINE's QAPD provides the elements of a QA Program for the conduct of operation, however, the staff is looking for greater level of detail, and that will occur when the FSAR is brought in.

Section 5 is currently left blank on decommissioning, but as we've noted that the staff has determined this level of detail is not needed at this phase for design and construction activities, which is essentially what they'll be doing for the most part now.

Next slide, please. We'll discuss some of the RAIs now. One of them had to do with definitions. When SHINE provided the definitions, the staff noted the applicant did not define the term "experiment." We held discussions with them and it was determined that no experimentation was going to occur at the facility, so they removed from Section 219 "experimental equipment." It was removed in its entirety, which was part of 15.8. Under Section 210 we noted under "Inspections" they discussed experiments, and they removed that also. So experimentation has been totally removed from the facility's QAPD.

Next slide, please.

MEMBER SKILLMAN: Paul, let me --

MR. PRESCOTT: Yes?

MEMBER SKILLMAN: -- just get a question in here. Going back to our discussion about criticality and the upper safety limit and

the comment made by the SHINE team that they will adjust the TSV uranium-235 concentration to ensure that the reactivity limit is met. If they were to use several different solutions, if you will, to get a target on K-effective, would that be considered an experiment? MR. PRESCOTT: I think that's really -- I got to apologize. I think that's really more of a question for the technical staff. I guess it would be deemed -- I would have to look to the SHINE staff to make a determination of whether or not it was an experiment under 50.59.

MR. COSTEDIO: Yes, when we look at the ANSI standard; this is Jim Costedio, research and test reactors actually will perform experiments as part of their charter and what they do. We don't plan on doing that. However, under the 50.59, Changes, Tests and Experiments, if there's some activity that comes up that would be deemed and experiment, we would evaluate that under 50.59. So I mean, it's not like a routine type of thing that we're going to be doing experiments at the facility.

MR. ADAMS: Can I add just a little bit here? So "experiment" is defined in the standard, and it's a long definition, but one part of it is

"any operation hardware target that is designed to investigate non-routine reactor characteristics or intended for radiation." So that definition of "non-routine reactor characteristics," depending on what the iteration that SHINE is doing could fall under that. However, if my understanding of the way SHINE is going to operate this facility that each and every time one of these TSVs is started, it is basically what I would call a critical experiment in that they would determine the criticality of the system uniquely each and every time. And I would expect that that operation would be under the auspices of the Quality Assurance Plan.

CHAIRMAN BLEY: Perhaps that's considered a test, I don't know, since it's routine. This is a non-routine --

(Simultaneous speaking.)

MR. ADAMS: Yes, it's --

(Simultaneous speaking.)

CHAIRMAN BLEY: -- routine, yes.

MR. ADAMS: -- we call them critical experiments, which indicates it's an experiment, but if you're doing it six times a week, that does -- at what point do you transition from a true

experiment, which once every seven or eight years I would do one when I worked at the research reactor because that's how often you needed to do that approach to criticality versus the way they're going to operate these facilities, that it's -- part of the routine operation is you make that determination of how close you are to criticality, that you've put enough solution into the vessel.

MR. SMITH: Just a point of clarification. You should probably refer to that as a sub-criticality measurement, not a criticality experiment.

MR. ADAMS: You're right. In this case you're absolutely correct.

MEMBER SKILLMAN: So where I am is words matter. And if my experience is any indication of what could occur, nine years from now some young chipper inspector shows up and says aren't you doing experiments? So it seems that because words do matter, this should be made clear so that the licensee is permitted to do the kinds of criticality validations that are necessary to run this facility safely. Words matter.

MR. ADAMS: Yes, they do.

MEMBER SKILLMAN: Thank you.

MR. PRESCOTT: Understood. Thank you.

On this next slide ANSI in ANS-15.8 does not define the meaning of either "audit" or "assessment," so there was quite a discussion with us between SHINE and the staff about how did you define it and would you be performing audits as typically thought of in the Appendix B world? Reason being we wanted to make sure that supplies that were going to supply the facility were adequately assessed of their ability to provide safe and good products to the facility. So we had quite a bit of discussion on that. And as you can see, they did provided a definition for that even though it's not required under 15.8, but they do discuss assessing and auditing in 15.8, but never define it.

Next slide, please. This RAI was a request to ensure that staff will be notified of changes to key definitions. And in the future should the licensee determine a change is warranted to a definition, we would see that change. As it was initially in the QAPD that they submitted they did not provide the definitions in there, and we wanted the opportunity should they change a definition such as a key definition as safety-

related the staff would know that it and it would not be buried in a lower-tier procedure which the staff would not be reviewing. So that managed to get changed.

Some key words that were changed to fit the facility that we discussed with them was "commissioning" and "management," where they changed out the word "reactor" for SHINE organization. And with the radiation facility for the commissioning as defined, which it states that is a process during which constructed reactor structure systems and components are made operational and verified to meet design requirements. And "management" means those persons within the SHINE organization whose responsibility and authority includes the QA Program. So those were changes that we had discussed with them.

MEMBER SKILLMAN: Paul, let me ask this.

MR. PRESCOTT: Yes, sir.

MEMBER SKILLMAN: I'm still stuck on my safety-related SSCs and safety-related activities.

MR. PRESCOTT: Yes.

MEMBER SKILLMAN: I'm curious why the staff didn't challenge the absence of the

definition of safety-related activities.

MR. PRESCOTT: We did. We did, sir. That was added to the definition to cover it. That was one of our RAIs.

MEMBER SKILLMAN: Oh, that's new information for me.

MR. PRESCOTT: Yes, that was changed.

MEMBER SKILLMAN: At least the current revision of the QAPD --

MR. PRESCOTT: When they talk about QL-1, QL-2, QL-3, that -- activities was also addressed in there.

MEMBER SKILLMAN: It is identified only twice and it is not identified on the --

MR. PRESCOTT: Of?

MEMBER SKILLMAN: -- definitions page of the QAPD. And that would be on page 9 of 25. Safety-related SSCs is identified. Safety-related activities are not described or addressed.

MR. PRESCOTT: Well, first off, the way it was broken down is the QA staff does not define safety-related. That's the job of the technical staff. We were part of it but we didn't have control over it. What we did question was the QL-1, QL-2, QL-3. And that we had included activities

which they didn't have before in their discussion of the QL-1, 2 and 3. And that's in the back. It's not in the definitions portion, but it's in the QAPD near the back of the document.

MEMBER SKILLMAN: And I'm there in the QAPD --

MR. PRESCOTT: Yes.

MEMBER SKILLMAN: -- and I will raise my question. Why isn't the term "safety-related activity" described? There is no definition of what is a safety-related activity. There is a definition of safety-related SSCs. That's on the current page 9 of 25 --

MR. PRESCOTT: Yes.

MEMBER SKILLMAN: -- of the QAPD.

MR. PRESCOTT: Yes, sir.

MEMBER SKILLMAN: But there is no definition of "safety-related activities."

Let me tell you why I'm boring in on this.

MR. PRESCOTT: No, I hear you. Keep going.

MEMBER SKILLMAN: Is the control of design calculations a safety-related activity? I believe it is.

MR. PRESCOTT: Yes.

MEMBER SKILLMAN: Are the calculations that show how the various components will be founded, how there are stresses and strains under seismic will be -- the concrete, is that calculation safety-related? I think so.

MR. PRESCOTT: Yes, sir.

MEMBER SKILLMAN: Some of the processes that were described by the applicant, I believe some of those processes are actually safety-related activities and need to be governed by procedures that are protected, if you will, under the QAPD.

MR. PRESCOTT: Yes, let me back it up a little bit. Just like Appendix B under Criterion 2 for QA Program it specifies that you will identify SSCs and activities that could affect quality, and therefore you have to control them. So the overarching QA Program would cover both your safety-related structure systems and components and those activities thereof. So therefore, like procedures, if it's a quality activity you have to have a procedure to address it.

MEMBER SKILLMAN: And I was looking for a definition that would communicate that because that definition is presently absent.

MR. ADAMS: And I think we will have that discussion with SHINE about going in that direction, because 15.8 clearly talks about not only structures but it talks about activities also. It has a list of activities affecting quality. It includes siting, design and purchasing, fabricating, handling, shipping, receiving, storing, cleaning, erecting. And I can go on and on and on.

MR. PRESCOTT: Right.

MR. ADAMS: But it's right there in 15.8 as being within the scope of the standard. So it

was --

MEMBER SKILLMAN: And mixing your TSV solution by the way.

MR. ADAMS: Right.

MEMBER SKILLMAN: Okay?

MR. PRESCOTT: Yes, it doesn't get specific. It just essentially covers in a broad sense any design phase of the operation. Design, fabrication, construction and testing are considered quality activities under 15.8. So if they didn't have a procedure, an inspector could go why is there no procedure for this quality

activity? So it is our opinion, just like the way we would address it in a current program, QA Program, for reactors it would be the same situation. They do not list every activity that's a quality activity. It's essentially understood that their program will cover quality activities. That's design control, procedure, procurement documents, inspection and test, M&TE, corrective action, all those things. All of that falls under the umbrella of your Quality Program.

MEMBER SKILLMAN: Yes, well, I'm comfortable with Al's explanation that this will be a topic to be discussed with the licensee. And my expectations are going to be that there is going to be a definition of what is a safety-related activity.

MR. PRESCOTT: Okay.

MEMBER SKILLMAN: So that for us now in the construction permit stage, but for the inspector 10 or 15 years from now we are aligned in what that means. Okay. Thank you.

MR. PRESCOTT: Okay. The technical reviewers for QA were not directly responsible for working with SHINE on the definition of "safety-related" as I had discussed, however, we did take a

major role in trying to shape what the scope was that was covered under the QA Program. The QA reviewers had a vested interest in ensuring Part 21 was appropriately addressed by SHINE. SHINE considered safety-related SSCs that would be covered under Part 21 to be basic components are defined in 21.3. So we feel comfortable that that has been addressed.

Next slide, please. We essentially talked about this already, that the QL-1 would apply to activities also which wasn't initially the case with their definition of QL-1. And QL-2 was covered by SHINE on what that is and the scope of that activity. Essentially that classification will include quality activities performed by the licensee to ensure that the QL-1 items are available and reliable to perform the safety functions when needed. And the QL-3 classification came about when the staff was questioning about what was the scope of non-safety and how did you define that? And so, they essentially came up with a QL-3 classification to cover that.

MEMBER SCHULTZ: Paul, your item (c) here on the slide, in the last item on the slide, you intend to have that written QL-3 rather than

QL-2, is that correct? You said QL-2. You show QL-3.

MR. PRESCOTT: Oh, yes. Yes.

MEMBER SCHULTZ: And so that occurs in (c), and then in your last bullet also.

MR. PRESCOTT: Okay. Thank you.

Next slide.

MEMBER SCHULTZ: (Off microphone.)

MR. PRESCOTT: Sure.

MEMBER SCHULTZ: (Off microphone.)

I'm sorry. I turned mine off. What I've heard from your description is that you had a dialogue with them related to the QL-2 classification, and based upon that you're satisfied that you have an understanding and that they have an understanding how they're going to proceed with that definition to appropriately define and rank the safety importance of the system's components?

MR. PRESCOTT: Yes, sir.

MEMBER SCHULTZ: And apply the appropriate conditions associated with the QA Program to those items?

MR. PRESCOTT: Yes, I heard that discussed that occurred earlier this morning. One

of the things that caught my attention was as you might be aware for safety-related the applicant define what their safety-related SSCs are. Then the technical staff reviews that and makes a determination of whether it's appropriate or not. And we can certainly question them. And I think what this helped do is define that line between what's QL-1 and QL-2, what supports the safety-related systems and what is essentially what we would define in the reactor world as important to safety, those things that support those safety-related SSCs. So I think it better defines for them what falls into that category, yes.

MEMBER SCHULTZ: But that's the categorization part. The second portion focusing on the QL-2 category is that then they're going to define some aspects of the program that apply to those.

MR. PRESCOTT: Yes.

MEMBER SCHULTZ: And that's available for staff review also once that's --

MR. PRESCOTT: Once that's --

MEMBER SCHULTZ: -- established?

MR. PRESCOTT: Right. Right. Right.

MEMBER SCHULTZ: Okay. So both the

classification as well as the application of the components of the QA Program will be reviewed by the staff?

MR. PRESCOTT: That's correct, yes.

MEMBER SCHULTZ: Good. Thank you.

MR. PRESCOTT: This was discussed earlier in the presentation, so I'll skip over this, but essentially as we said SHINE will address this at a later date and will be subject to staff review once they've developed and added it to the QAPD in sufficient detail.

And so, I'm kind of summarizing here finally that SHINE will not engage -- it is our understanding will not engage in experimental activities and will not modify or add experimental equipment to the plant.

And so in summary, the staff determined that the information that SHINE FSAR in 12.9, Quality Assurance, was sufficient and met the regulatory requirements and guidance to support the design and construction phases of the facility's life with the exception of operations and decommissioning and the QAPD is adequate for the near-term activities that SHINE would engage in.

Next slide. And of course as I said

earlier we weren't directly responsible for this, but as you're well aware 50.2 defines "safety-related," what safety-related is. For this facility the staff had it adjusted what that definition is to better fit the facility and to fit the needs of that facility. And we believe that it's an appropriate definition. And more importantly, it also includes for us -- because we're kind of the overseers of Part 21 -- but Part 21 was appropriately addressed by the facility. Even though that's not part of the QA review, we wanted to make sure we captured that, and it was appropriately captured. We believe it's an important regulation.

And that concludes my discussion, gentlemen.

MEMBER SKILLMAN: Paul, let me ask this question, and I want to be very careful how I ask this. Here before us is a medical isotope facility that's being treated as a research reactor. And those of us who have been involved in nuclear for decades watched a time when there was no Appendix B to 10 CFR 50. There was no QA Program until 1971. Some of us were around before then and know what it was like back then. And when Appendix B to 10 CFR

50 2 was published and then licensees and triple-S vendors were forced to -- "forced" is the right word -- to comply, there was a hue and cry that went on for a decade. And finally people got the message, this is really important.

So back to my point. Here is a team striving to license a medical facility as a research reactor. To what extent are they really bought in, from your perspective, on the importance of quality assurance?

MR. PRESCOTT: I think it's important to distinguish my --

CHAIRMAN BLEY: Microphone.

MR. PRESCOTT: I'm sorry. I want to be clear, it's not the staff's job to interject their perspectives. My job is to make sure that what they provided was adequate. I believe that what they've provided is adequate. I believe that the DCIP, the Construction Inspection Program, will provide adequate instructions and inspection procedures to inspect this facility. And I believe that knowing the inspectors as I do and being one myself that we will make sure that the facility is constructed properly or we will call them on it. It's that simple in my opinion.

MEMBER SKILLMAN: Yes, sir. Thank you.

CHAIRMAN BLEY: Thank you. We'll move on to the accident analysis for the production facility, Chapter 13b.

Joy, are you on the line?

PARTICIPANT: She just sent an email that said she got cut off.

CHAIRMAN BLEY: Okay. If she gets on, that will be good. If she doesn't get on, I have some comments she passed on to me.

I also have a few comments that Mr. Stetkar passed on to me in this area, so I'll try to interject them as we move through this. Well, I'll interject this first one just after you get started.

If in any of these areas we start to encroach on things that are proprietary, cut us off quickly and we save those questions for the closed session at the end. We don't all remember exactly what's proprietary and what's not, so you'll have to help us there a little.

MR. VAN ABEL: My name is Eric Van Abel. I'm going to discuss Chapter 13b, the RPF accident analysis for the SHINE facility.

If you remember from previous

discussions on the delineation between the IF and RPF, the RPF handles the target solution after it leaves the TSV and goes to be processed for isotope extraction and purification and associated processes. This presentation covers just the accident analysis in the RPF.

The two types of hazards assessments that were performed during the preliminary analysis were a HAZOPS study which looks at process upsets and deviations, what could happen to any given node in the system, and the preliminary hazards analysis which looked at an additional set of potential initiating events and accident scenarios based on the hazards present. We then grouped initiating events and accident scenarios together into common categories.

CHAIRMAN BLEY: This is a good point for me to interject John's first comments. And I think that we've already raised this I believe in the past, but I think it's worth doing. It will take me a minute to run through it. He notes that in 13b.12 that design basis accidents have been identified for potentially significant radiological consequences including maximum hypothetical external events, critical equipment malfunction,

fires, chemical accidents and may encompass the loss of off-site power and operator errors.

He goes on to say they don't seem address accident scenarios that may be initiated by failures of support systems or other malfunctions like spurious control signals that affect several processes throughout the radioisotope production facility. Examples of those would be -- and he gives a whole list of them: facility chilled water supply and distribution, radioisotope production and facility cooling system, facility instrument air, facility ventilation system for Zone 4, process vessel vent system and hydrogen detection, radiological integration control system, facility integrated control system, DC power system, facility fire detection and fire suppression system.

Then he asks why do accident analyses not evaluate the effects from failures or malfunctions of these support systems since they directly affect the front systems and where are the qualitative evaluations of potential radioisotope production facility accident scenarios dominated? So if you can address any of that now, otherwise this is something that we'll keep on the table as

we will be very interested in this.

MR. VAN ABEL: Yes, I'll make a couple comments on one. The process we went through was a comprehensive process. It goes through each node of the system, each vessel and process component and looks at deviations that could occur irrespective of how those deviations occur. So whether it came from a compressed air actuation signal issue that caused more flow into that tank or what the issue was, we looked at the deviations directly and what the consequences of those would be.

CHAIRMAN BLEY: In a HAZOPS sort of --

MR. VAN ABEL: In a HAZOPS sort of way, which is rolled into our preliminary hazards analysis in our ISA. We did look at those from the preliminary design information that we had available.

CHAIRMAN BLEY: So those are in your ISA, but we're not seeing that at this time?

MR. VAN ABEL: Yes.

CHAIRMAN BLEY: So I guess the place to leave this is we'll be very interested in that when we get to the operating and license stage.

MR. VAN ABEL: Yes, sir.

CHAIRMAN BLEY: And let me glance at his next one here.

MR. RESTREPO: (Off microphone.)

CHAIRMAN BLEY: Oh, sure. Go ahead.

MR. RESTREPO: Thank you. My name is Louis Restrepo. I'm the VP with Atkins. I'm the consultant for hazard analysis and accident analysis. A lot of those scenarios or events that you mentioned are explicitly identified as initiating events in the HAZOPS as far as the ISA process. So when you look at the ISA integrated safety analysis, you will see those explicitly for every note, every --

CHAIRMAN BLEY: Excellent. That's what we look for when we get a chance to see that. Go ahead.

MR. VAN ABEL: All right. Slide 3. Or, sorry. It's three on my page. It would be slide 50 on your page. Just shows an overview of the facility. Just to refresh you on the layout there, the RPF area is highlighted in the green boundary where the super cell is on the right there due to the extraction and processing of materials and the other hot cells near the bottom and the waste, then do solution cleanup operations.

The identification of the DBAs and IEs within the RPF included the HAZOPS and PHA that I mentioned. Those are pulled into integrated safety masses. The list of IEs and accidents that are identified in the ISG augmenting NUREG-1537 and the experience of the hazard analysis team in a range of disciplines. We had disciplines there from nuclear plant operations, nuclear process safety, reactor safety, people experienced in risk analysis and PRA and people experienced in hazard analysis processes themselves, HAZOPS and PHA analysis.

The design or the safety analysis done so far is based on the current preliminary design and we plan to update this as we go through detailed design and have more information available on the detailed layouts of system. Qualitative evaluations were performed within categories of accidents to identify the boundary on limiting accidents and scenarios in each category and then quantitative evaluations were performed to determine actual consequences.

Next slide. The categories are given here. I'm not going to go through each of these in detail. I won't spend much time on these.

Next slide, please. The first accident

is the maximum hypothetical accident, the MHA. You remember from our discussion last month we postulated an MHA with the radiation facility and the RPF. The IF MHA was not the limiting MHA. It's the RPF one that we'll describe in a moment here. The MHA is not required to be a credible event. As discussed before it's described in NUREG-1537 and it's just simply a non-mechanistic way to bound credible events to establish another limiting consequence. And it's a very conservative process. The RPF includes various processes that we looked at for MHA consideration including the isotope extraction processes, the target solution cleanup processes, the waste handling processes including the gas systems. The most limiting event was determined to be a simultaneous release of all five of our gas decay tanks at once, and we'll go through that sequence here.

So these NGRS gas storage tanks or gas decay tanks collect and store radioactive gas from the off-gas system, so the TSV is releasing gases that are captured by the TSV off-gas system. Those gases are then purged to these NGRS storage tanks. And these storage tanks hold the gases for decay for a period of greater than 40 days.

The hydrogen in that's released from the TSV during radiation is recombined by the TSV off-gas system through its redundant recombiners and it's monitored through its redundant hydrogen sensors before it's transferred to the NGRS. And NGRS also has hydrogen detection capability as well.

The five noble gas storage tanks are located in a reinforced concrete shielded cell. The penetrations to that cell are sealed to limit release of materials. And due to the low pressure of the tanks nominally up to 100 psi and the construction of the cell there's no generated missiles that would be able to breach the walls of the cell. The NGRS is assumed to be at maximum inventory at the time of the event, so we look at the worst possible situation of when this event could occur, and that's when the TSV off-gas systems have just transferred their least decay, their newest batch of gases to NGRS. The TSVs are assumed to be operating 10 percent over power for conservatism and the five NGRS tanks are all filled to capacity.

So the sequence of events. If the five noble gas decay tanks rupture simultaneously, the

contents are assumed to instantly be dispersed into the cell and the high radiation level in the exhaust duct work is detected by the RAMS and the RICS, the Radiological Integrated Control System, actuates the alarm for evacuation and the cell isolation dampers. The bubble-tight isolation dampers close in the inlet and outlet of the cell and the isolation dampers will be designed to close against postulated pressures from the event.

The assumed leak path factors are 10 percent of the activity release from the cell, bypasses passes the dampers and goes into the exhaust duct work out to the facility stack. And 10 percent of the activity leaks from the confinement boundary into the area where the workers are into Zone 2.

CHAIRMAN BLEY: And you also have an assumption of I think 10 minutes for getting out of some of these areas.

MR. VAN ABEL: Correct.

CHAIRMAN BLEY: Where are these assumptions justified, or is that something that will be later?

MR. VAN ABEL: So the leak path factors, we plan to do analysis during detailed

design to quantify the leak path factors. These leak path factors are based on the records that are available on the tanks and the size of the room and how much over-pressure you would get from that event, but we'll look at closely in detail design to make sure that's still valid.

CHAIRMAN BLEY: Same thing with the evacuation times?

MR. VAN ABEL: Yes, with the evacuation time we've done a calculation on evacuation times that show conservatively three-and-a-half minutes of actual transit time. And then you have to account for pre-action time, people to recognize the response time.

CHAIRMAN BLEY: I think Dick brought this up at an earlier meeting, but are you looking at upsets that could unbalance the ventilation system such that the pressures aren't the way you expect them to be which could make it almost impossible to open some of the doors, or are there assists built goes into the doors to make sure they can open under any DP in either direction?

MR. VAN ABEL: Do you want to talk about that now? I mean, we haven't designed the assist for the doors, but that's something we noted

and we'll keep track of while we're doing detailed design. I'm sure that you can open the doors certainly.

CHAIRMAN BLEY: Okay. Because most of your looking for accidents is looking for ways to release radioactivity, but if you have some additional problems with ventilation systems that just affect egress and access, that could make a big difference.

MR. VAN ABEL: Yes.

CHAIRMAN BLEY: Okay.

MEMBER SKILLMAN: Eric, for this event you've described it in 13b.215, and you make a statement here in the PSAR on your Chapter 13b.6, page 13b.6. The evacuation time is a conservative assumption. Workers in the RPF and IF are trained to immediately evacuate the area in response to a high-radiation alarm. Going to the end of that little paragraph: "Additional detailed radiological dose consequence modeling and analyses will be performed for certain areas of the facility to increase the evacuation time."

Why would you want to increase the evacuation time?

MR. VAN ABEL: The statement is saying

to increase the allowable time for evacuation, to get more people to evacuate, to show that this area is not as large of a concern.

MEMBER SKILLMAN: Then are you revising the words to communicate the allowable evacuation time? If that's what you meant, that's important. That would certainly have caused me to read this paragraph differently.

MR. VAN ABEL: Yes, not to make the evacuation take longer actually.

MEMBER SKILLMAN: Okay. Then let me suggest you look at 13b.215 and consider a word change.

MR. VAN ABEL: Okay. Understood. Thank you.

MEMBER SKILLMAN: Oh, one other question.

MR. VAN ABEL: Yes.

MEMBER SKILLMAN: On your slide 55 that you've just completed describing, and redundant bubble-tight isolation dampers on the inlet and outlet of the cell close. And we've got 10 percent leakage. What's the specified maximum leakage of the dampers, please?

MR. VAN ABEL: I don't know the

specified leakage off the top of my head, but there are standards that say what the leakage of that type of damper is.

MEMBER SKILLMAN: But these are the bubble-tight dampers that you're talking about, so it's supposed to be CCs per minute and not cubic feet per second.

MR. VAN ABEL: Yes, it's very long. They're called bubble-tight because you do a soap test and there's a rate of bubble growth.

MEMBER SKILLMAN: Would we be accurate in assuming that the equipment specification -- this is Q-1, that the equipment specification will require testing to confirm that bubble tightness?

MR. VAN ABEL: Yes.

MEMBER SKILLMAN: That's a yes?

MR. VAN ABEL: We'll talk about testing -- oh, we did? Sorry. I'm in Chapter 6. But, yes, there are testing requirements for that.

MEMBER SKILLMAN: Thank you.

MR. VAN ABEL: So in the figure on the right here the five tanks are shown in blue there. And all those tanks are assumed to rupture instantaneously.

Next slide, please. The gases pass

through the RVZ1 filters on their way through RVZ1. That includes charcoal adsorbers and HEPA filters, but no reduction is credited for those filters. The dose conversion factors are used to convert then the airborne material into doses to the public and to the workers using ICRP 30 conversion factors. SHINE is committed to using ICRP 72 during detailed design as well. The worker evacuation, as we discussed, was within 10 minutes. And the ARF and respirable fractions for noble gases used in the analysis are 1.0.

Next slide. The calculated dose consequences from the release are 3.6 rem TEDE to the worker. And that's below the 4 rem regulatory limit specified in 20.1201. And to the public it's 82 millirem, which is below the 0.1 rem regulatory limit specified in 20.1301.

These analyses are significantly conservative. There are five tanks that rupture simultaneously instantaneously with no mechanistic reason for that rupture. These tanks will have appropriate isolation between the tanks. They'll be safety-related tanks and seismically designed. And the isolation will ensure that if one tank ruptures it does not cause multiple tanks to

release their contents. But we assume that here just as a purely non-mechanistic reason to bound the event. And the reason that's important as we go through the accident analysis is that's a larger driving force for these leak path factors that we discussed. It's a greater over-pressure in the cell and a greater driving force.

CHAIRMAN BLEY: How do you ensure one doesn't affect the other? Do you have shield walls between them or something?

MR. VAN ABEL: We haven't done the specific details whether there will need to be a shield wall between the cell or whether -- these are relatively low-pressure tanks -- whether we can show that another means would prevent missiles or anything from damaging the other tank.

Another very conservative assumption is that all of the noble gas from the target solution is assumed to evolve into the gas space, 100 percent fraction between the gas space and liquid space. There's no hold up in the liquid at all. These are very small quantities of noble gases, and all that noble gas is transported to the NGRS. Small quantities in terms of grams, of course.

Another assumption is that the five

NGRS tanks are completely filled, which is beyond our normal plant operations we plan to be filling to and releasing and storing to for decay. And the fifth tank would be a spare capacity. So that was another conservatism. And the isolation dampers in the RVZ1 main duct work would also close on the high radiation signal, but we don't credit those. So that would actually trap a lot of the radioactive material on the exhaust duct work as well, but we don't credit that. That would produce doses to the public significantly as well.

Okay. Next slide, please. The next event is external events. We looked at potential external events as initiating events including design basis earthquake, tornados and high winds, aircraft crash into the facility. As we discussed in Chapter 3 of the PSAR, the SHINE production facility building is designed to survive postulated wind, tornado, seismic and aircraft crash loads. Safety-related SSCs are also designed as seismic and they will be shown to perform their safety functions under loading conditions of the design basis earthquake. Therefore, there were no consequences to the worker or public resulting from the external events category.

MEMBER SKILLMAN: Eric, I'd like you to back up a slide. I was kind of taken aback by your comment there is not much material there in grams. We learned a lesson a long time ago. That number is a big number and 6.02 times 10 to the 23 times a fraction of weight is an awful lot of atoms --

MR. VAN ABEL: Yes.

MEMBER SKILLMAN: -- which is an awful lot of activity.

MR. VAN ABEL: Yes, there are --

MEMBER SKILLMAN: And I just want to make sure that you and I are aligned on that.

MR. VAN ABEL: To clarify my comment, I was talking about noble gas solubilities in the water. And they're saying that there's not a large quantity of grams because that's related to the solubility of the water. The activity is certainly significant.

MEMBER POWERS: Noble gas solubility in water is not too high, either.

MR. VAN ABEL: No, it's not.

MR. VAN ABEL: Yes, depends on --

MR. SMITH: Do you know what the curie inventory of these tanks is?

MR. VAN ABEL: Yes, we should -- that

is proprietary.

MR. SMITH: Oh, I'm sorry.

MR. VAN ABEL: If you want to talk about that, we can talk about that during closed session.

The next scenario category is critical equipment malfunction. This DBA looked at malfunction or mishandling of equipment that could loss of radiological control. Vessel line and valve failures were looked at, misalignments of valves and other process equipment failures. The actual scenario that was most limiting was very similar to the MHA as we'll describe in a moment. The systems and components processing irradiated materials are located in shielded hot cells, process cells and tank vaults. There is significant shielding around all of our equipment that's handling irradiated materials from the radiation process, nominally four feet thick reinforced concrete shielding around these processes. The major processes are the molybdenum extraction and purification systems, the UREX and thermal denitration subsystems, the waste treatment systems and the noble gas removal system.

The limiting event was thought to be

the inadvertent release of a single NGRS storage tank due to a tank leak. Normally we will have safety interlocks that ensure that those tanks have the appropriate decay time before they're released and vented. Also this event assumes to release entire tank contents into the room unexpectedly due to a leak or failure in the tank.

The selection of what tank has the leak is the most conservative selection. That's the tank that has just filled with the most recent fission product activity from the off-gas system. It's just been filled to capacity and that results in the highest inventory, the highest number of curies inside of that tank.

The sequence of the event is the RCA ventilation is operating normally and the NGRS is operating normally prior to the event. So it's conservative to assume the ventilation is operating normally on that cell. And the most recent TOGS purge volumes are just transferred to that storage tank and the leak develops in the storage tank that instantaneously release the contents to the cell. High radiation levels are detected by RAMS and the RICS initiates the high radiation alarm enclosure of the isolation dampers on the cell. The

personnel evacuation of the RCA is assumed to occur within 10 minutes.

MEMBER SKILLMAN: How difficult will it be for an able-bodied worker to exit that quickly?

MR. VAN ABEL: So we looked at the facility. It's a relatively small facility. We looked at where a worker could be that would be most difficult to get out of and we found cells, being potentially located down in cells would be most difficult. And we calculated the time and we observed the estimates of how long it would take them to get out. And I think it's -- and the times are not -- as I mentioned before, we calculated three-and-a-half minutes as the maximum transit time.

MEMBER SKILLMAN: I'm not trying to be picky here, but when you do that study, how many doors, barriers, fences, hurdles, shields would a worker need to find his or her way through to achieve the three minutes? And including, as Dr. Bley said, where you're fighting against a delta P where the door is being held closed because leakage is to be in and not out.

MR. VAN ABEL: The facility is relatively open on the production levels. The most

doors would be someone from the IF facility. They'd have to get out of the IF facility through that door and get out of the RCA through the RCA personnel door.

CHAIRMAN BLEY: When you consider these, have you thought about what kinds of maintenance and operating activities people might be engaged in and the equipment they might be using for that and whatever protective clothing they might have to have for some of that, how that would affect their evacuation?

MR. VAN ABEL: Yes, well, we looked at conservative lower bounds for mobility, for transit times, walking speeds, climbing ladders. We tried to balance those effects that they would be potentially slower than you would expect with a person on the street.

Similar to as described before for the MHA, we assumed 10 percent of the airborne activity in the shielded cell as leaks through penetrations and 10 percent bypass of the bubble-tight isolation dampers. It should be noted that we used the same leak path factors here as we did for the MHA. That was a conservatism as we don't have the calculations in place to have the specific leak

path factors. But since only one of the five tanks is rupturing, the driving motive force for release from the cell is much less. It's roughly one-fifth. So these actual leak path factors are very conservative and would be less for this event. And we'll perform the leak path factor calculations during detailed design to validate the actual release fractions.

The calculated dose consequences are 3.6 rem to the worker and 82 millirem to the public at the site boundary at the fence.

The analysis itself is conservative. Again we're assuming 100 percent of noble gases are released from the target solution. We're assuming that the release is complete and instantaneous and that we don't take any credit for the main isolation dampers on the RVZ1 exhaust downstream of the filters to close, which would significantly reduce dose as well.

Next slide, please.

MR. ADAMS: I'm sorry. I don't want to interrupt, but can I ask a question about that slide you just had up?

MR. VAN ABEL: Yes.

MR. ADAMS: I noticed that the doses

from this critical equipment malfunction are the same as the MHA. Is that true?

MR. VAN ABEL: Yes, they round -- they're not exactly the same. They round in the interest of significant figures to the same. And the principal reasons for that are we picked the tank -- these tanks are decaying, so the one tank that has the most activity has the vast majority of activity because the other ones are decaying. And the other major conservatism is the leak path factors are the same. We just assume the same leak path factors for the MHA as a conservative initial basis, but they would be a lot less in the final design.

MS. BANERJEE: Can I ask a question about the door space? Are you going to have any interlock pressured doors that you have to open one before you can open another one? Did you consider those kind of doors?

MR. VAN ABEL: We have airlock doors. That would be -- want to say anything about interlocks?

MR. COSTEDIO: I don't know how they work.

MS. BANERJEE: Like you have to open

one before you open the other one?

MR. VAN ABEL: Yes, you have to open one and the other one has to be closed before that. We have airlock doors.

I'm sorry, Jim. You want to say something?

MR. COSTEDIO: Well, there would be lights, like a green light saying you can go, or what if somebody's coming in the other way?

MS. BANERJEE: Yes, that could take longer. That's right.

CHAIRMAN BLEY: Where are you going to have airlock doors? I kind of missed that.

MR. VAN ABEL: The Zone 3 ventilation is essentially a barrier between the Zone 2 and the RCA --

(Simultaneous speaking.)

CHAIRMAN BLEY: Between 2 and 3?

MR. VAN ABEL: The Zone 3 itself are essentially airlocks.

CHAIRMAN BLEY: Okay.

MR. COSTEDIO: Zone 3 is the airlocks. The space inside that airlock is Zone 3. There's six --

CHAIRMAN BLEY: Okay.

MR. COSTEDIO: -- airlocks in the facility.

MR. VAN ABEL: And it's a low-leakage airlock. It's not a pressure retaining containment airlock. It's just a low leakage set of double doors.

CHAIRMAN BLEY: Okay. So it's not like the zero leakage ones? I'm thinking --

(Simultaneous speaking.)

MR. VAN ABEL: No, it's not a containment airlock.

CHAIRMAN BLEY: So don't take a long time to get through those?

MR. VAN ABEL: No, it's not --

(Simultaneous speaking.)

CHAIRMAN BLEY: You just have to have one closed before the other opens?

MR. VAN ABEL: Yes.

CHAIRMAN BLEY: Okay.

MR. HENNESSY: No, although we haven't designed it yet, it's quite likely that during an emergency situation you'll just go through them. We need to look at that, though, yes.

MR. COSTEDIO: And train the folks.

MR. VAN ABEL: Next slide, please.

Next event category is inadvertent nuclear criticality in the RPF. This section of the accident analysis looked at where SNM could be located, and therefore where accidental criticality is possible. And the controls based on the preliminary design that would be implemented to make that likelihood.

The six main process areas that involve handling SNM are listed on the bottom of the slide there. It's receipt of fresh uranium --

CHAIRMAN BLEY: I'm sorry. I should have asked this one sooner. John had asked one. In the document you summarize some results of a maximum hypothetical and design basis accidents. And if we got it right, the maximum hypothetical is releasing from all five tanks and the design basis is only from one, but the comparison table shows essentially identical dose results from that. Why is that?

MR. VAN ABEL: Yes, that was Al's --

(Simultaneous speaking.)

CHAIRMAN BLEY: Okay. I'm sorry.

MR. VAN ABEL: The main reasons are the leak path factors being assumed the same for now, even though the leak path factor of a single tank

would be a lot lower. And the amount of curies in one tank is the majority of the curies in that system. Because the other tanks are decaying there's less curies.

CHAIRMAN BLEY: Okay. For the design basis, it's the most recent tank?

MR. VAN ABEL: Yes.

CHAIRMAN BLEY: Okay.

MR. VAN ABEL: Dissolution of uranium oxide in sulfuric acid is one of the processes. That's how we produce the target solution. The transfer of the target solution then to the TSV in the irradiation facility. The return of the target solution back through the extraction processes where we extract out the medical isotopes. The cleanup of irradiated target solution periodically. And then the conversion of the clean uranium product back to uranium oxide.

A preliminary evaluation of scenarios was performed that could lead to inadvertent nuclear criticality. This included looking at leaks in piping or process equipment, accumulation of material in unexpected places, vessel overflows, misdirection of fissile material into unexpected areas.

Engineering controls and administrative controls were identified based on the evaluation to ensure that each identified scenario is highly unlikely. And that's described in Chapter 6. The NCSEs will be performed during detailed design. And that's where we really look in detail at each process to ensure that there's a set of controls in place that demonstrates double contingency principle is met and that the criticality is prevented in each potential situation.

CHAIRMAN BLEY: John asked one question about criticality analysis, and I don't want to read the whole long thing he wrote, but basically in one area you rely on neutron-absorbing materials being present to prevent criticality. And he asks what type of neutron-absorbent material is included in the uranyl sulfate preparation tanks. And then suppose that a potentially critical concentration of uranium is present in that tank, but critically is presented by the absorber. How is sub-criticality ensured in subsequent downstream process piping that may not contain the same absorbing material?

MR. VAN ABEL: I'm not sure I totally followed the steps there.

CHAIRMAN BLEY: I'm not sure I do either, but I wanted to get it on the table.

(Laughter.)

CHAIRMAN BLEY: I'm trying to keep up with what he said as well as --

(Simultaneous speaking.)

MEMBER POWERS: The answer is you're geometrically safe downstream.

CHAIRMAN BLEY: Thank you.

MR. VAN ABEL: Yes, the process tanks are geometrically safe.

CHAIRMAN BLEY: He's right?

MR. VAN ABEL: Yes. The only ones that --

CHAIRMAN BLEY: That's where we go from there is to the geometrically --

(Simultaneous speaking.)

MR. VAN ABEL: Yes, the downstream ones are. The waste tanks are on the --

(Simultaneous speaking.)

CHAIRMAN BLEY: That makes sense. Thank you.

MEMBER POWERS: I vaguely understood what he was driving at, but it was not very articulate.

MR. VAN ABEL: And the --

MEMBER POWERS: But I do also know what your answer is.

MR. VAN ABEL: The SSCs that are part of these nuclear criticality safety controls will -- they will ensure that criticality is highly unlikely. Those will be designated as safety-related as discussed previously in Chapter 6.

Next slide, please. The RPF facility fire. We looked at initiating events that have potential to damage safety-related SSCs within the RPF and lead to radioactive release.

Fire events were considered for normal operations and for maintenance operations within the facility both inside and outside of our shielded process enclosures. Postulated fires included fires from equipment malfunction, ignition of transient combustibles, loss of material control, propagation of fires from other areas, and exothermic chemical reactions.

It was assumed that small quantities of combustible materials are in the SHINE process equipment. It's a working note that our processes are fairly small. Our pumps and that are small pumps, so it's not like we have large equipment,

and those assumptions will be verified during detailed design. MEMBER POWERS: One of the persistent hazards of reprocessing systems for uranium is always hydroxylamine. I don't think the UREX system actually uses hydroxylamine. I think they use something else. Maybe it's an ethyl hydroxylamine or something like that. What do we know about the storage and the combustibility and the hazards associated with that material?

MS. KOLB: This is Catherine Kolb. It does not use a hydroxylamine. The chemicals that were used in the UREX system were evaluated for their --

MEMBER POWERS: What is it that you use in the place of a hydroxylamine?

MS. KOLB: It's AHA, Acetic hydroxylamine acid.

MEMBER POWERS: The disadvantage of hydroxylamine of course is that it's obnoxious material. The advantage is we have a huge amount of experience working with it. I have no experience with the acetyl analog. What do you know about that? I've never used it. I've never stored it. I've never played with it, burned my fingers with it, all which I've done with

hydroxylamine. I know for instance hydroxylamine has an autocatalytic reaction of rust that usually bites you when you least -- when and where you least want to get bitten. And I don't know about your acetyl analog for that. What do you guys know? I mean, what's the experience base for this material?

MS. KOLB: Yes, I don't have the information with me right now.

MEMBER POWERS: How much of an inventory do you maintain on the site?

MS. KOLB: The inventory of that is small. It's less than --

MR. VAN ABEL: Yes, it's small.

MS. KOLB: It's --

MR. VAN ABEL: We can get back to that.

MS. KOLB: Yes, we can get back to you on that, but I mean, it's --

MEMBER POWERS: If you would.

MS. KOLB: Yes.

MEMBER POWERS: And anything you know about it, because the material is -- people have gone to it because it's less obnoxious than hydroxylamine, which periodically blows up facilities. Nowadays it blows them in China, thank

heavens, instead of here. But I mean, the history of hydroxylamine has really been --

PARTICIPANT: Checking amounts. Jeff, do you want to read it off?

MR. BARTELME: Yes, the bounding inventory of AHA --

PARTICIPANT: Identify yourself.

MR. BARTELME: Jeff Bartelme, SHINE. The bounding inventory in the PSAR is 111 pounds.

MEMBER POWERS: That's enough to get your attention. We've had some really horrific autocatalytic decompositions in hydroxylamine, but I have to say I just don't know this material. I mean, it's not one of my good buddies that I've had a chance to play with. And so, I ask you to make me smarter so I can ask you hard questions.

MR. BARTELME: Understood. Thanks.

MR. VAN ABEL: We have hot cell fire detection and suppression systems and the facility fire detection and suppression system which provide control to reduce fire consequence should one occur. And the most limiting fire that we thought was the fire affecting the moly eluate hold tank with a supercell. We'll discuss that sequence on the next slide.

The figure on the bottom there shows the supercell layout, the extraction portion on the left there and the moly eluate hold tanks would be inside of that part of the supercell.

The design basis fire is assumed to occur while we're processing radiological materials. The hot cell fire detection is activated and alerts operations personnel of the fire. The hot cell ventilation is automatically isolate and the fire suppression system, while not credited, would be activated automatically or manually to help reduce consequences. Due to the thick radiation shielding, nominally four thick of concrete, fire damage is limited to the interior of the hot cell.

The RVZ1 exhaust trains filter the release to help the filters that are assumed to remove 99 percent of particulates. And the charcoal adsorbers are assumed to remove 95 percent of halogens.

The calculated dose consequences are --

MEMBER POWERS: What kind of loading are you going to get on your HEPAs?

MR. VAN ABEL: Loading in terms of like --

do you want --

MR. RESTREPO: Well, given the combustible load -- the amount of radioactive material and chemicals are going to be small. They would be mostly bounded by the amount of combustibles that may be present. Right now there's going to be at least in the PSAR loading of about one pound per square foot, or something like that, which is really not going to be much loading on any HEPA filter --

(Simultaneous speaking.)

MEMBER POWERS: Well, that's getting awful close to a kilogram per square foot.

MR. RESTREPO: What?

MEMBER POWERS: That's going to be getting awful close to the maximum loading you can have on a HEPA, isn't it?

MR. VAN ABEL: The combustible loading of in the rooms would be one pound per --

MR. RESTREPO: Yes, one pound per square foot.

MEMBER POWERS: Not one pound on the HEPA?

MR. RESTREPO: No, one pound per square foot on the floor.

MR. VAN ABEL: On the floor.

MEMBER POWERS: On the floor?

MR. VAN ABEL: Of combustibles.

MR. RESTREPO: Which is not significant to even create a fire, never mind loading it.

MEMBER POWERS: Well, depends on how many square feet you have. I would be interested in what your projections are on the loading because --

MR. RESTREPO: We use them as part of the leak path factor calculations. And when we brought in MELCOR of course like that, we'd be looking at the byproducts and looking at the loading and those HEPA filters to see if there's any -- causing any problems with respect to that.

MEMBER POWERS: Yes.

MR. RESTREPO: And during the detailed design we'll be looking at that.

MEMBER POWERS: Yes, and you'd feel a little more comfortable if there was a roughing filter in front of the HEPA.

MR. RESTREPO: Oh, certainly. Certainly that's typically what you do for design in most cases.

MR. VAN ABEL: And we do have roughing

filters on the exhaust on the hot cells, too. They're just not credited.

MR. RESTREPO: We learned that from Rocky Flats real quickly.

MEMBER POWERS: Yes, I guess you and I know that pretty well.

MR. RESTREPO: Yes.

MEMBER POWERS: Yes, because I mean, the trouble with HEPAs is overloading them or getting them wet.

MR. RESTREPO: Or overheating it, because --

(Simultaneous speaking.)

MEMBER POWERS: Or overheating it, but I don't think you got enough heat load here to cause that. But especially in fire situations you can overload them pretty easily. But if you've got roughing, then you can -- I mean, roughing will take out most of your --

MR. RESTREPO: Correct, most of it.

MEMBER POWERS: -- load. And then you're just worried about the finer stuff.

MR. RESTREPO: But those are the kind of calculations we'll be doing during the detailed design once we know the configuration of it.

MEMBER POWERS: Yes, and when you get to that, that would be interesting to see what you come up with there.

MEMBER SKILLMAN: I would like to ask what is the material used for hot cell fire suppression?

MR. VAN ABEL: I do not remember at this time whether this is a water-based suppression system.

MEMBER POWERS: The answer is it better be a water-based system.

MEMBER SKILLMAN: It better be. Better not be carbon tetrachloride or you're going to make phosgene. You got to be careful what that is. But I'd be curious. And it's my fault for not looking that up. But I would just be curious what is your suppressant? Is it a halogen or is it a water material?

MEMBER POWERS: There have been some interesting hot cell experiments done in Great Britain on using non-water-based suppressants, and the problem is always the same: the plastics re-combust as soon as they take away the suppressant.

MEMBER SKILLMAN: So if it's water, you may have a criticality issue.

MR. VAN ABEL: Yes, the criticality -- the NCSEs will look at water from fire protection efforts.

MEMBER POWERS: Well, I'd assume the collection trays would all be geometrically safe.

MR. VAN ABEL: Yes, the --

(Simultaneous speaking.)

MEMBER POWERS: It wouldn't --

(Simultaneous speaking.)

MR. VAN ABEL: -- geometrically safe sumps that drain the geometric --

(Simultaneous speaking.)

MEMBER POWERS: Yes, the drain pans, you just need to make them geometrically safe. Yes, because these halogen systems, they just -- I mean, as soon as you quit spraying them, air comes back in. All plastic cabling and tubing and things like that just reignites on you.

CHAIRMAN BLEY: That's interesting. The other side though, trying to make fire protection drain system geometrically safe we'll practically assume it works, and it will have to drain somewhere. And if that doesn't work, you need to look at the reliability of those systems to operate properly. Because you have all the water

in the world if you keep putting it in.

MEMBER SKILLMAN: I think we better be careful in what we talk about in chemicals. I'm looking at Chapter 13. I'm on page 13b.45 and most of this is proprietary and withheld. So we're being pretty liberal about talking about materials and amounts of material, but it looks to me like we need to be careful what we're saying here.

MR. VAN ABEL: Yes. Thank you. The dose consequences for this event are 0.58 rem TEDE to the worker and less than one millirem to a member of the public at the site boundary.

Next slide, please. Next accident category we looked at was accidents with hazardous chemicals produced from licensed materials. This DBA category looked at accidents involving chemicals produced from licensed materials or chemicals that could affect the safety of licensed materials. Chemicals are generally stored and used in small -- less than 1,000 pound quantities at the SHINE site.

The chemicals support a variety of process operations including the isotope extraction process, the target solution production as we produce the uranyl sulfate, target solution

cleanup, and waste processing.

We looked at a number of initiating events that could lead to a release of hazardous chemicals from licensed materials including failures of tanks or vessels and piping components associated with them, failures of tanks or vessels specifically due to fires inside or outside the tank vaults themselves, exothermic chemical reactions, spills of hazardous chemicals during handling operations, and unstable degradation products involving TBP and nitric acid related to the red oil events.

The analysis assumed that postulated IEs impact the entire inventory of the chemical in a particular area.

Eleven chemicals were identified as requiring further analysis based on their toxicity, dispersibility and inventory. And we used the five factor chemical dose formula to determine the source material released to the environment. And the modeling for that was done using the EPIcode.

For the nitric acid and n-dodecane release scenarios, those scenarios occurred in hot cells with confinement isolation dampers, and the bubble-tight dampers are credited with reducing the

leak path factor to 0.1.

The calculated chemical dose concentrations for the nearest resident are below the PAC 1, 2 and 3 levels and the worker concentrations are below the PAC 2 values.

In the next slide we had some questions from one of the members last meeting on specific values for the red oil prevention features. And Cathy Kolb is going to go over those specific values.

MS. KOLB: Yes, we had a request to present this slide with the nominal values and controls for red oil prevention features.

This slide is the area of the plans where we use tributyl phosphates and nitric acid in the same process, the UREX system. These are the different contactor sections in the use in the UREX system. We are using the recommendations from the Defense Nuclear Facility Safety Board TECH-33 documents for the basis for our controls. These are the nominal values for expected temperatures and concentrations as presented mostly in Chapter 4 of the PSAR. We intend to establish specific limits with margins of safety to the DNFSB recommendations for final design.

In addition we have a solvent control program that is part of the administrative controls in Chapter 14. That includes controls that were identified in our ISA, our initial ISA for our solvent residence times, solvent wash quality control and solvent sampling analysis to monitor impurities in degradation products from tributyl phosphates that contribute to red oil events.

We also have an intention for sizing the ventilation systems per the recommendations in DNFSB/TECH-33 for sizes compared to the maximum expected amount of red oil you should have -- you could potentially have in a vessel.

MEMBER POWERS: You should have a limit on --

MS. KOLB: Pardon?

MEMBER POWERS: You should actually have a limit --

(Simultaneous speaking.)

MS. KOLB: Well, we don't have a limit. We have the recommendation from the documents. It's 208 grams potentially generated of red oil per millimeter square vent area. So we're using that.

And I realize that some operating experience you don't necessarily know because there

can be hold up in tanks and site solubility and nitric acid. So we're aware of those.

MEMBER POWERS: Yes, the biggest problem is hold up and aging. I mean, the central element of the DNFSB is just not to accumulate. And it's the post-operational cleanup that becomes the critical element here in your designed system so that you're not solely accumulating red oil degradation processes. And circulate -- your cleanup system is where you use your solvents and things like that. It's the cleanup on those that's really critical for -- I shouldn't use the word "critical" in this context -- real crucial.

In this regard one has to always remind people DNFSB recommendations are not based on some physical limit. They represent a lower bound on where incidents had been observed and they will fall when we find an incident at 120 degrees instead of 130. It will come down when we get incidents at 8 instead of 10. And they're strictly empirical-based. There's no physical phenomenon that's eliminating the red oil reaction here. And so the real crucial things are your cleanup and avoiding of holdup in the systems.

MS. KOLB: We understand your comments

and we agree with those. I need to point out that we do not plan to have a solvent wash system. We plan to periodically replace our solvents and dispose of it.

MEMBER POWERS: You'll replace it?

MS. KOLB: Correct.

MEMBER POWERS: Yes, then it becomes a question of what "periodic" means. Periodic can be once a decade.

MS. KOLB: I can't remember the exact words in the PSAR, but it was between six months and a year for --

MEMBER POWERS: Yes, well, that means how many times --

MS. KOLB: Yes, per cycle. It's something we need to look into.

MEMBER POWERS: How many cycles it's gone through is really the operative time schedule here, not some calendar time.

MEMBER SKILLMAN: Catherine, let me do a hearing check. I think you said 200 milligram -- or excuse me, 200 grams per square millimeter. That sounds like an awful lot of material. Maybe I misheard.

MS. KOLB: You did not mishear. That

is the value that's cited in that document. One square millimeter vent area every 208 grams of potentially generated red oil. And we went back to the source documents from where that came from. It was Savannah River documents. And those are the correct units. It wasn't a unit error.

MEMBER SKILLMAN: Thank you. Okay. Thank you.

CHAIRMAN BLEY: But back to what Dana said, they looked at the size vent they had when it blew up and said it has to be better than that. Don't know how much better.

MEMBER POWERS: Actually I think the vent sizes came from some experiments that were done with Fauske & Associates, if I'm not wrong, in his -- that does still strike me as -- I would not use that as an absolute. I would use that when I had a big amount. Something that size is going to be pretty large. I wouldn't use it if I had 208 grams, which is a number that you might have in your process. I wouldn't say I can get away with one square millimeter. I think you have to be careful about extrapolation, but I do believe that the vent sizing was based on some combination, experiments that were done at Fauske & Associates.

The problem with that and the problem with all experiments with red oil is that it's done -- nobody knows that they really had red oil. In fact, the only thing I know about red oil is it's very definitely not a oil and it may not even be red. And so all you know is Fauske & Associates was working with something and got those numbers. And that is true of every laboratory experiment done with red oil that I know of. They were working with something. It may or may not be red oil. The problem with red oil is after an event it's destroyed the evidence. The only reason we call it red oil is that after the first run some people found some red slimy stuff on the site and they assumed that that was the culprit.

CHAIRMAN BLEY: Two things: One, this is more comforting than where I was thinking you were the last time by your distance from these. And some of the more modern facilities I've seen built recently track to eliminate an element that they can completely. And if they can't, they both have process controls. And before they move a batch they sample and test to make sure they're okay.

MS. KOLB: If you could go to the next

slide? These are the areas in the facility where we do not plan to actually have organics and nitric acid in the same process, but they have the potential if there was carryover or an upset. So these controls here, you're talking about eliminating things. I mean, like of course for denitration you can't control -- you can't use temperature as a control because it is by necessity operating above the temperature. But in all cases we use at least two of the controls. And sampling, we will have sampling for transfer between -- for sampling of organics into containers where they are not supposed to be.

Yes, this is the end of --

CHAIRMAN BLEY: John Stetkar had a few more comments in general, but they all link to assumptions. So everywhere through the safety analysis he's asking for justification of the assumptions. And we all have asked about that.

Joy Rempe had a whole section that dealt with that, and I won't go through it. But then two others that are worth mentioning. The one is tied to it. It's the general lack of detailed of information at this time. And she mentions that the ISA, HAZOPS, and PHA weren't available for our

review since SHINE has decided to delay submitting those until operating license time. There's an increased potential for significant design changes after construction. We've talked about that a few times. We'll certainly be looking for their documentation for how they selected and defined DBAs and what was the process to consider uncertainties?

She mentions that the PSAR stated that analyses were completed with codes that have not been validated. We talked about that some. We'll be looking for that later. She asked why are common cause failures ignored? For example, why wouldn't there be multiple failures of the TOGS to recombine the hydrogen produced in the TSV? And that the TSVs fabricated from zirc-4. SHINE is relying on testing being performed at Oak Ridge to provide guidance about fabrication process. Hydrogen uptake corrosion and irradiation performance. We'll also be interested in that.

And then she has a fairly long section I won't read on acceptable margin. And I think you began to cover some of those areas now on the red oil. She asked about those. And I think that's all that. We really have to put those forward.

And then John Stetkar responded to seeing the SHINE aircraft crash paper, and had a few comments. He had a lot of comments, but maybe the one that looks forward is he mentions he's not familiar with conditional probabilities for concrete wall penetration. However, considering the vintage; they came from 1974, it might be suspect. The staff has access to much more detailed assessments that were performed after 9/11 in support of subsequent aircraft impact analyses, so he doesn't think that we ought to hang too much on figure 1 from the paper and there might be better work on that. So we'll be interested in that later on.

MR. LYNCH: Just to clarify --

(Simultaneous speaking.)

CHAIRMAN BLEY: Essentially all aspects of the aircraft crash we might want to look at in more detail later.

I'm sorry. Go ahead, Steve.

MR. LYNCH: I just want to clarify. That was a staff submission, not a SHINE submission.

CHAIRMAN BLEY: I'm sorry. I quote from him.

MR. LYNCH: I just don't anybody to be confused. That's fine.

CHAIRMAN BLEY: Or I misread it, yes. So it didn't come from you guys.

(Laughter.)

MR. LYNCH: But I understand.

CHAIRMAN BLEY: Okay. I guess at this time if there's no more questions from the Committee -- we do want to go into a closed session to address some of these other issues, but before that I want to get the public phone line open.

MR. LYNCH: We still have one more presentation.

CHAIRMAN BLEY: Oh, I'm falling asleep. You guys are right. No, no, we want to hear from you. And since it was almost break time, I was losing track of things. We'll hear from you after the break. We're going to recess for 15 minutes and come back at 2:45.

(Whereupon, the above-entitled matter went off the record at 2:28 p.m. and resumed at 2:45 p.m.)

CHAIRMAN BLEY: We're back in session.

Steve, we'll give it back to you. I apologize for almost forgetting about you guys.

MR. LYNCH: No problem. I mean, if -- we've got plenty more to talk about, or we can go home. All right.

CHAIRMAN BLEY: Pay me now or pay me later.

MR. LYNCH: Yes. All you have to do is click, or I can click it.

MR. MORRISSEY: All right. Hello. My name is Kevin Morrissey. I'm from NMSS, fuel cycle. I'm the ISA reviewer. I also am a project manager on MOX and a criticality reviewer.

MR. HAMMELMAN: I'm Jim Hammelman from NMSS. I'm a chemical safety reviewer.

MR. MORRISSEY: And this presentation is about the radioisotope production facility accident analysis staff review. I'm the presenter today, but the review of the accident analysis is in fact an ISA review, and it's a technical review of all of the different technical reviewers, the crit guy, the rad guy, the chem guy, and the fire guy.

The technical reviewers, in fact -- and you've heard through some of this other stuff, they do the review of the programmatic aspect from the license of the rad program and the crit program and

the fire program and the chem program, and they also -- they are responsible for actually the specific accident sequences based on the consequences. The rad guy does the rad consequence, et cetera, et cetera.

This review is -- as you probably know, this is sort of a regulatory hybrid, because some part of this facility looks like a Part 70 processing facility, and the other part is a reactor. It is being licensed under Part 50, and the -- in fact, in Part 70, there are no provisions for issuing a construction authorization other than for a plutonium facility. MOX, in fact, was done under a two-part license, but basically everybody else is done in one part, which requires that they provide a full application in all its splendor and detail.

So the idea that it's done -- the review is actually done under a Part 70 thing. The guidelines under Part 70 are not well defined for construction authorizations, and under Part 50, as you've seen, this slide actually I took from the 13A people, who are basically doing the same thing. They are meeting the Part 50 requirement. So 50.34 has the contents, and 50.35 talks about the

issuance of construction permits.

The guidance for this thing is 15.37, and there is an ISG. And if you look at 15.37 and ISG, it kind of looks a lot like 15.20, although it's basically 15.20 with a 15.37 cover.

The purpose of the review. The purpose is to provide reasonable assurance that the proposed design of the SHINE facility has incorporated adequate capabilities and features to prevent or mitigate potential accidents and protect the health and safety of the public and workers.

So, basically, the accident analysis is going to be the way that, you know, if you're going to protect the health and safety of the public and the workers, this is what you are protecting them against. Right? So the accident analysis defines the facility hazards that need to be protected against and support the establishment of the design basis.

You know, there is a definite coordination between Chapter 3 on design basis and the chapter on engineering safeguards and the chapter on accident analysis and the chapter on tech specs. You know, hopefully they are all connected in a meaningful semi-consistent way.

Right? They'll tell you that.

The staff performed their review of the technical information presented in Section 13B, and basically the evaluation, in terms of the -- what the staff was trying to accomplish was reviewing that the ISA came, and team makeup used to perform the accident analysis was satisfactory.

In that stage, in fact, when we talk about the team and stuff, we begin to look at the qualifications that they need, you know. It is nice when you bring in the heavyweights and you do the initial analysis.

But one of the parts of the ISA that is important is maintaining the ISA through the facility when you are making changes to the design, when things happen, and so keeping the ISA up to date and making sure you have the right team to continue to maintain and evaluate the ISA is also an important, you know, consideration of our review.

The hazard evaluation process used to identify credible hazards and support the finding of the design basis of the facility, I think somebody asked the question before about credibility, and it's like there's a definition of

credibility. It is actually in 15.20, and they talked about that, and it has the three pieces.

Well, every single licensee copies that definition down, because they are required to provide the definition of "credibility." So, of course, they take out -- the one out of the guidance, and they paste it into their application.

One of the problems we have had with credibility is sometimes people assume that things are incredible because of certain features of the facility. Well, that can't happen because it's process held, or that can't happen because we have a building around it, or that can't happen for some reason.

So I think somebody mentioned earlier that -- I think it was Dr. Tripp here, our criticality guy, when he talked about, you know, some of the problems we have had in terms of credibilities. They are required to look at all credible hazards, you know, not just the worst crit and the worst chem and stuff. They need to look at all credible hazards and the ones that meet performance criteria.

So, you know, even though you can bound some of these accidents and present them, which

they did, and you can even group them together -- you know, we have a definition called the standard type, which means it's the same accident with the same controls, but it may be initiated in a lot of different ways.

You know, a good example of that would be like a crit analysis where you have moderated control. There is a lot of different ways moderated can get -- you know, initiate an event. So it's -- you don't need -- if you have 100 different ways to initiate a vent, you don't need 100 accident sequences. You can describe it in one, which is called the general type of accident sequence, and I believe in general that is what SHINE has done.

We also look at the ISA methodology used to create accident sequences, estimate likelihoods and consequences, designate possible controls, and estimate the risk to workers in public. This is a risk-informed performance-based, you know, method. So the determination of likelihoods and consequences, which are the two basic factors in risk, are a requirement.

The identification and analysis of possible credible accident scenarios is complete.

You know, the facility itself, in terms of the part that our staff looked at, is fairly similar to other fuel cycle facilities. We had similar chemical processes. We have similar criticality concerns. We have similar fire.

They do have a larger, clearly, radiological component, which other fuel facilities don't have. So that part is a little different for us, but is also part of our analysis.

And the last piece is the identification of safety controls. You know, in Part 70 we call them IROFs. Here we are going to call them SSCs. So, and I think there was a question before which talked about defining, you know, safety-related.

And they define safety-related and basically have committed that all things which used to be called IROF have been scratched out now and are called SSCs, and these are all safety-related components which will be included as part of the tech specs.

This is just some of the processes. I don't know how familiar you are with the facility. These are the processes that were reviewed and are separated from the reactor part of the facility.

MR. LYNCH: Just one thing. We are at -- the irradiation events. They are not reactors. I just wanted to clarify that.

MR. MORRISSEY: Oh, okay. Right. I said the word "reactors." You're right.

MR. LYNCH: Just wanted to clarify.

MR. MORRISSEY: Critical irradiators. And I'm a reactor physicist, so that hurts.

CHAIRMAN BLEY: So, Kevin, when you reviewed --

MR. MORRISSEY: Right.

CHAIRMAN BLEY: -- did you look for ways in which this could become a reactor?

MR. MORRISSEY: Well, you know, initially, I think there was a proposal where their K-effective was so close to critical that it might have smelled slightly like a reactor.

CHAIRMAN BLEY: If we need to, we can move this kind of discussion to the closed session. I don't know if we'll get there, but --

MR. MORRISSEY: Well, that --

CHAIRMAN BLEY: -- I think as long as we don't use numbers, we're okay.

MR. MORRISSEY: Well, that's -- I will not use the number.

CHAIRMAN BLEY: Okay.

MR. MORRISSEY: Okay? I'll just say small.

CHAIRMAN BLEY: And now it's not so close?

MR. MORRISSEY: Now it's not so close. And they could discuss that now.

MR. LYNCH: And I think with the reactor discussion, a lot of it is more of a philosophy thing with -- I think we are able to apply the appropriate safety considerations we need to. It is mostly how a reactor is defined in our regulations. If you look at it in more academic terms, the term "reactor" is applied to a lot of different types of facilities.

If a reactor wasn't defined the way it is in the regulations, we could probably more loosely call what SHINE had, you know, a reactor. But it's -- we're working on a regulatory --

CHAIRMAN BLEY: You know, we all get that. What I was getting at is, are there ways you can get above K equals one? And did we look hard enough to see if there are --

MR. LYNCH: For the RPF, we can talk about that here, and we can talk about the other

side of the facility at -- when Kevin is done with his presentation.

CHAIRMAN BLEY: Right.

MR. LYNCH: Yes, I understand.

MR. MORRISSEY: Okay. Accident event types. And I think this is basically just a rehash of what they just presented. They looked at the maximum hypothetical accident. You know, and the maximum hypothetical accident is -- I think this is a Part 50 thing, you know. And in Part 70 space, sorry, but we don't look at hypotheticals. We look at rare accidents, and we look at all accidents which meet a certain criteria.

So, I mean, none of the things that, you know, that have the bounding accident is probably nice in terms of your understanding, but we need a list of all accidents which make what we call the performance requirements, which exceed those limits. And as part of the risk demonstration you have to demonstrate acceptable risk for everybody who reaches that threshold, for every accident.

They looked at external events, and then the usual, the chem, the rad, the crit, and the fire. And I think they showed you this, the

maximum hypothetical accident. It's all radiological accidents. And I don't have the good picture that they had, but it's a picture of the -- it would the picture of the noble gas removal system in those five tanks.

External events. They looked at seismic, high winds, and aircraft impacts. And as you saw in their slide, they assume that these had no consequences. And so the details of that I guess we haven't really dug into in detail, because it's still -- I guess it's still at a fairly high level. So this is -- would be on our list.

When actually the application for the operation for the facility comes in, we will actually look at what are called vertical slices. We will take certain events, and we will go up to the site and dig down through the documentation, through the drawings, through the -- look at the assumptions. All that information wasn't provided to us, you know, in this preliminary level.

And chem accidents. They looked at tank vessels and failures, inside and outside vaults, exothermic reactions, and handling notes. And this is really just to be a general discussion about the types of things they looked at.

Radiological accidents, loss of containment, you know, in tanks and vessels and pipes, overfills, mishandling of equipment, and equipment malfunctions.

Criticality accidents -- and I think they talked about criticality, but they really never talked about the accidents in general, you know, changes in geometry, concentration, transfers, blockage of lines, loss of power, dilution, and fires, you know, and the different types of fires that they looked at, even down to, you know, lightning and leaks and fires in other areas.

So the accident analysis review -- so we reviewed the postulated accident scenarios that are represented of the range of events that are possible in the facility. We looked at the safety systems and defense-in-depth features of the design, provided for the accident sequences, and reviewed the design features, the different prevention and mitigation of potential accidents.

And a lot of these design features are really the stuff that is being constructed, the building, the process cells, the major components within the facility.

We talked about the fact that, you know, the level of detail -- and it's like, you know, I've been at the NRC for 11 years, and every review we have ever done, the staff has really good arguments about the level of detail that is required, you know, to do, you know, this type of review.

And this one is challenging, I said, because, you know, from our point of view, we are used to the full application. And so now, you know, and I'm going to give you a couple examples of, you know, the level of review that we received in our application. So, you know, these are the things that we are pushing forward.

You know, I read -- when I first came here, I read the MOX Carr review, and I thought while they didn't really draw a nice clean line and say, "You've met this line. Here you go, have this construction authorization." It said, "Oh, and we need to do this, and we need to do this, and we need to do this, and we need to do this." And I thought it was rather strange that they pushed all this stuff.

You know, our review is such that we push everything forward. And now I'm reading a

slide that talks about everything we are going to push forward.

And you saw this this morning, too, in the criticality case where we talked about all of the conditions now they want to impose upon the construction authorization. And I don't think that really just applies to criticality. I think that really applies to -- really, to all the accident analysis. And I thought some of the -- and I hadn't seen this stuff before, but I thought some of the suggestions were good.

So the licensee needs to demonstrate that all accident sequences meet performance requirements, and the emphasis here would be on "all." The licensee needs to provide detail on safety controls and their safety functions needed for demonstration of acceptable risk for all accidents.

You know, some of -- and I'm going to give you examples. You will see some of the high level of detail and -- you know, because I can say, "Well, I'm going to control criticality, and I'm going to have moderator control and concentration control."

And you go, "Wow, that sounds really

good." And then, but how do you do that, right? If it was administrative controls, you know, what -- what is the mechanism for doing that? And how would you meet the standard of "highly unlikely"?

The licensee needs to provide detailed likelihoods and consequences for all accident sequences.

Initially, we didn't receive very much accident sequence detail at all. We asked RAIs, and we got tables, and I'll show the example of something from those tables. You know, as a mathematician, I like numbers. And there were no -- there is really not a lot of numbers on the table.

We talk about preventers and mitigators, and the formal demonstration of highly unlikely requires, you know, the detail, something that looks like a number. The licensee needs to provide detailed information on the management measures needed to support availability and reliability safety controls.

One of the things in Part 70 is -- and the example that always comes to my mind is my automobile. When I buy a brand-new automobile, I understand it is a variability and reliability. It

is probably going to start almost 100 percent of the time. When I hit the brakes, it is going to work.

Some of you -- and I have had this a few times, have had a 15-year old car. And, you know, 15 years later, is this control as available and reliable as the day it was new? You know, they are making certain assumptions about reliability, likelihood, and failures, and it's like they need to review and maintain their ISA.

And we talked about this before -- the importance of maintaining your ISA, and that just doesn't mean doing maintenance on components, it means reviewing your analysis and determining whether or not your assumptions are still valid.

Licensees need to provide the expected content of technical specifications for safety-related controls and detailed technical specification data. I find this one interesting, because I have never seen a fuel facility decide what the heck you put in tech specs. I remember tech specs, because I was a reactor guy for 20-something years. And it had a lot of numbers, and it had a lot of data, and it had a lot of stuff.

If I open an ISA, I see a list of safety controls. So I'm curious -- and we have regulations on the content of tech specs. But in this hybrid, you know, regulatory model is, what information are we going to stick in tech specs? And I think this is going to be an interesting part of Step 2, you know, of this journey.

The licensee needs to provide specific human actions versus the generic actions credited in the PSAR to prevent unmitigated accidents. You know, lots of times they specify programs. You know, the safety control will be the crit program or the conduct of operations program. Obviously, the details will be there.

I think one of the questions we had asked them -- and I think somebody has asked -- you had asked a similar question about that long list of questions about initiating events, one of the staff I think already had questions -- what about human failures, you know? A lot of these processes in fuel cycle facilities are done with a lot of administrative controls.

I think they probably have the lowest likelihood -- you know, the highest likelihood? The highest likelihood in terms of failure. And

our review didn't really seem to see that that area was well addressed.

The evaluation findings and conclusions. The applicant has provided reasonable information on the performance and methodology used to evaluate accidents. The applicant has proposed and analyzed a set of accidents that should be representative of the possible range of events that may happen in the areas of the SHINE facility that we review.

The analyzed set of accidents provides insights into the types and number of safety systems and safety features needed for a facility which rarely define the define basis. The potential accidents may be prevented or mitigated by administrative controls, engineering controls, and trained personnel actions.

The staff concludes that the proposed preliminary analysis and the preliminary safety design, including the engineered safety features, should, with reasonable assurance, protect the health and safety of the workers.

MEMBER SKILLMAN: I've got to ask you.

MR. MORRISSEY: Okay.

MEMBER SKILLMAN: I'm fundamentally

confused.

MR. MORRISSEY: Okay.

MEMBER SKILLMAN: On your Slide 14, six bullets.

MR. MORRISSEY: Okay.

MEMBER SKILLMAN: As you well pointed out, it seems odd that these are pushed forward to the operating FSAR. But then, on Slide 15, you present four bullets that say, "Hey, everything is okay. Let's proceed with construction." What is the assurance that the six bullets on 14 do not have elements that are going to backfire if a construction permit is issued?

MR. MORRISSEY: I don't think there are any absolute guarantees that everything -- you know, the standard for determining, you know, that a construction permit is a standard based on reasonable assurance.

The level of detail isn't available to make detailed conclusions, yet for the operating license you need a detailed conclusion to protect -- you know, what is more important, the construction or the operation? You know, because the health and safety of the public isn't really affected by the construction of a facility.

MEMBER SKILLMAN: Really.

MR. MORRISSEY: Directly. You know, the fact that there is no material, and, you know, some of the design changes or some of the changes that would have to be made. I was just at a facility last week doing an inspection, and they were evaluating in total the impact on all the changes made in the construction of the building. I forget how many thousands of changes.

It took 24 man-years of analysis, about a billion ANSYS runs and stuff, too, for them to evaluate the -- you know, the composite impact of making all of the changes they made to the building, which meant, you know, things they screwed up, things that -- where codes and standards couldn't be met, things where they had to deviate because of, you know, certain physical properties. You know, that is part of the onus that is on the licensee.

What assurance can we give them that they won't -- wouldn't change a design? We approve a design or a concept of a design.

MR. LYNCH: And I think for the purposes what -- the assurance we were looking for for issuing a construction permit was assurance

that they had appropriate methodologies to address these issues. And that's why we asked for representative samples of their accident analysis methodologies for the different types of accidents.

We recognize that they wouldn't be able to give us -- you know, demonstrate that all accident sequences meet performance requirements. So we asked them to give us a representative sample, so that we could look at their methodology and see how you're going to apply that going forward. So they convinced us that they had the appropriate methodology to address these issues later.

MEMBER SKILLMAN: Thank you.

MR. MORRISSEY: And I guess that's it. And I did want to just show you, as an example -- and this one is actually red oil, and this comes from the -- directly out of their response to a question, which is please provide us, you know, actions in containment.

And the purpose of this is really to show you the level of detail in this phase compared to the level of detail we will expect for the operating phase, you know. And this is actually the red oil one, and the controls to prevent the

thing where the solvent residence time and the conduct of operations program. Good enough for you, you know? It's like you make the call. It's not like -- like football, and you make the call, you know, they have a lot of --

MEMBER POWERS: But in the spirit that they have the right kind of process, procedures for --

MR. MORRISSEY: That's right. They have the tools -- and that's what he basically just said. They have the tools to evaluate in detail the acceptability performance.

MEMBER POWERS: Wouldn't they yield to the most recent pronouncements by an esteemed regulatory body on this subject? I mean, they're aware of what it said. They seem to have an appreciation of what kinds of thinking go into this. And so, I mean, the detailed questions of, what residence times are and how they accumulate, and things like that, it's not surprising they can't answer that question.

MR. MORRISSEY: Right.

MEMBER POWERS: But, I mean, it seems to me, I mean, they seem to recognize the ventilation standard came from experiments by Vousk

& Associates that -- they do not know all the details and the dirty secrets that the community has on this subject, because they are kind of new, but they will learn real quickly.

It seems for this particular case that it's adequate. The questions that I think Dick is raising and that I raised, the facility does not seem to, as part of its design, said what if I'm shut down for six months? A year? Or two? And when I think about what we did on the MOX facility, we spent at least one entire subcommittee meeting just talking on that subject.

MR. MORRISSEY: Right.

MEMBER POWERS: Because I thought I -- in their -- the reason for that of course was that particular facility had an enormous potential of being laid up for months at a time.

MR. MORRISSEY: Right.

MEMBER POWERS: So they had to -- but I don't think this facility is immune from that either, and dictum in concrete before --

MR. MORRISSEY: Right.

MEMBER POWERS: -- which you have to subsequently chip out.

MR. MORRISSEY: I think that's a good

point, because you're asking the ISA in their SECY analysis for them to evaluate, you know, maintenance and shutdowns and other stuff. But it's like, well, what if you -- you know, you have to shut down the facility for three years.

MEMBER POWERS: And there's no reason to think that, I mean --

MR. MORRISSEY: Once again, they could have -- do they have the right tools to do the evaluation and maintain the safety of the company? Yes. I think they do.

MEMBER POWERS: I mean, I think this is not an impossible event, to be shut down for a year or two.

MR. MORRISSEY: Okay. Any other questions?

MR. LYNCH: If you want, I could say a word or two about criticality on the other side to follow up. I just didn't want -- I wanted to let Kevin finish his --

MR. MORRISSEY: Okay. And, you know, one of the reasons we brought Jim up here was because the rad guy and the crit guy and the fire guy all have individual chapters in the PSAR. But the chem program doesn't. So Jim's evaluation of

the chem program is actually -- will end up in an SER, which is in my section at the moment.

So, but if anybody certainly had any questions relative to the chem program, now would be the time because Jim forgot. And he knows the red oil stuff, too, which is -- always bring your red oil guy with you.

MR. HAMMELMAN: And can explosions, I was --

MEMBER POWERS: Well, can you talk about this analog to hydroxylamine that they are using?

MR. HAMMELMAN: No, I can't. I did -- when I was doing a preliminary review of their process, I was looking up all of the process chemicals. I did not -- you know, just doing a quick Google search, I did not see any red flags at that stage, but I was sort of -- I want to see when they get their flow sheets spelled out a little bit more clearly, the equipment sizes.

You know, I'm not -- let me back up a little bit. You know, it looks to me like they have identified the rock hazards. They have the pieces of equipment -- let's put it -- the structures that are put in place right now, the

shielding, the ventilation, those are the right ones I think to protect -- or let's put it this way, to mitigate the kind of hazards that they are going to have in their facility. But they still have to go through this detailed design and detailed analysis.

As they do that, I expect the hazard analysis to go down to a more detailed level, and then at that point in time they will understand the specific process parameters that they need to control to prevent accidents, so they don't rely as much on the shielding, on the ventilation system.

MEMBER POWERS: The UREX process was developed within the DOE communication.

MR. HAMMELMAN: Yes, AEC.

MEMBER POWERS: And they clearly have done some sort of safety analysis on it. Do we have access to that?

MR. HAMMELMAN: I suspect we do. I haven't seen that yet. But I have seen -- I have seen some safety analysis of, you know, bench scale facilities, but I haven't seen a full rigorous one yet.

MEMBER POWERS: Has DOE ever developed this UREX -- applied this UREX process on the scale

that they are thinking of doing this?

MR. HAMMELMAN: Smaller scale is my understanding. Everything I have seen is smaller scale.

MEMBER POWERS: My impression is it's all much smaller scale.

MR. HAMMELMAN: Yes, yes. And then you mix your settlers. These are not tremendously larger, but it's all -- this is all shielded laboratory scale.

MEMBER POWERS: It would be interesting to see what a DOE facility, even though it's a smaller scale, thought about some of these things.

MR. HAMMELMAN: Yes. There's the red oil, and then they've also got -- there's another organic ion exchanger that has to be checked out a little bit more, too.

MEMBER POWERS: I mean, I think we saw -- I mean, at least I came away from the MOX review saying that, for that facility, the hydroxylamine was a bigger issue than the red oil issue, because they had focused an enormous amount of talent on the red oil issue.

I mean, they bought this kid in from -- kid -- this scientist in from -- everybody is a kid

to me. From France who seemed to -- to know more about red oil than I would know if I stayed up nights studying. But the hydroxylamine has been a headache for everybody.

Now, we've got a new material, and of course we found out hydroxylamine was a problem not from the laboratory but from using it in big facilities. And we don't have that same experiential base on this other material. So you're --

MR. HAMMELMAN: Yes. And the -- and I think -- with the UREX, I think -- my recall fades with time, but, you know, you are not doing the plutonium recovery. You know, it's very different than in PUREX where you're shoving fission products. And I'm pretty sure the -- you know, the plutonium ends up going out in the -- in your --

MEMBER POWERS: Well, we've taken that separately, because we are --

MR. HAMMELMAN: Right.

MEMBER POWERS: -- doing the products in a line.

MR. HAMMELMAN: Right. In the PUREX process, that is right.

MEMBER POWERS: The stuff these guys

want to recover is a pain in the ass.

(Laughter.)

MR. HAMMELMAN: Absolutely.

CHAIRMAN BLEY: Are there any other chemical issues that you see as needing a lot more work as we go toward the operating license phase?

MR. HAMMELMAN: The two I have -- I have not seen, but will continue to check, the toxicity issues. But I have worried more about the energetic, the explosive type of things. The only two that I have seen, and been through it several times, is this red oil potential in a couple locations where you've got some evaporators, you've got heat, nitric acid, potential of organic. You've got to look close there. And the other place is there is an organic ion exchanger --

MEMBER POWERS: That was the bottoms in the evaporator.

MR. HAMMELMAN: Yes.

MEMBER POWERS: Because that's where you accumulate.

MR. HAMMELMAN: And there is an organic ion exchanger, and there is a history there that if you don't treat those right they can explode, too.

CHAIRMAN BLEY: Okay. Anything else

from the Committee?

Well, before we go into closed session, just a couple of things. We'll entertain comments from the public on the phone line and in the room, but I also want to point out toward the end of the session we are going to be talking some about preparation for the full Committee meeting.

MR. COSTEDIO: This is Jim Costedio. We have some ACRS questions from the last meeting that --

CHAIRMAN BLEY: Oh.

MR. COSTEDIO: -- and that we want to respond to.

CHAIRMAN BLEY: That's perfect. Let's do that now.

MR. COSTEDIO: I don't know if this would be a good time to --

CHAIRMAN BLEY: Let's do that now before we have public comments. Yes.

MS. KOLB: I'm Catherine Kolb. I'll be -- I just have a couple of slides on responses to questions from the previous ACRS Subcommittee meeting.

Next slide?

The first concerns there was a

discussion about the rain on snow loading in the design of the facility. SHINE facility is designed per ASCE 7-05 for determining rain on snow surcharge loading. SHINE is located in a 25-pound-per-square-foot snow region per Figure 71 in that standard.

Per the standard, a rain on snow surcharge load of five pounds is required only for locations where the snow loading is 20 pounds per square foot or less but not zero per the inspection.

However, in our design we used a snow load -- standard snow load of 30 pounds per square foot for the structural design for conservatism because in the graph the dividing line between the two sections is in the same county slightly north of our facility, so we use that for conservatism.

I also want to note that the -- this is two feet thick and there are other -- we used other methods in the standard for determining the rain or, I mean, for regular snow loading.

Next slide?

We also had some questions about the design of our facility and the layouts where the TRPS, the control system, and RICS for the RPF

control system trains were located in the same room. There were questions about the fire hazard analysis for this.

For the SHINE system, structures, and components, our design, in accordance with IEEE Standard 384-2008, the standard criteria for independence of Class 1E equipment and circuits, SHINE performed the following per our detailed design. We will evaluate the locations of the TRPS and RICS components with respect to the fire area destinations, and we will ensure that the electrical and control system train separation, including the consideration of fire hazards, is performed in accordance with the applicable IEEE and NFPA standards.

We have captured this in our corrective action program for inclusion into detailed design.

CHAIRMAN BLEY: Good. Thank you.

MS. KOLB: You have one more? We had a discussion about the amount of detail contained in the PSAR for some systems, especially some non-safety-related systems, and how we determine their classification, to try and determine that those systems designated as non-safety-related in PSAR Table 3.5-1 will be designed such that their

operation or failure will not have an adverse impact on any safety function.

The classifications of systems we verify during detailed design when official documentation is developed and when the final safety analysis is completed. Additional details for the non-safety-related systems we placed in the FSAR based on the final design.

CHAIRMAN BLEY: You have a corrective action program in place now.

MS. KOLB: We do.

CHAIRMAN BLEY: And will that same program transition as you go to operations?

MS. KOLB: We will.

CHAIRMAN BLEY: Okay. So it will be a continuing process. Excellent.

MEMBER SCHULTZ: Catherine, at the end of the discussion that you had today on snow loading, you seemed to indicate that that's not the limiting criteria that has determined the thickness of the roof, the strength of the structure, is that true? Because the discussion on snow loading still is a bit confusing to me, so I'm hoping that that's not really the determining factor.

MS. KOLB: I don't believe it is the

determining factor. It was used in a combination. There are different load combinations used in the structural design where you use like 75 percent of the load, of this -- from snow and some from live loads and some dead loads, and different combinations for determining the structural.

So it is one of the factors. It's not the limiting factor. This slide was clarification on how we were applying the rain on snow surcharge specifically, which isn't required for our facility, but we included an equivalence extra of five pounds because of the location of the facility.

MEMBER SCHULTZ: Okay. Five pounds isn't a lot of rain. So I'm not criticizing the standard, but I am thinking more in terms of the beyond design basis event. I'm sorry. We've been focusing on that for Fukushima for quite some time.

And if I knew that a county or, you know, somewhere nearby was five pounds more, you know, or five inches more than where I was, I would certainly assume that right from the outset.

MS. KOLB: Which is what we did. We assumed the next higher -- I mean, the ranges are several counties --

MEMBER SCHULTZ: Appreciate that piece, but I really wanted to focus on your final comment, which was this is not the defining aspect of the structural strength associated with the roof load. Is that right, Eric or Bill?

MR. HENNESSY: Yes, that's correct. This was just to clarify the --

MEMBER SCHULTZ: That's fine.

MR. HENNESSY: -- specific question, but, no, that's not the --

MEMBER SCHULTZ: Good.

MR. HENNESSY: We've got airplane crashes and seismic events in that area.

MEMBER SCHULTZ: Yes, I thought so. That's a good clarification, then, thank you.

CHAIRMAN BLEY: Thank you very much.

MR. LYNCH: We had a few issues that --

CHAIRMAN BLEY: Same kind of thing. Come on up.

MR. LYNCH: I'll speak quickly.

CHAIRMAN BLEY: We're right on track, or almost. Actually, we're ahead a little bit.

MR. LYNCH: I am just going to run through some of the problem items that we have identified. One overall thing to let you know

where we're at, so we have provided you draft chapters of the SER. We're working on polishing those up, but we haven't had any substantial content changes. But it looks a lot more attractive now than it did when you saw it. So we plan on cleaning that up and getting that to you very, very soon.

CHAIRMAN BLEY: Which means roughly?

MR. LYNCH: It should be early next week.

CHAIRMAN BLEY: Okay. That's good, because the week after is our full Committee meeting.

MR. LYNCH: Yes. Yes, absolutely.

CHAIRMAN BLEY: And you'll want a letter at that time.

MR. LYNCH: Yes. As we get that cleaned up, that's a -- starting tomorrow, that is my primary focus. It has gone through legal review now, and we've got some additional polishes from our subject matter experts based on our meetings here.

MEMBER SCHULTZ: Steve, we discussed today some recent RAI responses that have come in that -- are those being --

MR. LYNCH: Yes.

MEMBER SCHULTZ: -- incorporated in the SE?

MR. LYNCH: Yes. No, absolutely. For the recent RAI responses that came in, they -- those had to do with our discussion in Chapter 6, and those are in the safety evaluation currently, and I will provide those responses to you as well.

MEMBER SCHULTZ: Thank you.

MR. LYNCH: So to just kind of run through this, so we also -- we also looked at the rain/snow load again to verify that we were satisfied with SHINE's analysis. And based on the assigned importance factor of 1.2 to that and the 100-year interval of recurrence, we -- and looking at the ASCE standard, we decided that their assumption of the 30 PSF for the ground snow load was adequate for the design of their facility.

Regarding the designation of the RVZ3 as non-safety-related, and why that is adequate for construction, we looked at that again and essentially we saw that for the RBZ3 it only consists of airlocks that control access to the RCA. And since these airlocks are contained within a seismic category of one structure, which is the

RCA, and that they don't rely on any powered systems in order to operate, for the purposes of construction we were comfortable with these remaining non-safety-related, and will reevaluate this designation come the operating license.

As far as the treatment of defense-in-depth, design against common mode failures, this -- I think this was -- we misspoke when we were -- this had to do with the conversation on relying on independence to prevent common mode failures. Going back through our evaluation and looking at the PSAR, yes, independence is a factor, but we are also looking at redundancy and diversity as well as part of that.

So I think that was -- that's a clarification I wanted to make, that we understand the importance of diversity and maintaining -- and defense-in-depth.

As far as the designation of an FVZ4 as non-safety-related for construction -- and this had to do with making sure that there was adequate cooling in the control room in the event that you lost your HVAC and how this could impact your digital control systems in the control room.

We have an RAI on this subject. This was RAI 3.5-7, and we had similar concerns. And in this SHINE committed to a number of different safety-related cooling systems that would be designed specifically for the control room to maintain cooling and/or heating during an event.

And one of the design requirements that they have with control room is that any equipment that could be affected by excess heating or cooling, that they would make sure that the equipment could withstand until they could get sufficient time to allow portable heating or cooling to be installed, if necessary. And these are all items in this list that we are going to verify at the operating stage.

CHAIRMAN BLEY: That sounds good. When you give us the revised SER, there will be a -- I'm wondering if you'll have two lists, one list of conditions on a construction permit, and a second list of any other commitments that they have made.

MR. LYNCH: Yes. Those are both --

CHAIRMAN BLEY: Be clear --

MR. LYNCH: You'll have two tables that will be there.

CHAIRMAN BLEY: Perfect.

MR. LYNCH: And, let's see, then another clarification on the evaluation of the facility child water supply and distribution system. I want to confirm that, yes, this was evaluated, and how it's used in the facilities that supports non-safety-related systems, essentially it's used as heat transfer for the RPCS, which is also non-safety-related.

It is not needed for emergency cooling, for any sort of emergency cooling. The pools for the irradiation units are sufficient. And as far as this -- the chilled water support of the RCA ventilation system, that ventilation system is not needed for cooling. That is only -- its only real function that is needed is for confinement.

Looking at the fire evaluation areas, that is something that is still what we are looking at ongoing. At the operating license stage, we will look at the safe shutdown analysis for this. We do note that there is some separation of the ESFAS A and B in different fire areas with the marshaling cabinets for different control systems in separate areas.

But this is something that we are going to continue to be looking at with the operating

license. But we do recognize that it is important -- it is important.

For the irradiation facility accident analysis, there were a few comments we had on this. There were some errors that you had identified in our SER that have been corrected. One of those was that we misspoke, saying that the TOC circulated nitrogen gas, which it doesn't. That was corrected.

The other issue that was brought up was talking about the primary system boundary being able to -- the pressures it could withstand. SHINE assumed in its calculations conservatorily -- conservatively the PSB being able to withstand 200 psi from -- and how this relates to deflagrations. The maximum that they anticipated from that event would only be 50 psi. And we are going to continue to evaluate the pressure design at the FSAR stage.

One thing I wanted to note here as far as review methodology -- and some of the -- because we've had -- we've got some comments on kind of the global analysis of the facility, and this is one thing -- looking back, now that we have gone through all the chapters, and I think what could have better communicated this would have been a

better organization of how we presented chapters, because I think, as we heard today, there is a lot of systems that we look at that were overarching for the entire facility.

Radiation protection is something that we look at globally. Quality assurance is a global issue. Same with emergency planning and operations, accident analysis, even though we break that into -- you know, there's different types of accident events.

And even though we're kind of merging two different accident analysis methodologies, we still looked at the entire facility and how a bounding accident -- we looked at different types of accidents in each facility, but still identified a single maximum event that -- for the entire facility that was bounding.

And we looked at, you know -- there's a cooling system that we looked at for the entire facility. The confinement, you know, as far as engineered safety features go, is for the entire facility. Looked at structures, systems, and components. We had a single definition for the entire facility of what is safety-related.

So there's a lot that is overarching,

and I think if we were to do this all again, talk about all of that first, and then talk about the specifics in between, but I do -- I did want to emphasize that we did a comprehensive review looking at the entire facility and what was interconnected between those. And as far as how we're going forward with our review, we do have that list of regulatory commitments recognizing that there is outgoing development of the design of the facility.

For those areas that we would like to confirm some of SHINE's design as it is getting constructed, we are imposing permit conditions, to request that SHINE provide us additional information as they are constructing.

As far as design changes go, that was another thing that came up, and how we're going to look at that. There will be a change control process if they need to make changes during their -- during construction, it may be similar to 50.59. There will also be instances where they will need an amendment to their permit in order to make certain changes.

A good example of that would be right now in the radioisotope production facility SHINE

has committed to .06 margin of subcriticality. If they want to change that, that's going to need to be an amendment, and we will look at that.

As follow up from today, I just wanted to highlight a few items that we have identified that we need to come back to with more information on, and we will put that in writing for you. And, also, any additional information that SHINE submits on the docket in response to anything you've brought up today, we will also provide that to you, as long as -- along with the staff's evaluation of those responses. And we will have all of that to you in advance of the full Committee meeting.

But the items that came up that I'll make sure we get back to you on from the staff's perspective explicitly are taking a closer look at the need for extended lay-up provisions. We are going to take a closer look at that. Going to make sure that you have our most recent RAIs related to the criticality safety.

And we are going to have a conversation with SHINE actually, right after this meeting probably, about providing a definition of safety-related activities.

I do think one item I had noted earlier

in the day that I think we cleared up during the quality assurance discussion was how SHINE established their Quality Level 1 and Quality Level 2 designations. But we can still provide more information on that if you are interested.

And then, just to close out, I did -- to follow up on the last presentation, as far as subcritical philosophy and the different sizes of the facility, on the RPF side maintaining subcriticality is more directly tied to safety. That's the whole real that they are maintaining subcriticality, and how Part 70 is written is from a safety point of view.

For the irradiation units, maintaining subcriticality is more of a self-imposed condition by the licensee. In this instance, I don't see their proposed operating subcriticality being any safer than if they were to be operating critical. We do need to be looking at big reactivity insertions, power excursions, things that could result in accidents.

But I don't think inherently going from their normal operating subcriticality to K equals one creates a new -- creates an accident scenario. The staff is still evaluating -- obviously, SHINE

is licensed to operate subcritical. They need to maintain that subcriticality. And the staff is still evaluating how we want to treat, you know, a criticality event should it happen.

At this point in time, SHINE is telling us that a criticality event will not happen, and how we are -- based on the reactivity feedback effects that we are seeing preliminarily, it looks like should the facility go critical, reactivity feedbacks would bring that back subcritical without any action.

CHAIRMAN BLEY: You're probably right. We'll talk about it a little more in closed session. But by saying they're not a reactor, and saying they're going to not become critical, you're avoiding a string of requirements in licensing that we talked about in our first meeting, I think. And that might be part of the issue, but the other part is, is there a safety issue associated with that? And we'll probably talk about more today.

MR. LYNCH: Sure.

CHAIRMAN BLEY: Anything else?

MR. LYNCH: Nope. I think that will do it.

CHAIRMAN BLEY: Anything from the

Committee?

At this time, Maitri, can we get the phone lines open? Or, if you're listening in back there, get our phone lines open, please.

At this point, is there anybody in the audience here in the room who would like to make a comment on the record? If so, please come up to the microphones, identify yourselves, and make your comments. Nobody in the room.

Is there -- I'm not sure if the phone line is open yet. But if somebody is on the line and would like to make a comment, please speak up now. Anyone? Going, going, gone.

At this time, we are going to close the meeting, which means we will have to close the public phone line.

(Whereupon, the above-entitled matter went off the record at 3:45 p.m. and resumed at 4:03 p.m.)

CHAIRMAN BLEY: So at this point, we'll go back to a public transcript.

MR. LYNCH: If you --

CHAIRMAN BLEY: Yes?

MR. LYNCH: If you want to -- we're happy to still be on the public record, because we

want to provide a couple of clarifying statements with respect to SHINE not being designated a reactor, if that's all right.

CHAIRMAN BLEY: That's fine. And I think you want those on the public record.

MR. LYNCH: Yes. There's no reason for it not to be.

CHAIRMAN BLEY: So please go forward.

MR. LYNCH: Sure. I can do it. Actually, can I use the computer?

CHAIRMAN BLEY: Sure.

MR. LYNCH: I want to -- I think that might help illustrate one of my points.

All right. So I think there was some confusion at the initial meeting on how the ISG is written. Let me blow this up.

So I'm just going to -- for example, we'll just call this Chapter 5 on coolant systems. So they wrote their application. They followed the ISG. And the first part of Chapter 5 says, 5A.1, critical DNS reactor. And then it lists -- says what you need to do for that, and essentially it says go back to NUREG-1537 if you're a heterogeneous reactor.

And you get 5A.2, which talks about

everything you need to do with your homogeneous reactor. And here it provides new guidances. That was the main -- the primary purpose of the interim staff guidance was to provide guidance on how to fill out your application if you are a homogeneous reactor, because the original NUREG only dealt with heterogeneous reactors.

And then we have 5B for the RPF. So when China filled out their application, in order to be complete and follow the guidance, when they got to 5A.1, heterogeneous reactor, they basically said, "We don't have one. We're not filling out this part of the application."

Then, they went to 5A.2, and instead of homogeneous reactor they called it what they have, the irradiation facility or irradiation units. And then they followed 5B for the RPF.

Now, as far as requirements for the reactor versus being a subcritical utilization facility, going through Part 50, there is no difference in how we are licensing them as far as the requirements we are imposing on them for reviewing the construction permits.

Essentially, how the regulations have been written, it doesn't distinguish between

criticality levels and impractical standpoints. What we are looking at is in the broadest terms we have utilization facilities and production facilities, and there aren't a whole lot -- the regulations for production facilities are generally lumped in with utilization facilities, since the last operating production facility we had was West Valley, and that ceased operations in 1972.

So the regulations on production facilities haven't evolved all that much since then. But as far as how they have evolved for utilization facilities, our distinctions have been between nuclear power reactors and research reactors and everything else essentially.

So if we're looking at licensing SHINE, the report says nuclear power reactor doesn't apply just like it doesn't apply for all of the research reactors. And everywhere where it just says generic utilization facility, like we've got SHINE licensed as, or will be licensed as, we apply those regulations, and all those same regulations apply to the research reactors where it just talks about generic utilization of facilities.

The difference in the licensing is between the two different classes of licenses that

we have under the Atomic Energy Act. We have Class 103 licensees which are for commercial facilities, and Class 104 licensees which are for research and development facilities.

SHINE is unique in that it is, at least as far as I know, it may be the only non-power utilization facility we have licensed as a commercial facility. All of the existing research reactors we have are all licensed as research facilities.

Again, the main difference in terms of the spirit of the Atomic Energy Act of why those two classes were created was to make sure that we didn't overburden facilities that were going to push forward nuclear development in the country.

In practical respects, in how that looks in the regulations, we don't generally differentiate between Class 103 and Class 104 licensees. It is mostly scaled on a safety perspective, looking at power plants and nuclear power reactors and everyone else again. So there we don't see that granularity.

And going through all the regulations in Part 50, our licensing SHINE looks very, very much like our licensing the research reactors.

MR. ADAMS: Okay. Can I make a comment or two? So we are addressing this. We are working on a rulemaking that, if we are successful, you are going to see a new definition for what we call non-power utilization and production facility.

An interesting quirk with the regulation is that we are here because they have chosen to be a commercial facility. And the definition of "commercial" is more than 50 percent of the cost of operating the facility is devoted to conducting commercial activities. You know, if they were at 49 percent, I'm not sure we would be here, because the regulations require commercial Class 103 facilities to come before you, utilization facilities, production facilities, but not non-commercial ones.

CHAIRMAN BLEY: But we do see some research reactors, too.

MR. ADAMS: The only research and test reactor you see is NIST, because it is a test reactor, and by statute test reactors have to come in front of you, but not non-commercial research reactors. So it's the fact that we are here being driven by the fact that there -- more than 50 percent of the cost of running the facility is

being devoted to commercial activities.

You know, I know why that regulation came into being, and, you know, there is historical reasons for it. But it's -- you know, it's just the interesting quirk of the regulations.

When they were writing the ISG, at that point in time we were looking at a potential application from BMW for liquid homogeneous reactors that, you know, looked a lot like SHINE only instead of a -- you know, instead of the accelerator-based neutrons they actually -- you know, they actually pulled the control rods and it went critical.

So that's -- at that point in time, research helped us write the ISG, which had the homogeneous reactor information in it, which we applied just about completely to SHINE, plus the radioisotope production facilities was also introduced at that time, and that is -- you know, if we come to you with other applications, anybody who is fissioning uranium is going to have, you know, one of these radioisotope production facilities.

So you are going to see that -- you know, you are going to see that again in different

flavors as we move forward. It's not just --

CHAIRMAN BLEY: Okay. Thanks.

MR. SMITH: One question on homogeneous versus heterogeneous. I find that distinction almost absurd in the sense that, what is homogeneous about any of these systems?

MR. ADAMS: Well, so historically there have been two types of homogeneous reactors that we have licensed. You know, the liquid homogeneous reactors where the -- you know, the fuel, the moderator, you know, the system is homogeneous, and also we have licensed about a dozen of those over the years. They have all been decommissioned.

There is also a small research reactor called a AGN-201, which is a solid homogeneous reactor, where there it's -- fuel is mixed in with a polyethylene, so, again, it's a, you know, fuel moderator mixture, which for -- you know, for a nuclear calculation probably -- you know, for doing nuclear calculations looks a lot like -- you know, like your classic homogeneous rector.

So, I mean, that's the -- you know, the two types of reactors we face, and that -- you know, that carries through back to the ANS research and test reactor standards. And, you know, it's

basically what you -- you know, what you saw in the early days of teaching nuclear.

MR. SMITH: But what do you do for reviewing one differently than the other?

MR. ADAMS: Well, the main difference is what Steve pointed out there, that we -- that we developed the ISG to NUREG-1537 for homogeneous reactors. Now, you might say, well, why didn't it exist before? When we wrote NUREG-1537 in 1996, all of the liquid homogeneous reactors were gone. The soft homogeneous reactors were licensed. We saw no homogeneous reactors on the horizon, so we wrote NUREG-1537 for what we thought was on -- you know, was on the horizon.

And so, and BMW came in and said, "Look what we've got." We said, "Uh-oh. We need to go back, and it's time -- you know, it's time to bring homogeneous systems into NUREG-1537."

And, indeed, if you look at NUREG-1537, there are -- you know, there are differences, you know, when you go from, you know, fuel cladding to, you know, the tank, I mean, there are -- you know, there are different safety and design and review considerations.

CHAIRMAN BLEY: Anything else from the

Committee? Thank you. Anymore?

MR. LYNCH: Nope.

CHAIRMAN BLEY: Okay. What I think I want to do is real briefly highlight a couple of things I think you ought to talk about at the full Committee meeting when we do our normal go around the table, get comments from all the members, including any thoughts the members have about things we ought to do at the full Committee meeting. And we will only have two hours or maybe a little more if we need more.

I think we need to at least have -- and this should be real short -- a brief history of the need for a moly-99 facility. A bit on the evolution of the requirements. As they have gone through the ISG, and as we have gotten to this point, and how you came up with what you were doing to license this one, and some of what you did just now would be helpful there.

There was a discussion about what people need to have for the construction permit vice the operating license. And then, a brief overview of the SHINE design and location, Chapter 1, 2, kind of stuff, but consolidated. And then, for me, Chapters 4 and 13 are the big ones we ought

to focus on, but we will hear what other members have to say.

At this point, I think I will start with Steve and come around the table, just general comments and comments about the full Committee meeting.

MEMBER SCHULTZ: General comments about today. I appreciate, once again, the presentations by the SHINE organization as well as the staff. I thought they covered the material on the agenda very well and provided a lot of good information to us related to what we need to consider.

With regard to the full Committee, yes, I think we really do need to have an emphasis related to the construction permit versus the operation, because that is just something that people have forgotten. Needs to be refreshed, particularly for this application.

I think the chapters that you mentioned are appropriate, but I really thought what we covered today and the connection between Chapter 6, as well as -- as well as 3 -- 3, 6, and 13, the safety analysis, are probably the high points to hit.

CHAIRMAN BLEY: Okay. Thanks, Steve.

Dick?

MEMBER SKILLMAN: Yes. First of all, I appreciate the SHINE team's flexibility to come back again and again, but also to respond to the previous meetings. So thank you for that, and thanks to the staff for that very same behavior.

I think what will be important for the full Committee is to know what administrative processes are in place to make sure that once a construction permit is provided that the important issues we have talked about are incorporated. In other words, we didn't leave anyone behind. We really circled back around, and the items that we thought were important to preserve the integrity of the construction phase remain.

Thank you.

CHAIRMAN BLEY: Okay. Thanks.

Dana?

MEMBER POWERS: Well, it strikes me that we're going to have to organize this very carefully. And I would encourage the staff in their opening comments to cover the material on construction permits versus permits at the outset.

I think SHINE is going to have a problem delving into any depth, because they are

going to have to describe an unusual facility, like its location and what all it does, and I think that's going to eat up your time rather completely and leave you the opportunity merely to list down the kinds of things you've thought about as far as safety and mitigation of accidents.

I think you're going to have to give the rest of the Committee enough understanding of what the system looks like that that is going to essentially exhaust the time you have available. And you're just going to have to rely on questions to bring up the detail that we've gone into in the Subcommittee about safety systems and things like that.

I don't think you can anticipate plunging into details just because the time is going to be so compressed for you, and that's just a function of the full Committee. That's why we have Subcommittees, to go into that detail.

And so I would -- my recommendation is to concentrate on what the system does and how it does it and have some backup slides to go into the details on the safety systems and protocols, because I just don't see you having any kind of time to go into that. And if you shortcut the

description of the facility, then you're going to get all kinds of questions based on confusion about what it is actually trying to do and how it is trying to do it.

CHAIRMAN BLEY: Ron?

MEMBER BALLINGER: I agree with Dana. Without having a decent enough description of the system, when you talk about the safety systems, all we are just going to do is get back to the description of the system before -- so you can talk about the safety systems, because they are so tied to the description of the system.

MEMBER POWERS: I think you can probably elude to things, put them down as the last note on a viewgraph, or things like that, but I just don't see how you can go into any kind of detail here at all. And it's okay. It's at the construction phase. And you're -- I guess we're going to do it in kind of a overview-ish fashion. That's all you can do.

MEMBER BALLINGER: I think you might get some additional questions related to the siting issue, proximity to industrial facilities to train track business, because this is going to be -- people are going to look at this and they are going

to say, "Well, this is a reactor." "Well, maybe it's not a reactor, but maybe it could be a reactor."

And so, therefore, you know, if we fly something into this thing, what kind of -- how does the design deal with that?

MEMBER POWERS: Yes. I certainly wouldn't hesitate to point out the geography and siting on that and the thinking about -- like you say, it -- it's a reactor in that it's a source term.

MEMBER BALLINGER: Right. Because there is that airport.

MEMBER SKILLMAN: I think we're going to be in a situation where the members have been so focused on issues pertaining to CPRR, containment pressure and radiation reduction, that type of thinking, that the notion that this facility really doesn't have containment, it has confinement, and confinement is for the release of those five gas tanks.

Now, there is an awful lot more in the safety analysis, but getting that on the table as a way to think about what this facility is all about would be very helpful, because that takes a whole

lot of stuff off the table. We don't have a containment. this is confinement, and it's really this event that kind of drives that design.

I think that would probably move the discussion fairly far down the road, if it's presented carefully.

MEMBER POWERS: Good suggestion, Dick.

CHAIRMAN BLEY: Yes. I think I agree with everybody here. My focus on Chapter 4 and 13 -- Chapter 4 is how the thing works, and I think you have had more than half of us already. So you were only talking to half of us at this full Committee, really, the first time.

But making sure everybody understands what it's for, how it works, how the two pieces tie together, and what the more serious kinds of accidents could be, and what their limits are, is pretty much it, until you get --

MEMBER SCHULTZ: But the other piece --

CHAIRMAN BLEY: -- by question.

MEMBER SCHULTZ: The other piece that is -- will take some thought and preparation, and I think when we got to this piece today it made a lot of sense, but that was the description to the Committee of what is the responsibility of the

applicant and the NRC staff review with regard to safety of the facility for the construction part and what is coming in the operations part, because that general description will not -- is going to be a bit fresh to the Committee, and needs to be stated as it was in the discussion -- some of the parts of the discussion today, whereas it's the responsibility to do the following, not get down into the gory details and understand how the safety of this facility is going to run, and when the PRA is done, and so on and so forth. So --

CHAIRMAN BLEY: That's kind of driving me to where I started. But usually we have this -- the applicant go first, and then the staff. I think if we began with the staff, with an overview of, you know, why we need the moly-99, how you got here in a licensing process, why we're doing -- how we're doing the construction permit, and some -- and more details are going to follow on how that --

MEMBER SCHULTZ: Yes. I would --

CHAIRMAN BLEY: -- first, and then come back to the applicant for the design.

MEMBER POWERS: And, in fact, I would have the staff come up twice. The first one would be that overview. The second one would be the --

would be their conclusions about the application.

CHAIRMAN BLEY: Right. But not -- not replicating what they've heard from --

MEMBER POWERS: Yes. Not --

CHAIRMAN BLEY: -- from them.

MEMBER POWERS: -- not convoluting that overview of the licensing view with the evaluation of this particular application. So you can keep it clean. And so I would say, begin with the staff. And if it's a relatively long presentation for that overview, long compared to what we have done in the past, then the applicant, describing their facility and what their thinking is, and then the staff come in and say, "Here is what we think about what they submitted to us" --

CHAIRMAN BLEY: For the construction permit.

MEMBER POWERS: -- for purposes of the construction permit. That will get around some of this time constraint, which is pretty horrible in this case, just because it's such a unique facility. I mean, they're all unique, but a lot of them we have seen the basic thinking behind them, whereas this is -- this is special.

CHAIRMAN BLEY: But that final one, you

folks -- would be your thinking on it, why it is good for the construction permit, and how the process is going to track the issues that have been raised through the operating license phase.

Now, do either of you have things you don't like about what we just talked about that you think ought to be focused differently? And I think we are looking at two hours. Maybe a half hour, an hour, and a half hour kind of --

MEMBER POWERS: Yes.

CHAIRMAN BLEY: And somebody will drag you out into details somewhere, and we'll try to control you to some extent.

MS. GAVRILAS: So we're going to work with NAPRI and try to minimize redundancy in the two -- in the presentations, because we don't have time and conform to the flow that you just described.

MEMBER POWERS: Licensees should come armed with a lot of backup slides, because what is -- only about half of -- you're talking to only about half the Committee. It's a vocal half.

(Laughter.)

So don't get rid of any of the viewgraphs that you presented to us. Have them --

be ready to respond to what you think is off-the-wall questions, because the half you're not talking to is maybe not the quietest members.

MS. BANERJEE: Right now, we have two hours. Shall I ask for another half hour?

CHAIRMAN BLEY: I think let's try to keep it to two, I think, unless --

MEMBER POWERS: I think John is so -- I would say nervous as a whore in church, but that would be bad on the record, so --

CHAIRMAN BLEY: Yes, we are on the record. Dana is famous now.

(Laughter.)

And I hate to say this, but watch the news, because we might not be here in two weeks. There is a lot of rumors going around, and it's not going to -- that the government is going to shut down. If that happens, we can't meet. We don't have --

MS. GAVRILAS: If the government shuts down -- it's real. So if the government does shut down, we're going to have to reschedule everything, because the whole timeline is going to slip.

CHAIRMAN BLEY: And there's lots of congestion here in the next three months. So if

the government shuts down, it's going to be a mess all around. Yes.

Okay. Anything more? Maitri?

MS. BANERJEE: We are not allowed to work, actually, when we are shut down.

CHAIRMAN BLEY: Right. So we can't do it.

Okay. Well, I would like to add to everybody else's thanks. Good presentations, great discussions, and we really appreciate how prepared everyone was for this meeting.

MS. GAVRILAS: Thank you. And OGC just whispered we don't have a closed part to the full Committee meeting, right?

CHAIRMAN BLEY: That's true.

MS. GAVRILAS: Okay.

CHAIRMAN BLEY: Yes. We'll avoid --

MS. GAVRILAS: Thank you.

CHAIRMAN BLEY: -- the proprietary stuff.

Meeting is closed. Thank you all. Members, too.

(Whereupon, the above-entitled matter was concluded at 4:29 p.m.)

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Advisory Committee on Reactor Safeguards
Radiation Protection & Nuclear Materials Subcommittee
Meeting on the SHINE Construction Permit Application

September 22, 2015

Chapter 6b – Radioisotope Production Facility Engineered Safety Features and Nuclear Criticality Control

Eric Van Abel, SHINE
September 22, 2015

Overview

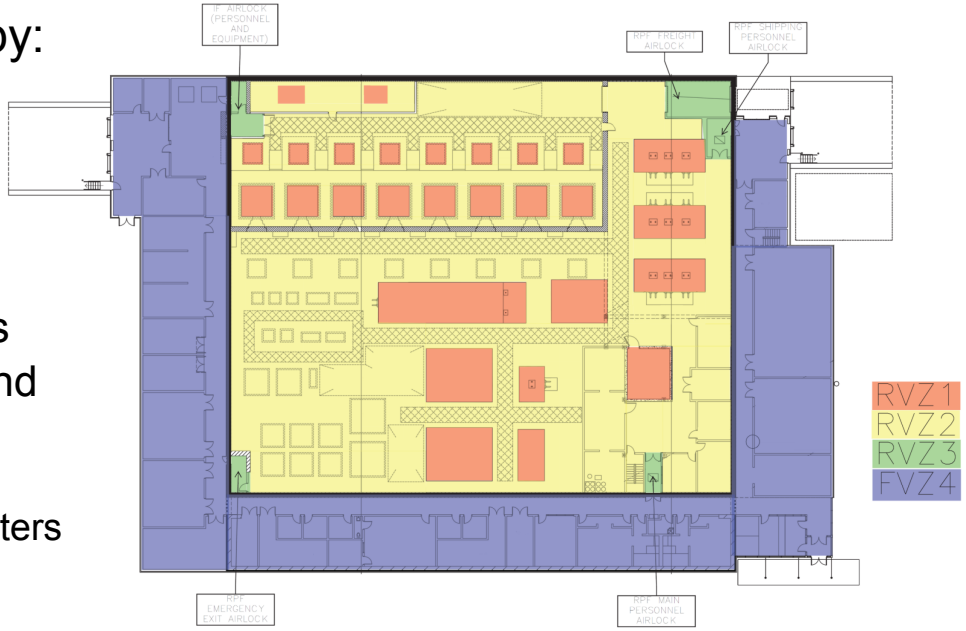
- Chapter 6b covers Engineered Safety Features (ESFs) and criticality safety controls in the radioisotope production facility (RPF)
- ESFs are passive or active features designed to mitigate the consequences of postulated accidents and ensure radiological and chemical exposures within acceptable limits
- The criticality safety controls section of the PSAR covers Nuclear Criticality Safety (NCS) controls for the RPF and structures, systems, and components (SSCs) where uranium could be present in sufficient mass for a nuclear criticality accident
  - Irradiation facility (IF) is not within the scope of this section
  - Ensure that nuclear processes remain subcritical during normal and credible abnormal conditions

Engineered Safety Features

- Five design basis accidents (DBAs) are addressed for the RPF
- Three DBAs require ESFs to mitigate consequences
  - Critical equipment malfunction
  - RPF Fire
  - Accidents with hazardous chemicals
- Confinement is a low-leakage boundary surrounding radioactive materials (or hazardous chemicals produced from licensed materials) that could be released during an accident
  - Similar to the IF, confinement is provided by the structure of the cells containing radiological material and the ventilation system ductwork
- ESF functions for the RPF are provided by confinement system barriers, active valves and dampers, and the Radiological Integrated Control System (RICS)

Engineered Safety Features

- Confinement functions provided by:
  - Biological shielding, including the walls of hot cells, tank vaults, and pipe trenches
    - Including associated penetration seals
  - Isolation valves on piping systems
  - RCA Ventilation Zone 1 (RVZ1) and RCA Ventilation Zone 2 (RVZ2)
    - RVZ1 hot cell isolation dampers
    - RVZ1 and RVZ2 ductwork up to filters
    - RVZ1 and RVZ2 filters
    - RVZ2 isolation dampers
  - RICS



Locations of ventilation zones

- SHINE protects public health and safety via a confinement system
  - Radionuclide inventory in any one confinement area is approximately 10,000 times less than a power reactor
  - Low dispersion forces due to relatively low temperature and pressure of processes

Actuation of Engineered Safety Features

- During normal operation, hot cells are maintained at negative pressure relative to surrounding environment, resulting in in-leakage of air
- In the event of a DBA that releases radioactive material (RAM) into a hot cell, the RAM would be confined by the walls of the cell itself
  - Airborne radioactive material in the hot cell is transported into the ventilation system
  - High radiation detected in the ventilation system initiates confinement isolation signal
  - Confinement isolation signal automatically closes bubble-tight isolation dampers on the inlet and outlet ventilation ports of the cell
- Dampers also close automatically on loss of power (fail-closed)
- ESF actuation thresholds determined during detailed design and will be set low enough to ensure 10 CFR 20 limits are not exceeded

Engineered Safety Features Requirements

- SSCs that perform ESF functions are safety-related and will meet the single-failure criterion
- Duct and housing leak rate tests are to be performed in accordance with ASME N511
  - Specific acceptable leak rates will be based on final safety analysis
- Bubble-tight isolation dampers will:
  - Maintain functional integrity during normal operations and accident conditions
  - Maintain acceptable leak-tightness following DBE
  - Maintain structural integrity under fan shut-off pressure
  - Provide damper position indication
- Low leakage seals are provided on each penetration
  - Overall leakage rates will ensure that assumptions in the accident analysis are bounding

Engineered Safety Features

- As with the IF components, RPF ESFs are periodically tested to ensure perform their safety functions when required
  - Penetration seals, isolation valves, bubble-tight isolation dampers, gloveboxes and other components relied upon for confinement will be tested prior to and during operation
  - Required testing, including testing intervals, will be specified in the Technical Specifications with the Operating License (OL) application
- No emergency cooling systems for RPF processes are required
  - Following loss of RPCS (process chilled water system), systems requiring cooling are shutdown until cooling can be restored
    - Process Vessel Vent System (PVVS) blower continues to operate via the uninterruptible electrical power supply system (UPSS)
    - PVVS blower is small and is not expected to require cooling beyond natural convection
  - Fission product decay heat removal requirements are minimal and accomplished via natural convection

Radioisotope Production Facility Criticality Safety

- The design of the RPF and the NCS program will ensure that an inadvertent nuclear criticality is highly unlikely
- The facility and NCS program will be designed to meet the requirements of the following, as modified by Regulatory Guide 3.71:
  - ANSI/ANS-8.1-1998 (R2007)
  - ANSI/ANS-8.3-1997 (R2012)
  - ANSI/ANS-8.7-1998 (R2007)
  - ANSI/ANS-8.10-1983 (R2005)
  - ANSI/ANS-8.19-2005
  - ANSI/ANS-8.20-1991 (R2005)
  - ANSI/ANS-8.21-1995 (R2011)
  - ANSI/ANS-8.23-2007 (R2012)
  - ANSI/ANS-8.24-2007 (R2012)
  - ANSI/ANS-8.26-2007 (R2012)

Nuclear Criticality Safety Program

- NCS program will contain the following elements:
  - Criticality safety policy statement
  - Verification and validation requirements
  - Nuclear criticality safety evaluation (NCSE) requirements
  - Training and qualifications
  - Implementation of criticality safety controls and limits
  - Configuration control
  - Audits and inspections
  - Criticality safety non-compliance processes
  - Criticality safety guidelines for fire fighting
  - Emergency preparedness plan and response procedures manual
  - Criticality detection and alarm system requirements
  - Testing and calibration of active controls
  - Criticality safety controls program

Nuclear Criticality Safety Program

- The design of the RPF will adhere to the double contingency principle (DCP)
- A NCS training program will be developed and implemented for personnel in the SHINE facility
  - Analysis of jobs and tasks performed by worker to ensure learning objectives are appropriate for the respective workers
  - NCS staff will be trained and qualified in accordance with ANSI/ANS-8.26-2007
- Criticality Accident Alarm System (CAAS) will provide for detection and annunciation of criticality accidents for emergency response
  - As described in Chapter 7, each area requiring coverage is covered by at least two detectors
  - CAAS is safety-related and powered from the UPSS
  - Personnel will be trained to recognize criticality accident alarm and how to evacuate safely through the quickest and most direct routes practical

Nuclear Criticality Safety Program

- Prior to implementing changes that involve or could affect SNM, it must be determined that the entire process will be subcritical under both normal and credible abnormal conditions
  - SHINE configuration management program will include criticality safety controls
  - Criticality safety controls will not be changed without appropriate review by qualified criticality safety engineers
  - NCS controls will be incorporated into operating procedures and equipment drawings and explicitly identified to ensure they are not changed without review
  - 10 CFR 50.59 will be used to determine if a license amendment request is required

Nuclear Criticality Safety Controls

- SHINE will follow the technical practices for the use of each controlled parameter as described in Section 6b.3 of the Interim Staff Guidance (ISG) augmenting NUREG-1537, Part 2
  - SHINE will assume the most reactive conditions for those parameters not controlled
- NCSEs and analyses will be used to identify parameters within a system and the necessary controls
- SSCs that are identified as NCS safety controls required to prevent or mitigate criticality accidents (e.g., criticality-safe sump catch tank geometry) will be safety-related
  - Safety-related SSCs receive the full measure of the SHINE QAPD
  - Administrative controls to ensure criticality safety will be described in the OL application and implemented through facility procedures
  - Per the Technical Specifications, written procedures shall be established, implemented, and maintained covering activities in the criticality safety program

Nuclear Criticality Safety Controls

- Passive engineered controls (e.g., geometry of tanks) are used as the preferred means to ensure NCS
- SHINE uses subcritical by design vessels and piping for the RPF processes
  - Each of the RPF process tanks, with the exception of the liquid waste processing tanks, are criticality-safe by geometry
  - Tanks are designed to be criticality safe at the most reactive uranium concentration, ignoring realistic saturation behavior
  - The absence of appreciable quantities of fissile material is measured and independently verified prior to the transfer to the waste processing tanks
- Pipe runs are single-parameter criticality-safe by geometry
- Criticality-safe tank vaults are connected via a non-valved gravity drain to a criticality-safe sump catch tank, which is criticality-safe by geometry

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Advisory Committee on Reactor Safeguards
Radiation Protection & Nuclear Materials Subcommittee
Meeting on SHINE Construction Permit Application

Chapter 6b
Engineered Safety Features & Nuclear Criticality Safety

Osiris Siurano-Perez, Project Manager, U.S. NRC
Christopher S. Tripp, Sr. Nuclear Process Engineer, U.S. NRC

September 22, 2015

Engineered Safety Features

- Engineered safety features (ESFs) are mitigative, not preventative

- In the radioisotope production facility (RPF), ESFs mitigate three design basis accidents:
  - Critical equipment malfunction
  - RPF fire
  - Accidents with hazardous chemicals

Engineered Safety Features - Confinement

- SHINE PSAR Table 6b.2-2, presents five SSCs related to confinement:
  - RVZ1 hot cell isolation dampers, ductwork up to filters, and filters
  - RVZ2 isolation dampers, ductwork up to filters, and filters
  - RICS provides confinement isolation signal
  - Isolation valves on piping systems
  - Hot cells, tank vaults, and pipe trenches

Design Standards

- ASME N510, Testing of Nuclear Air Treatment Systems, 2007
- ASME N511, In-Service Testing of Nuclear Air Treatment, Heating, Ventilating and Air-Conditioning Systems, 2007
- ASME AG-1, Code on Nuclear Air and Gas Treatment, 2009
- IEEE 379, Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems, 2000
- Regulatory Guide 1.53, Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems, 2003
- International Mechanical Code, 2012

Conclusions on RPF Confinement

If the RPF is designed and constructed as described in PSAR Section 6b, the radiological consequences from accidents will be reduced by the confinement ESFs to values that do not exceed the applicable limits of 10 CFR 20 and the chemical exposure criteria specified in PSAR Section 3.5b.

Nuclear Criticality Safety (NCS)

- Prevention of inadvertent nuclear criticality in the Radioisotope Production Facility (RPF); protection against consequences of nuclear criticality
- Acceptability based on ISG to NUREG-1537, Part 2:
  - Commitment to elements of an NCS Program applicable to design and construction
  - Commitment to principal design criteria and design bases (referred to as “technical practices” in ISG)
- Subcritical under normal and credible abnormal conditions
- Compliance with double contingency principle (DCP)
- Criticality accident alarm system (CAAS) & associated emergency planning

Staff Review

- Section 6b.3 of SHINE PSAR, as supplemented by responses to RAIs

- Criticality code validation report, NCS Manual, and preliminary NCS Evaluations (NCSEs)

- Staff considered principal design criteria and design bases to provide reasonable assurance the final design will ensure subcriticality under normal and credible abnormal conditions

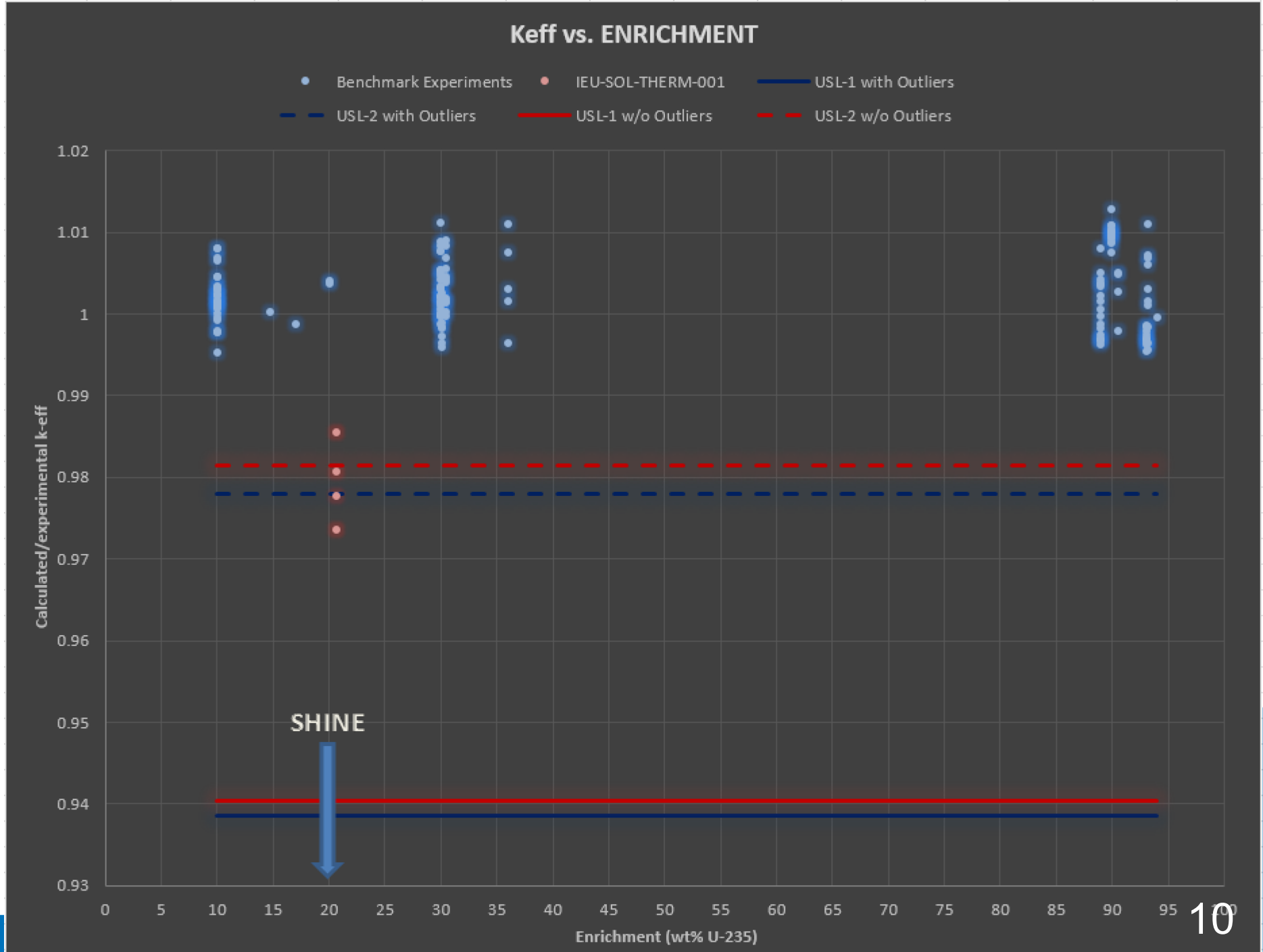
Summary of Application

- Commitment to CAAS meeting 10 CFR 70.24
- Commitments to NCS Program applicable to design and construction
  - Based on ANSI/ANS-8 programmatic standards
  - Development of NCSEs for limits and controls
  - Management measures applicable to design and construction (e.g., configuration control)
  - Use of qualified NCS staff
- Commitments to design criteria for NCS
  - Compliance with DCP based on controlled parameters
  - Determination of Upper Safety Limit (USL) for k_{eff}
  - Ensuring “credible” criticality events are “highly unlikely”

Validation & Subcritical Margin

- Safety limits on controlled parameters based on computer code methods (MCNP) validated using critical benchmark data*
- Few benchmarks exist for 20wt% ^{235}U and uranyl sulfate solution
- Closest experiments to SHINE conditions underpredicted k_{eff} by ~2.9%; discarded as outliers
- RG-3.71 indicates outliers should only be rejected based on "inconsistency of the data with known physical behavior"
- Preliminary research shows critical volumes in benchmark evaluation underestimated by ~3% and may account for low k_{eff}
- SHINE RAI response (6b.3-34) indicated it identified problem as being with the model and not the benchmark; will incorporate the 4 experiments in question into future revision of validation report
- Until complete, SHINE will adopt minimum subcritical margin of 0.06

*International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHECSBE), INL


Keff vs. ENRICHMENT
Benchmark Experiments
IEU-SOL-THERM-001
USL-1 with Outliers
USL-2 with Outliers
USL-1 w/o Outliers
USL-2 w/o Outliers
Calculated/experimental k-eff
1.02
1.01
1
0.99
0.98
0.97
0.96
0.95
0.94
0.93
SHINE
0
5
10
15
20
25
30
35
40
45
50
55
60
65
70
75
80
85
90
95
100
Enrichment (wt% U-235)

Technical Practices for NCS

- ISG to NUREG-1537, Part 2, indicates applicant should commit to acceptance criteria for use and modeling of controlled parameters (e.g., mass, geometry, moderation…)
- PSAR did not contain those commitments
- Initial review of preliminary NCSEs showed the applicant generally modeling parameters consistent with industry practice (as specified in ANSI/ANS-8.1 and related standards)
- Commitments to technical practices for modeling provide conservative margin that is part of subcritical margin to provide assurance of subcriticality under normal and credible abnormal conditions

Likelihood of NCS Scenarios

- PSAR initially said criticality would be made “not credible”
- Definition of “not credible” consistent with 3 criteria in NUREG-1520, Rev. 1 (p. 3-27), but allowed use of reliance on control features to make determination
- NUREG-1520 does not allow reliance on any features that may credible fail or be rendered ineffective as the result of a change
- SHINE agreed to instead use “highly unlikely” standard based on event frequencies consistent with NUREG-1520, Chapter 3, “Integrated Safety Analysis”
- SHINE has committed to use of preferred control hierarchy:
  - Passive engineered over active engineered
  - Engineered over administrative
  - Fixed geometry preferred means of control

Proposed Permit Conditions

1) SHINE will provide technical basis for design of the CAAS, including method for determining detector placement, prior to installation

2) SHINE will provide basis for determining that criticality is “not credible” prior to installing process equipment

Proposed Permit Conditions

3) SHINE will submit summaries of NCSEs for each process area prior to installing process equipment

Summaries will include:

- List of NCS hazards
- List of NCS controlled parameters
- Description of normal and abnormal conditions
- Description of approach to meeting the DCP
- List of anticipated engineered controls and assumptions

4) SHINE will account for production of fissile isotopes other than ^{235}U in performing NCSEs/calculations

Discussion

Chapter 11 – Radiation Protection and Waste Management

Mike Launi, Sargent and Lundy
Ernest Wright, Sargent and Lundy
September 22, 2015

Radiation Sources

- Source terms for the SHINE facility are presented in PSAR Tables 11.1-1 through 11.1-3
  - Contain assumptions for nominal, limiting (i.e., including operational uncertainties), and bounding (used for accident analyses) cases
  - Contain values for at shutdown (after irradiation), post shutdown (after decay), and post extraction

Normal Operations

- Airborne, liquid and solid radioactive sources are considered
  - Airborne sources are from gases produced as a byproduct of Mo-99 production
  - Liquid sources are present at a number of locations (there are no radioactive liquid discharges from the facility)
  - Solid sources exist in several locations
- Activities designed such that the estimated annual doses to the maximally exposed individual (MEI) and the nearest resident are below the dose constraint specified in 10 CFR 20.1101(d) for normal operations
  - Calculated per International Commission on Radiological Protection (ICRP) 30, considering both direct exposure and potential environmental pathways (vegetable/meat/milk ingestion)
  - SHINE will incorporate age-dependence per ICRP 72 for FSAR
  - MEI = 9.0 mrem per year
  - Nearest resident (0.33 miles) = 0.6 mrem per year

Radiation Protection Program

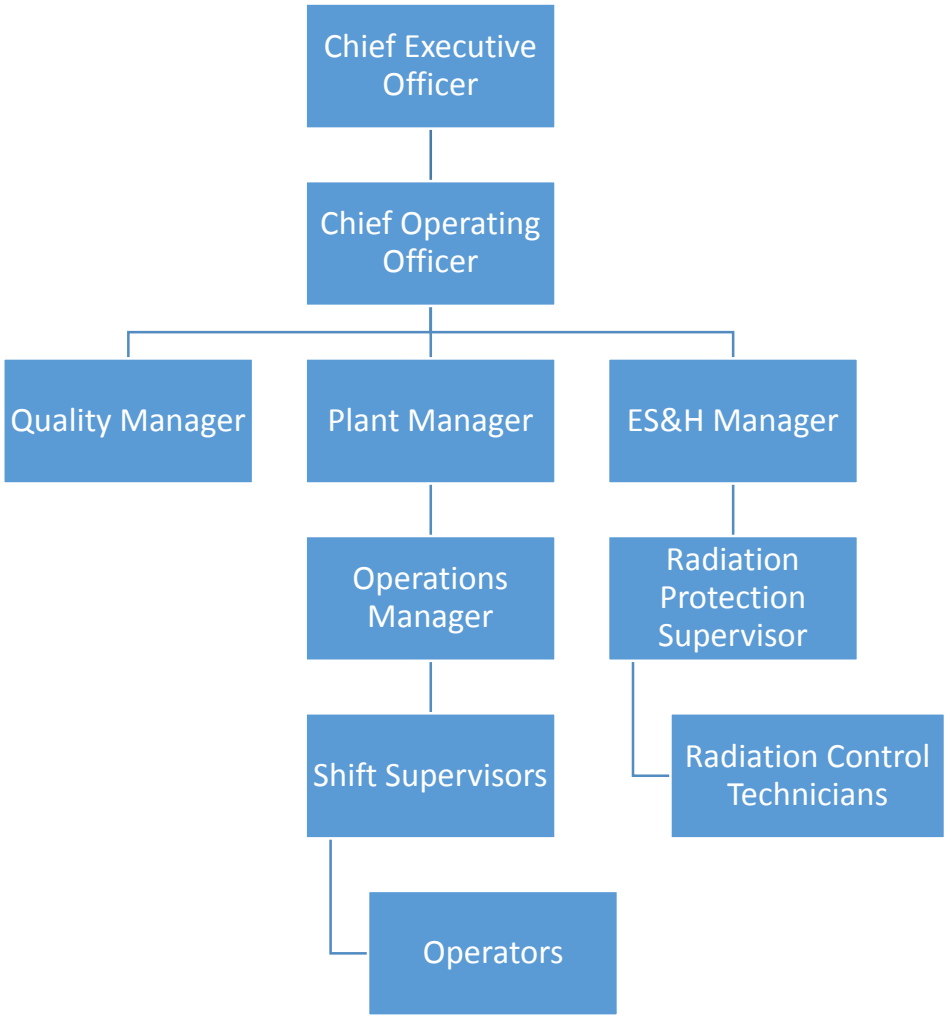
- Radiation Protection (RP) program will meet regulatory requirements
  - 10 CFR 20, Subpart B
  - Uses guidance contained in Regulatory Guide 8.2
- Objectives
  - Prevent acute radiation injuries (non-stochastic or deterministic effects)
  - Limit the potential risks of probabilistic (stochastic) effects (which may result from chronic exposure) to acceptable levels

Radiation Protection Program

- Preliminary administrative exposure limits

Type of Dose	10 CFR 20 limit (rem/year)	SHINE preliminary admin limit (rem/year)
Adult worker TEDE	5	0.5
DDE and CDE	50	5
Eye LDE	15	1.5
Skin or extremity SDE	50	5

Radiation Protection Program



Radiation Protection Program

- Commitment to written Radiation Protection procedures
  - RP procedures are prepared, reviewed and approved
  - Work in radiologically controlled areas is performed in accordance with radiation work permits (RWP)
  - Uses guidance contained in Regulatory Guide 8.10
- Commitment to Radiation Protection training
  - Uses guidance contained in Regulatory Guides 8.10, 8.13, and 8.29, and ASTM E1168-95
  - Personnel entering restricted or controlled areas are trained or are provided escorts who have been trained
  - Retraining conducted at least annually

ALARA Program

- SHINE is committed to an operating philosophy that maintains occupational exposures to radiation consistent with As Low as Reasonably Achievable (ALARA) principles
  - Installing temporary and permanent shielding of radioactive material
  - Use of time and distance to minimize exposure to personnel

ALARA Program

- Design considerations for maintaining personnel exposures ALARA include designing structures, systems, and components (SSCs)
  - Such that radioactive material, to the greatest extent practical, is remote handled and isolated from on-site personnel by shielded compartments and hot cells.
  - For reliability and maintainability, thereby reducing the maintenance requirements on radioactive components.
  - To reduce the radiation fields and control streaming, thereby reducing radiation exposure during operation, maintenance, and inspection activities.
  - To reduce access, repair and removal times, thereby reducing the time spent in radiation fields during operation, maintenance, and inspection.

Radiation Monitoring and Surveying

- Personnel monitors
  - Personnel entering radiologically restricted areas wear personnel monitoring devices
- Continuous air monitors
- Continuous tritium detectors
- Stack release monitoring
  - Continuous noble gases, aerosols, iodine, and tritium effluent monitoring
- Radiation area monitors
- Control point monitoring
  - Portal monitors, friskers, hand and foot monitors, and small article monitors
- Criticality monitoring

Radiation Monitoring and Surveying

- Radiation surveys conducted:
  1. to ascertain radiation levels, concentrations of radioactive materials, and potential radiological hazards
  2. to detect releases of radioactive material from facility equipment and operations.
- Comply with 10 CFR 20
- Uses guidance contained in
  - Regulatory Guide 8.2
  - Regulatory Guide 8.7
  - Regulatory Guide 8.9
  - Regulatory Guide 8.24
  - Regulatory Guide 8.34
  - ANSI N323-1978

Radiation Exposure Control and Dosimetry

- Unrestricted area is the area beyond the site boundary.
- Restricted area access is limited by SHINE for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials
  - Radiation areas (>5 mrem/hr at 30 centimeters)
  - High radiation areas (>100 mrem/hr at 30 centimeters)
    - Not accessible during routine operations
  - Very high radiation areas (>500 rads/hr at 1 meter)
  - Airborne radioactivity areas
  - Contaminated areas
  - Areas of “caution” (e.g., areas with a potential for soluble uranium intake)

Contamination Control Equipment and Facility Layout

- General, equipment and facility layout design considerations to prevent the spread of contamination to the facility and the environment and to facilitate eventual decommissioning in accordance with 10 CFR 20.1406
  - Process equipment containing irradiated material is located within shielded compartments or hot cells
  - Access to and egress from restricted areas is strictly controlled via administrative procedures (i.e., radiation work permits) and passive confinement structure design
  - The use of embedded pipes is minimized; shielded pipe trenches provide for liquid and airborne confinement and leakage detection, and cover blocks allow for inspection

Environmental Monitoring

- Radiological environmental monitoring in accordance with 10 CFR 20.1302
  - Considered guidance from Regulatory Guide 4.1 and NUREG-1301
- Radiological Effluent Monitoring Program (REMP)
  - Direct radiation exposure (24 locations)
  - Airborne exposure (5 continuous air samplers)
  - Groundwater (site test wells)
  - Ingestion exposure (milk, at least first 5 years)
- Preoperational Baseline Monitoring

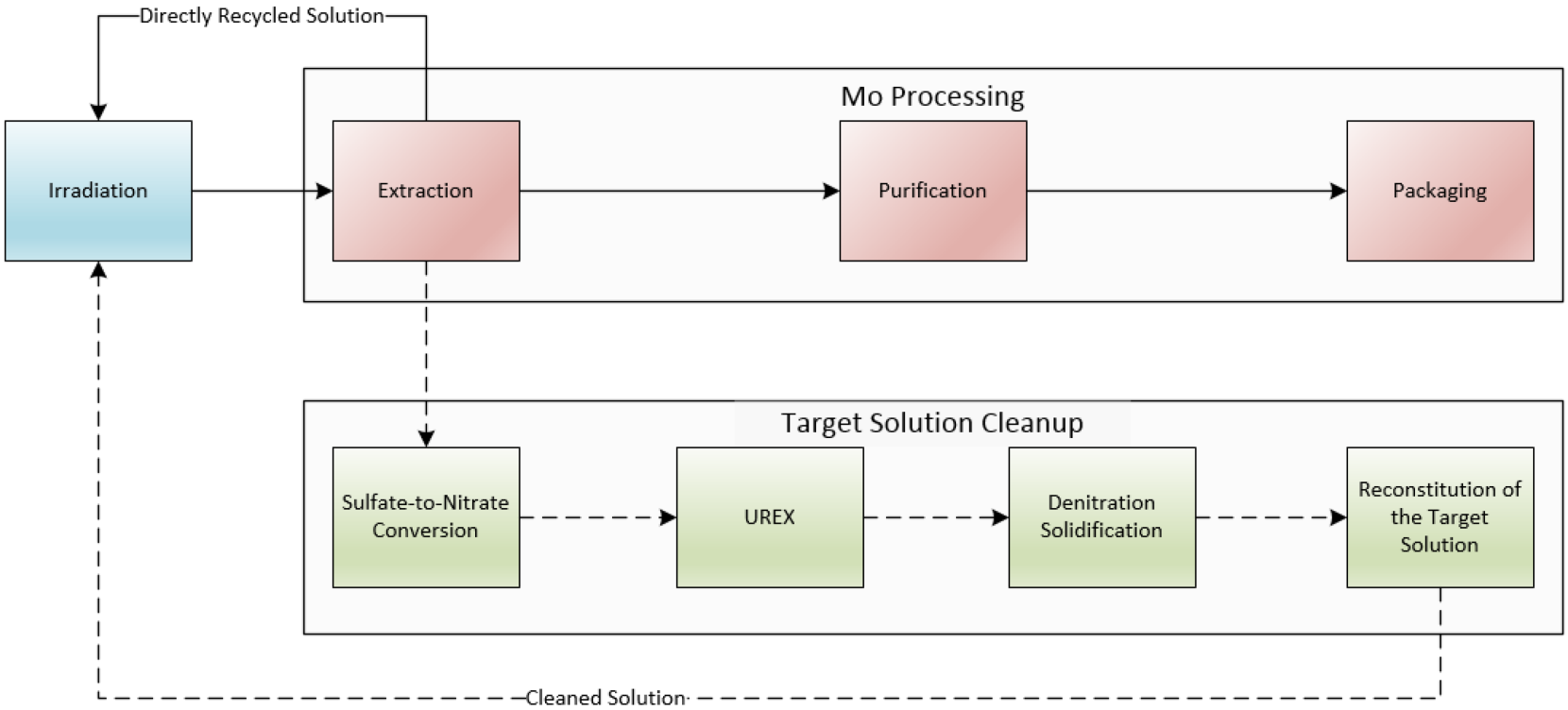
Respiratory Protection Program

- Process and engineering controls incorporated into the design of the heating, ventilation and air conditioning systems are the primary means of controlling the concentration of radioactive material in the air
- Respirators may also be used to maintain doses ALARA
- Respiratory protection program meets 10 CFR 20, Subpart H
- Fume hood and glovebox operations and maintenance involving uranium-235 processing uses guidance contained in Regulatory Guide 8.24

Radioactive Waste Management Program

- SHINE will comply with federal regulations related to radioactive wastes
  - 10 CFR 20, “Standards for Protection Against Radiation”
  - 10 CFR 61, “Licensing Requirements for Land Disposal of Radioactive Waste”
  - 10 CFR 71, “Packaging and Transportation of Radioactive Material”
  - 40 CFR, Chapter I, Subchapter F, “Radiation Protection Programs”
  - 40 CFR, Chapter I, Subchapter I, “Solid Wastes”
  - 49 CFR, Chapter I, Subchapter C, “Hazardous Materials Regulations”

Process Summary



Radioactive Waste Management

- Wastes classified as low level waste
- Waste streams, including quantities, composition and classification (A, B, C or GTCC) estimated based on preliminary design
  - Solids – neutron generators, extraction columns, resins, Target Solution Vessel (TSV) Off-Gas System (TOGS) zeolite beds, miscellaneous equipment, glassware and trash
  - Aqueous liquids (solidified) – extraction and purification wastes, Uranium Extraction System (UREX) raffinate, spent caustic scrubber solution, decontamination wastes
  - Organic liquids (processed offsite) – TBP/dodecane from UREX
  - Gaseous wastes – off-gas from the TSV is held for decay, processed through a caustic scrubber with off-gases from RPF vessels, passed through charcoal and HEPA filters, then released via the facility vent stack

Radioactive Waste Controls

- Key features of the pollution prevention and waste minimization program include:
  - Incorporation of radioactive waste minimization design features.
  - Employee training and education on general environmental activities and hazards regarding the facility, operations, and the pollution prevention program, as well as waste minimization requirements, goals, and accomplishments.
  - Responsibilities for pollution prevention and waste minimization.
  - Requirements for employees to consider pollution prevention and waste minimization in day-to-day activities and engineering.

Release of Radioactive Waste

- Radioactive wastes are processed and packaged as required to meet the waste acceptance criteria of licensed disposal facilities
- The SHINE facility does not discharge any material from the Radiologically Controlled Area (RCA) to the sanitary sewer
- Gaseous wastes are treated on site prior to release, analogous to the processes used in pressurized water reactors

Waste Stream Summary (1 of 2)
Based on Preliminary Design and Conservative Assumptions

Description	Class as Generated	Amount as Generated	Shipment Type and Number	Proposed Destination
Neutron Generator, Extraction Columns and Miscellaneous Trash (Solid)	A	4400 ft^3/yr	LSA, 3 per year	Energy Solutions
Coolant Cleanup Ion Exchange Resin (Solid)	A	48 ft^3/yr	LSA, 1 per year	Energy Solutions
Spent Solvent (Liquid)	A	22 gal/yr	LSA, 1 per year	Diversified Scientific Services, Inc.
Tc/I Columns (Solid)	C	16 gal/yr	Type B, <1 per year	Waste Control Specialists (WCS)
Zeolite Beds (Solid)	GTCC	0.4 ft^3/yr	Type B, 1 per year	WCS
Cs/Ce Media (Solid)	GTCC	16 gal/yr	Type B, <1 per year	WCS

Waste Stream Summary (2 of 2)
Based on Preliminary Design and Conservative Assumptions

Description	Class as Generated	Amount as Generated	Shipment Type and Number	Proposed Destination
Spent Washes	A	2100 gal/yr	LSA, 19 per year (Combined Solidified Liquid)	Energy Solutions
Rotary Evaporator Condensate	A	200 gal/yr		
UREX Raffinate	B	27,000 gal/yr		
Caustic Scrubber Solution	A	20,000 gal/yr		
Decontamination Waste	A	400 gal/yr		
Spent Eluate Solution	A	2600 gal/yr		

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Advisory Committee on Reactor Safeguards
Radiation Protection & Nuclear Materials Subcommittee
Meeting on SHINE Construction Permit Application

Chapter 11.1 Radiation Protection Program
Chapter 11.3 Respiratory Protection Program

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Technical Reviewers
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September 22, 2015

Radiation Protection Program

- Nature and magnitude of radiation sources
- Shielding and ventilation system design
- ALARA considerations
- Radiation monitoring, surveillance, and dosimetry
- Contamination controls
- Environmental monitoring
- Respiratory protective equipment

Regulatory Basis and Acceptance Review

- Regulatory Requirements
  - 10 CFR 50.34, “Contents of applications; technical information,” paragraph (a), “Preliminary safety analysis report”
  - 10 CFR 50.35, “Issuance of Construction Permits”
- Acceptance Criteria
  - NUREG-1537 and ISG, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria”

Areas of Review

- Shielding and ventilation system for IF and RPF
- Monitoring and dosimetry
- Occupational and public doses
- ALARA considerations
- Contamination controls
- Effluent and environmental monitoring
- Usage of respiratory protective equipment

Review Procedures and Technical Evaluation

- The staff performed a thorough and complete section-by-section evaluation of the technical information presented in Section 11.1 of the SHINE PSAR, as supplemented by responses to RAIs, to assess the adequacy of the radiation protection design features for SHINE's IF and RPF in support of the issuance of a construction permit
- In a similar manner, the staff performed a review of the respiratory protection program provided in Section 11.3 and compared it to 10 CFR 20 requirements and guidance

11.1.1 Radiation Sources

Key Aspects

- Staff's assessment of the shielding around the IU and RPF showed that the dose rate of <1 mrem/hr contained in the PSAR was reasonable. PSAR was supplemented by response to RAI 4a2.5-1
- Staff's review of zoning designations noted that Zone 1 (intended for routine occupancy) could have airborne activity concentrations between 0.01 and 1.0 DAC. PSAR was supplemented by response to RAI 11.1-1. Assurance needed from SHINE that concentrations up to 1.0 DAC in Zone 1 are ALARA
- Staff's review of effluent release source term noted that a sufficiently complete environmental pathway dose assessment had not been performed. RAI 11.1-9 issued. Response received, but further evaluation is needed

11.1.1 Radiation Sources

Supplemental Issues – after finalization of design of certain components or deferral to FSAR

- Source terms and final shielding design for liquid waste storage and Mo extraction and purification system
- Confirmation that Zone 1 (up to 1.0 DAC) air concentrations are ALARA
- Environmental exposure pathway dose assessment and compliance with Part 20 public dose limits and ALARA

11.1.2 Radiation Protection Program

Key Aspects

- Proposed facility organization and lines of authority for RPM – staff determined that RPM will have necessary independence and adequate staffing
- Training to be provided to staff and visitors – staff determined that training will be meet requirements of 10 CFR 19 and be consistent with Regulatory Guides 8.10, 8.13, and 8.29
- Radiation Safety Committee responsibilities – staff determined that RPC will be chartered and should be able to provide relevant oversight role
- Use of Radiation Work Permits – staff found that a commitment to use RWPs was contained in the PSAR and that the proposed nature of the RWPs should be effective in the management of radiation exposures by the SHINE staff
- No RAIs for 11.1.2

11.1.3 ALARA Program

Key Aspects

- Overall program considerations – staff noted that SHINE proposed to update and modify traffic control, security, access control and HP procedures as design and layout as experience is gained
- Program design considerations – staff determined that easy access to equipment requiring maintenance is provided, equipment with high radiation level is compartmentalized, and adequate provisions exist for storage and use of mobile shielding
- PSAR was missing a management commitment to develop and implement an ALARA program. RAI 11.1-3 requested such a commitment and SHINE provided it

11.1.4 Radiation Monitoring and Surveys

Key Aspects

- PSAR identified several types of sampling and monitoring equipment located within RCA, at RCA exits, and at plant stack
- CAMs will be used in controlled and restricted areas; however locations were not specified
- Control point monitoring will be performed by portal monitors, friskers, hand/shoe monitors and tool monitors
- Written surveillance program procedures are to be developed
- Staff found the level of specificity acceptable for the PSAR stage; no RAIs needed for this area

11.1.5 Radiation Exposure Control and Dosimetry

Key Aspects

- External dosimetry: All personnel entering restricted area will wear beta-gamma dosimeters (exchanged quarterly). Exposures >25% of admin limits will be investigated and reported to RPM
- Internal dosimetry: PSAR states that a combination of air concentrations, *in vivo* measurements, and excreta measurements will be used to demonstrate compliance with Part 20
- Support facilities: Facilitate RCA entry and exit, personnel decontamination, PPE storage
- PSAR needed to be supplemented regarding radiation area designations (RAI 11.1-4) and whether VHRAs will be included in facility design (RAI 11.1-5). Responses were sufficient to close both RAIs

11.1.6 Design Considerations for Decommissioning

Key Aspects

- PSAR describes features for draining, flushing and decontaminating equipment and that the equipment is designed to minimize buildup of radioactive material
- No RAIs were necessary and the staff found reasonable assurance that 10 CFR 20.1406 could be met

11.1.7 Environmental Monitoring

Key Aspects

- PSAR identifies direct exposure monitoring plus sampling of air, groundwater, and foodstuff
- Proposed program was considered by the staff to be not sufficiently robust and three RAIs were prepared:
  - RAI 11.1-6 requested clarification of air sampler vs. monitor
  - RAI 11.1-7 requested the applicant to re-assess its commitment to not regularly sample milk from nearby dairy animals (cow and goat)
  - RAI 1.1-8 requested the applicant to re-assess its position on the number of direct exposure monitoring locations using TLD
- Responses to all three RAIs were satisfactory

11.3 Respiratory Protection Program

Key Aspects

- The proposed respiratory protection program is consistent with the hierarchy of protection intended by Subpart H of 10 CFR 20, i.e., HVAC system considerations first, followed by the use of respiratory protective equipment only when HVAC controls are not practical or are ineffective
- The PSAR states that a fit factor for certain devices will be at least 500 times the assigned protection factor (APF) for the device. For devices where the fit factor is 1000, this means that the APF will approach 500,000 – a level that may be beyond the capabilities of most quantitative fit testing methods (clarify in FSAR)
- The description of the respiratory protection program is generally consistent with regulatory requirements and guidance documents, and represents an adequate foundation on which to construct the program elements that have been generally described
- No RAIs for 11.3

Evaluation Findings and Conclusions

- The staff's independent review determined that there was reasonable assurance that the occupational radiation exposure limits in 10 CFR 20.1201 would be met based on the shielding and ventilation system controls included in the design of the plant, as supplemented by:
  - Final design information for certain components (supercell and liquid waste storage tanks)
  - Administrative control measures such as posting and access controls
  - An analysis that ensures that the Zone 1 ventilation system controls are ALARA

Evaluation Findings and Conclusions

- Commitments made for the radiation protection organization and the manner that they are intended to be operated are acceptable at the PSAR stage regarding protection of the facility staff, the environment, and the public from unacceptable exposure to radiation

- The bases for the ALARA procedures and facility design elements for limiting access and personnel exposure give reasonable assurance that doses to occupational workers and the public will be maintained below regulatory limits and ALARA

Evaluation Findings and Conclusions

- The general types of monitoring and surveillance equipment for the tasks associated with facility operations plus SHINE's commitments to key Regulatory Guides give reasonable assurance that radioactive material and associated radiation exposures will be detected, monitored, and sampled consistent with 10 CFR Part 20 requirements
- The program for posting and access control regarding Restricted Areas, Controlled Areas, and Unrestricted Area, proposed access controls, and area radiological posting methodology is sufficient to meet the requirements of 10 CFR Part 20. The proposed external and internal radiation monitoring of all individuals required to be monitored provides reasonable assurance that the requirements of 10 CFR Part 20 will be met

Evaluation Findings and Conclusions

- The description and level of detail pertaining to plant design features that are intended to contain leakage from systems, monitor leakage if it does occur, minimize the buildup of contamination in process systems, and facilitate decontamination of systems and components are acceptable at the PSAR stage to support the staff's evaluation that the requirements of 10 CFR 20.1406 can be met

- The description of the respiratory protection program is generally consistent with regulatory requirements and guidance documents and represents an adequate foundation on which to construct the program elements

Discussion

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Chapter 11.2

Radioactive Waste Management

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U.S. Nuclear Regulatory Commission

September 22, 2015

Regulatory Basis and Acceptance Criteria

- Regulatory Requirements
  - 10 CFR 50.34, “Contents of applications; technical information,” Paragraph (a), “Preliminary safety analysis report”
  - 10 CFR 50.35, “Issuance of Construction Permits”
- Acceptance Criteria
  - NUREG-1537 and ISG, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria”

Radioactive Waste Management

- The SHINE facility will generate large volumes of radioactive wastes spanning waste disposal classes LSA to GTCC and all transportation classes
- The SHINE facility proposes extensive use of decay in storage, leading to on-site retention of wastes with attendant radiation protection issues
- The design of the Waste Staging and Shipping Building will be presented in the FSAR

Staff Review

- The staff review consisted of evaluation of the information presented in Section 11.2 of the SHINE PSAR, as supplemented by responses to RAIs, to assess the sufficiency of SHINE's proposed radioactive waste management program, controls, and releases of radioactive material in support of the issuance of a construction permit

- Staff used the Acceptance Criteria in NUREG-1537 and ISG, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria"

Summary of Application

- PSAR Section 11.2 describes:
  - 11.2.1 the proposed Radioactive Waste Management Program
  - 11.2.2 proposed Radioactive Waste Controls
  - 11.2.3 the identified requirements for Releases of Radioactive (solid) Waste

- Waste processing systems are described in Sections 9b.5 and 9b.7

Radioactive Waste Management Program

Key Aspects

- Staff's assessment of the program objectives, management and supervisors responsibilities, program elements such as self-assessments, audits, and record-keeping and documents control is that it presents a sufficient administrative structure to assure releases of gaseous and solid radioactive wastes are in accordance with the regulations
- Elements of the program that will be reviewed at the FSAR include the waste management charter, waste management procedure development, and how these elements are integrated into the conduct of operations

Radioactive Waste Controls

Key Aspects

- The radioactive waste management program is a subset of facility pollution prevention and waste minimization program, assuring that radioactive waste minimization is an objective
- Staff review of the tables in PSAR Section 11.2.2 concluded that the controls proposed for each identified stream of solid radioactive waste should assure appropriate processing and packaging for storage, transportation, and eventual disposal. The controls proposed in PSAR Section 11.2.2 include sampling of waste streams to support characterization of the waste streams and quantification of radioactive content in waste packages

Radioactive Waste Controls

- Liquid waste chemical characteristics and radioactive material content are significantly different than those found at nuclear power reactor plants
- In response to RAI 11.2-5 SHINE committed to solidification demonstration test runs during facility commissioning to assure the adequacy of the PCP to produce a packaged product meeting disposal site waste acceptance criteria
- Staff will review development of radioactive waste management operating procedures in the FSAR
- Staff will review waste package storage and handling within the waste staging and shipping building in the FSAR

Releases of Radioactive Waste

Key Aspects

- SHINE proposes no liquid radioactive effluent releases
- Gaseous radioactive releases addressed in Sections 11.1.3 and 11.1.7
- Staff review concludes that SHINE has identified the requirements for adequate packaging of solid wastes for transport and disposal and has committed to comply with 10 CFR Part 20, Subpart K and the waste acceptance criteria for the potential waste disposal sites

Evaluation Findings and Conclusions

- The proposed radioactive waste management program should provide sufficient administrative structure to assure compliance with the regulations and processes for continuous improvement
- The proposed radioactive waste controls should be sufficient to assure adequate packaging of solid wastes prior to disposal. Implementation of these controls will be further reviewed at the FSAR
- SHINE has identified the requirements for releases of radioactive wastes and committed to adhering to these requirements

Evaluation Findings and Conclusions

Staff assesses that the SHINE proposed approach to radioactive waste management is sufficient at the PSAR stage to justify issuance of a construction permit

Discussion

Chapter 12 – Quality Assurance Program Description

Jim McIntyre, Sargent and Lundy
September 22, 2015

SHINE Quality Assurance Program Description (QAPD)

- 10 CFR 50.34(a)(7) requires a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components (SSCs) of the facility.
- SHINE is required to implement the guidance contained in Parts 1 and 2 of NUREG-1537 (and the associated Interim Staff Guidance (ISG)) to meet regulatory requirements.
  - NUREG-1537 states that following ANSI/ANS-15.8 provides an acceptable method of complying with the program requirements of 10 CFR 50.34.
  - SHINE has developed the QAPD in accordance with ANSI/ANS-15.8-1995 (R2013).

SHINE Quality Assurance Program Description (QAPD)

- The SHINE QAPD describes the administrative and engineering controls for ensuring compliance with requirements, and applies to the design, construction, and operation of the SHINE facility.
  - SHINE will apply a graded approach to those items and activities that could affect the quality of safety-related SSCs and other components not specifically designated as safety-related.

SHINE QAPD – Graded Approach to Quality

- QL-1 shall implement the full measure of this QAPD and shall be applied to safety-related SSCs and to safety-related activities.
- QL-2 is applied to selected SSCs and activities intended to support or protect the safety function of safety-related equipment. Quality Assurance Program elements are applied to an extent that is commensurate with the item's importance to safety. Implementing documents establish program element applicability.
- QL-3 is applied to nonsafety-related SSCs and activities and does not support or protect the safety function of safety-related SSCs or activities.

SHINE QAPD – Design, Construction, and Modifications

- In accordance with ANSI/ANS-15.8-1995 (R2013), the SHINE QAPD contains the following requirements for establishing, managing, conducting, and assessing the program of controls over the design, construction, and modification of the SHINE facility:
  1. Organization
  2. Quality Assurance Program
  3. Design Control
  4. Procurement Document Control
  5. Procedures, Instructions, and Drawings
  6. Document Control

SHINE QAPD – Design, Construction, and Modifications (cont.)

7. Control of Purchased Items and Services
8. Identification and Control of Items
9. Control of Special Processes
10. Inspections
11. Test Control
12. Control of Measuring and Test Equipment
13. Handling, Storage, and Shipping
14. Inspection, Test, and Operating Status
15. Control of Non-Conforming Items and Services
16. Corrective Actions
17. Quality Records
18. Assessments

SHINE QAPD – Facility Operations

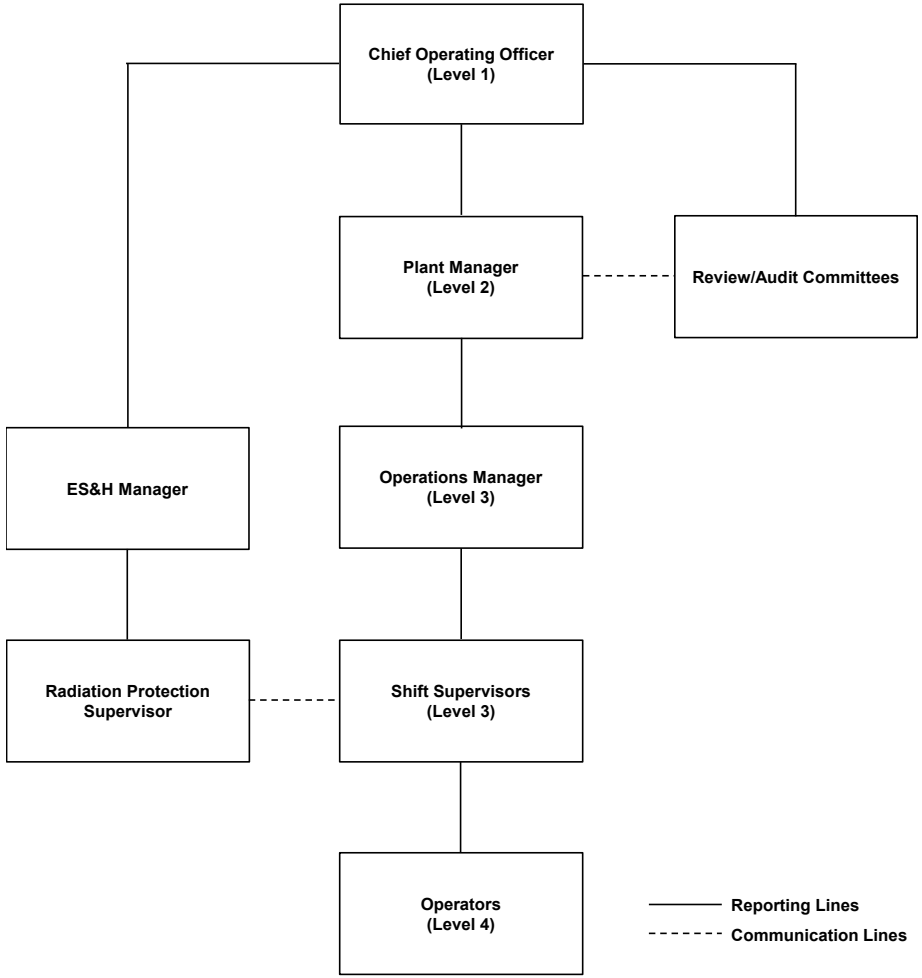
- In accordance with ANSI/ANS-15.8-1995 (R2013), the SHINE QAPD contains the following elements:
  1. Organization
  2. Quality Assurance Program
  3. Performance Monitoring
  4. Operator Experience
  5. Operating Conditions
  6. Operational Authority
  7. Control Area

SHINE QAPD – Facility Operations (cont.)

8. Ancillary Duties
9. Emergency Communications
10. Configuration Control
11. Lockouts and Tagouts
12. Test and Inspection
13. Operating Procedures
14. Operator Aid Postings
15. Equipment Labeling

Operational Structure

- Revised the based on comments at previous ACRS meeting.
  - Plant Manager changed to Level 2.
  - Operations Manager changed to Level 3.
  - The COO establishes review and audit committees, holds approval authority for those activities, and ensures that the appropriate technical expertise is available.
  - The ES&H Manager reports to the COO.



Review and Audit Activities

- Review and audit committees with the appropriate expertise and experience are established and members, designated by the COO, provide an independent assessment of the operation.
- The scope of the review function and the audit function are in accordance with Sections 6.2.3 and 6.2.4 of ANSI/ANS-15.1-2007, respectively.
  - Upon completion of a review, a written report of any findings and recommendations of the review committee shall be provided to SHINE Executive Management.
  - Deficiencies identified during an audit will be entered into the corrective action program.
  - Deficiencies uncovered that affect nuclear safety shall immediately be reported to Level 1 management.

Procedures

- In accordance with ANSI/ANS 15.1-2007, SHINE shall prepare, review, and approve written procedures for the following basic topics:
  1. startup, operation, and shutdown of the IU;
  2. target solution fill, draining, and movement within the SHINE facility;
  3. maintenance of major components of systems that may have an effect on nuclear safety;
  4. surveillance checks, calibrations and inspections required by the technical specifications;
  5. personnel radiation protection, consistent with applicable regulatory guidance. The procedures shall include management commitment and programs to maintain exposures and releases as low as reasonably achievable in accordance with applicable guidance;
  6. administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect nuclear safety;
  7. implementation of required plans (e.g., emergency, security); and
  8. use, receipt, and transfer of byproduct material.

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Chapter 12
Quality Assurance Program

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September 22, 2015

Regulatory Basis and Acceptance Criteria

- Regulatory Requirements:
  - 10 CFR 50.34, “Contents of applications; technical information,” paragraph (a)(7), requires a description of the quality assurance (QA) program
- Acceptance Criteria:
  - NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Part 1, “Format and Content,” and Part 2, “Standard Review Plan and Acceptance Criteria”

Acceptance Criteria

- Regulatory Guide 2.5, “Quality Assurance Requirements for Research and Test Reactors”
  - “The general requirements for establishing and executing a quality assurance program for the design, construction, testing, modification, and maintenance of research and test reactors in [American National Standards Institute/American Nuclear Society] ANSI/ANS-15.8-1995 provide an acceptable method for complying with the program requirements of 10 CFR 50.34, “Contents of applications; technical information.”

Staff Review

- The staff performed a thorough and complete section-by-section evaluation of the information presented in Appendix 12C of Chapter 12.9 in the SHINE Preliminary Safety Analysis Report (PSAR), as supplemented by responses to RAIs, to assess the adequacy of SHINE's Quality Assurance Program Description (QAPD), 2000-09-01, Revision 6 in support of the issuance of a construction permit

- Staff applied the guidance outlined in ANSI/ANS-15.8-1995, "Quality Assurance Program Requirements for Research Reactors." This standard provides criteria for QA in the design, construction, operation and decommissioning of research reactors

- Areas of review included all of the applicable QA program requirements outlined in the standard

Summary of Application

- SHINE provides an adequate description of its QA program for the design and construction phases of plant life
- In Section 3, “Facility Operations,” of SHINE’s QAPD, specific details were not provided. The staff determined details were not necessary to support the issuance of a construction permit. A more detailed evaluation will be deferred until receipt of the Final Safety Analysis Report (FSAR) supporting an operating license application
- In Section 5, “Decommissioning,” of SHINE’s QAPD, the applicant stated that this section would be updated at a later date. Therefore, the staff deferred the review of this section until the receipt of an FSAR supporting an operation license application

Staff Review of the QAPD

In evaluating SHINE's QA requirements, the staff determined additional information was needed to ensure the applicant had addressed the full scope of requirements outlined in ANSI/ANS-15.8-1995. Specifically, the following requests for additional information (RAIs) were addressed:

- RAI 12C.2 requested SHINE clarify the basis for not including the definition of 'experiment' in the QAPD

  — SHINE responded that it did not plan to conduct experiments or utilize experimental equipment

Staff Review of the QAPD

- RAI 12C.1-6 part (b) requested SHINE to clarify the difference between the definition of ‘audit’ and ‘assessment,’ as used in Section 2.18, “Assessment”

  — SHINE stated that it defines both ‘audit’ and ‘assessment’ as “a planned and documented activity performed to determine by investigation, examination, or evaluation of objective evidence the adequacy of and compliance with established procedures, drawings or other applicable documents, and the efficiency of implementation”

Staff Review of the QAPD

- RAI 12C.1-4 requested SHINE to clarify the definition of ‘safety-related’ and why it was acceptable to maintain key definitions that were used in the SHINE QAPD in a stand-alone administrative procedure

  — SHINE moved all the applicable definitions to the QAPD. The modified definitions of ‘commissioning’ and ‘management’ were acceptable, as the facility does not meet the definition of a ‘reactor’

Staff Review of the QAPD

- RAI 12C.4 requested SHINE to provide a performance based definition for the Part 4 (i.e., “that the potential for an inadvertent criticality accident is not credible”) of the 6-part definition of ‘safety-related SSCs’ or provide a discussion as to why it is not necessary. The staff also asked how SHINE’s definition aligns with ‘basic component’ in § 21.3

  — Part 4 now states, “That all nuclear processes are subcritical, including use of an approved margin of criticality”

  — In part (b) of the RAI, SHINE stated it considers safety-related SSCs, as defined in part (a), to be basic components, as defined in § 21.3(3)

Staff Review of the QAPD

- RAI 12C.E2-6 part (a) requested SHINE to clarify if the Quality Level (QL)-1 classification applies to safety-related activities, as well as SSCs. Additionally, part (b) asked SHINE to clarify how the definition of the QL-2 is based on safety significance. Finally, part (c) requested how the QL-2 classification is intended to be applied only to selected nonsafety-related SSCs and activities
  - SHINE revised the QL-1 definition to also apply to activities
  - SHINE revised the definition of the QL-2 classification to be based on safety significance and application of the full scope of its QAPD
  - SHINE stated that the QL-2 classification is intended to be applied only to selected nonsafety-related SSCs and activities. A QL-3 classification was inserted for strictly nonsafety-related SSCs

Staff Review of the QAPD

- RAI 12C.5-1 requested SHINE to provide additional information regarding the QA requirements that apply during the decommissioning phase

  — SHINE responded that the term ‘decommissioning’ had be been removed from Section 1.1 and 1.2 of the QAPD and revised Section 5, “Decommissioning,” to state that it will be updated at a later date

Staff Review of the QAPD

- RAI 12C.1-6 part (a) requested SHINE to clarify the basis for not including the definition of ‘experiment’ in the QAPD
  - SHINE responded that it did not plan to conduct experiments or utilize experimental equipment
  - SHINE revised Section 2.10 to remove the phrase ‘experiment fabrication’ and removed Section 2.19, “Experimental Equipment,” from the QAPD

Evaluation Findings and Conclusions

- The quality program described in the SHINE PSAR meets the regulatory requirements and acceptance criteria for the issuance of a construction permit, with the acknowledgement that additional requirements will need to be considered for the operational and decommissioning phases of plant life

- Based on staff review, it is concluded that this level of review of SHINE's QA program is adequate because any required changes to the QAPD are subject to review and acceptance prior to issuance of the FSAR

Evaluation Findings and Conclusions (cont.)

- The staff determined that the information to be included in SHINE FSAR Section 12.9, “Quality Assurance,” is sufficient and met the applicable regulatory requirements and guidance to support the issuance of a construction permit in accordance with 10 CFR 50.35:
  - A complete description in Section 3, “Facility Operations,” and Section 5, “Decommissioning,” is not necessary to support the issuance of a construction permit. Therefore, the staff deferred a more detailed evaluation at receipt of an FSAR
  - The staff has found acceptable the applicability of SHINE’s definition of ‘safety-related for its facility’s SSCs and activities. Also, the staff determined that the third definition of basic component in Part 21 is adequate to apply to the SHINE facility

Discussion

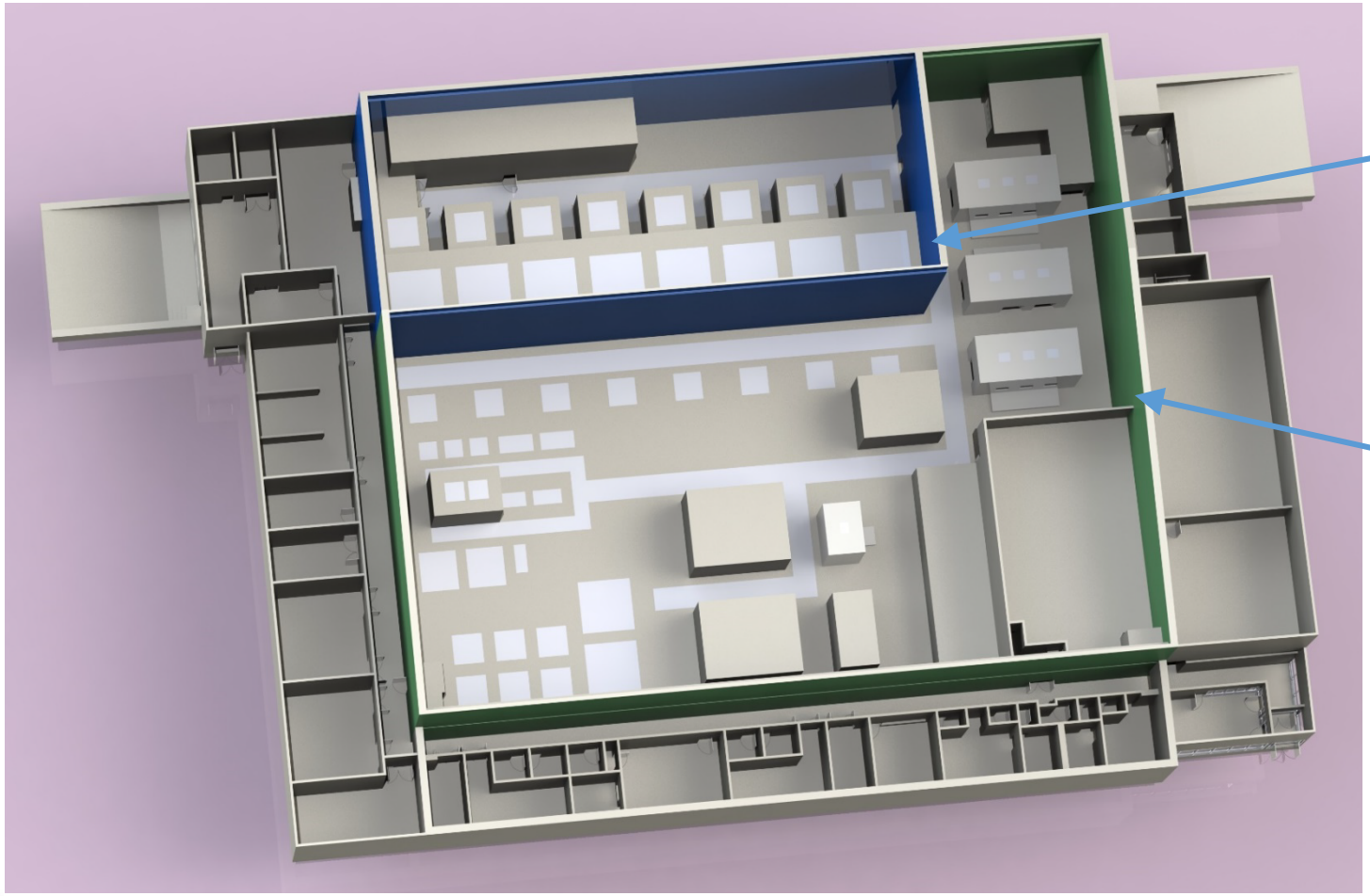
Chapter 13b – Radioisotope Production Facility Accident Analysis

Eric Van Abel, SHINE
September 22, 2015

Overview

- Radioisotope production facility (RPF) handles target solution and radioisotope processing outside of the irradiation facility (IF)
- Two types of hazards assessment were used to evaluate potential hazards in the facility
  - Hazard and Operability Study (HAZOPS), which evaluates process upsets and deviations
  - Preliminary Hazards Analysis (PHA), which developed an initial set of potential initiating events (IEs) and accident scenarios based on the hazards present
- IEs and potential accident scenarios grouped into common categories

Overview



Design Basis Accidents

- The bases for the identification of DBAs and IEs within the RPF include:
  - HAZOPS and PHA within the Integrated Safety Analysis (ISA)
  - List of IEs and accidents identified in the Interim Staff Guidance (ISG) augmenting NUREG-1537
  - Experience of the hazards analysis team in a range of disciplines
- Based on current preliminary design for the processes and facility, and will be re-evaluated during detailed design
- Qualitative evaluations were performed within categories to identify the bounding or limiting accidents and scenarios
- Quantitative evaluations were then performed to determine consequences of the DBAs

Radioisotope Production Facility Accidents

- Analyzed accident categories
  - Maximum hypothetical accident (MHA)
  - External events
  - Critical equipment malfunction
  - Inadvertent nuclear criticality
  - RPF Fire
  - Hazardous chemicals produced from licensed materials

Maximum Hypothetical Accident

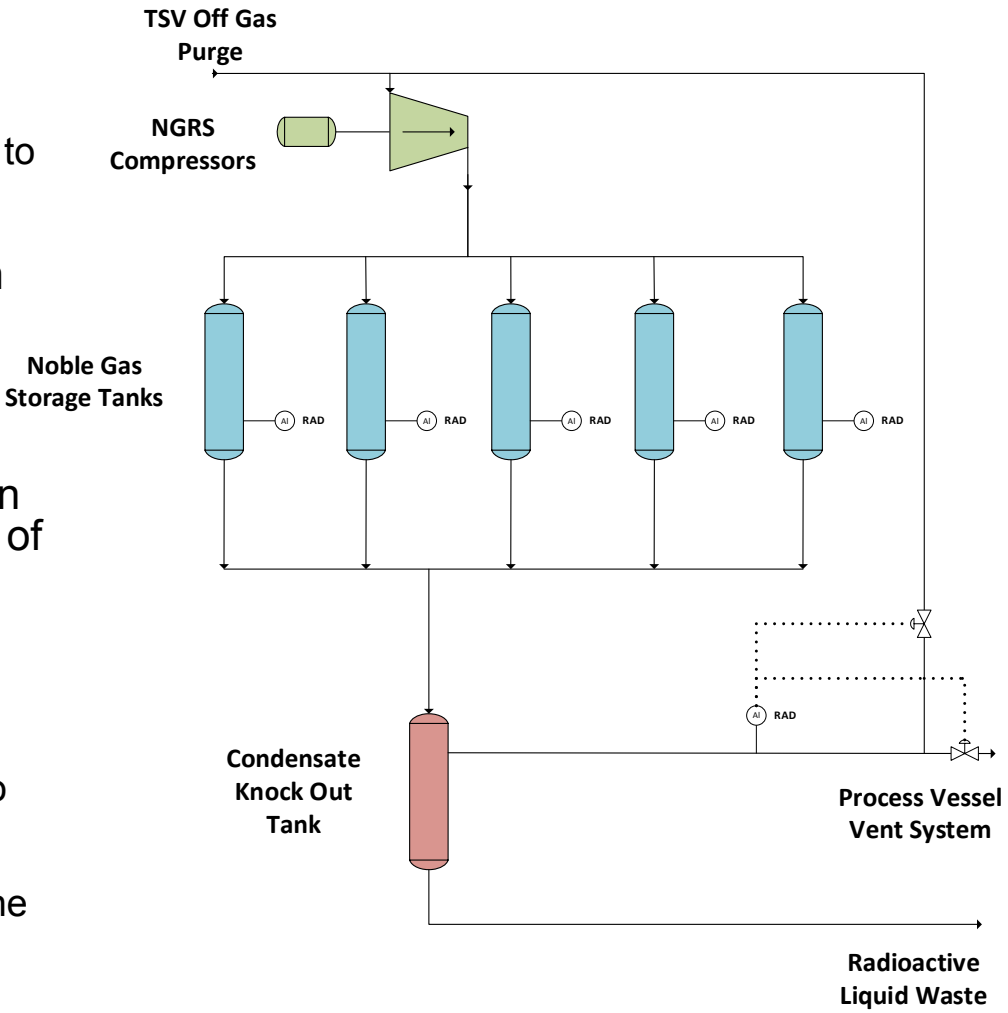
- An MHA was postulated in the RPF and IF
- The MHA is not required to be a credible event, and bounds credible events to establish an outer limit consequence
- The RPF includes:
  - Molybdenum extraction, purification, and packaging systems
  - Target solution cleanup systems (including uranium extraction (UREX) and thermal denitration)
  - Waste processing systems, including gaseous wastes
- The most limiting event was determined to be a simultaneous release of the inventory in the five noble gas removal system (NGRS) gas storage tanks

Maximum Hypothetical Accident

- The NGRS gas storage tanks collect and store radioactive gas from the Target Solution Vessel (TSV) off-gas system (TOGS), hold the gases for decay, and allow for monitoring of the gases prior to filtered release
  - Hydrogen in the off-gas is recombined in the TOGS system and is ensured below the acceptable limit before transfer to NGRS
- The five gas storage tanks are located in a reinforced concrete shielded cell
  - Penetrations and access doors are sealed to limit release of materials from the cell
  - Due to low pressure and cell construction, generated missiles would not be able to breach walls of cell
- NGRS is assumed to be at the maximum inventory at the time of the event
  - TOGS gases just transferred to NGRS
  - TSVs assumed to be operating at 110% of licensed power limit
  - The five NGRS tanks are filled with inventory from previous cycles

Maximum Hypothetical Accident

- The five noble gas decay tanks rupture simultaneously
  - Contents are instantly released to storage cell
- High radiation levels in exhaust ductwork detected by Radiation Air Monitoring System (RAMS)
- Radiological Integrated Control System (RICS) initiates alarm and cell isolation
- Redundant bubble-tight isolation dampers on the inlet and outlet of the cell close
  - Isolation dampers will be designed to close against postulated pressures
- Leak path factors:
  - 10% of the activity released into the cell assumed to bypass the isolation dampers
  - 10% of the activity leaks from the confinement area and exposes personnel



Maximum Hypothetical Accident

- The gases in the exhaust ductwork are passed through the Radiologically Controlled Area (RCA) Ventilation Zone 1 (RVZ1) filters (charcoal adsorbers and HEPA), but no reduction occurs
- Dose conversion factors used:
  - International Commission on Radiological Protection (ICRP) 30, Federal Guidance Report 12
- Worker evacuation within 10 minutes
  - Leakage from NGRS cell assumed to be instantaneous
  - Workers trained to immediately evacuate the area
- Airborne release fraction (ARF) and respirable fractions (RF) for noble gases are 1.0

Maximum Hypothetical Accident

- Dose consequences
  - Worker: 3.6 rem TEDE
    - On-site doses below 5 rem regulatory limit specified in 10 CFR 20.1201
  - Public: 0.082 rem TEDE (site boundary)
    - Public doses below 0.1 rem regulatory limit specified in 10 CFR 20.1301
- The MHA consequences are conservative
  - Five tanks assumed to simultaneously, instantaneously rupture with no mechanistic cause
  - 100% of the generated noble gas is assumed to leave the target solution and be transferred to the NGRS
  - The five NGRS tanks are completely filled, which is beyond planned operations
  - Isolation dampers in the RVZ1 exhaust ductwork downstream of the final filters also automatically close, but no credit is given to these dampers in the analysis

External Events

- Potential external events were evaluated as IEs for the SHINE facility
  - Design basis earthquake (DBE)
  - Tornado or high-winds
  - Aircraft crash into facility
- The SHINE production facility building is designed to survive postulated wind, tornado, seismic and aircraft crash loads
- Safety-related structures, systems, and components (SSCs) are analyzed under loading conditions of the DBE to ensure they can perform their safety function
- No consequences to the worker or public due to external events

Critical Equipment Malfunction

- This DBA analyzes malfunction or mishandling of equipment that could lead to loss of radiological control
  - Vessel/line/valve failures
  - Valve misalignments
  - Other process equipment failures
- Systems and components processing irradiated materials are located within shielded hot cells, process cells, tank vaults, or trenches
  - Major systems handling radioactive materials are:
    - Molybdenum extraction and purification system
    - UREX and thermal denitration subsystems
    - Waste treatment systems
    - Noble gas removal system

Critical Equipment Malfunction

- The limiting event was the inadvertent release of an NGRS storage tank due to a tank leak
  - Safety interlocks ensure that the appropriate decay time has elapsed prior to normal venting
  - Assumed leak results in releasing entire contents into noble gas storage cell
- Most conservative selection of tank and event timing:
  - Tank that experiences the leak is currently receiving new TOGS purge volumes
  - Tank just filled to capacity
  - Results in highest potential inventory of radionuclides in the tank

Critical Equipment Malfunction

- Event sequence
  - RCA ventilation and NGRS are operating normally prior to the event
  - Most recent TOGS purge volume just transferred to NGRS storage tank
  - Leak in storage tank assumed to instantaneously release the entire contents of tank to noble gas storage cell
  - High radiation levels detected by RAMS in RVZ1 exhaust ductwork
  - RICS initiates high radiation alarm and closure of bubble tight isolation dampers
  - Personnel evacuation of RCA occurs within 10 minutes

Critical Equipment Malfunction

- 10% of the airborne activity in the shielded cell assumed to leak out through penetrations, and 10% assumed to bypass the bubble-tight isolation dampers
  - Leak path factor calculations will be performed during detailed design to validate release fractions and ensure conservative values are used
- Dose consequences
  - Worker: 3.6 rem TEDE
  - Public: 0.082 rem TEDE (site boundary)
- Analysis is conservative:
  - 100% of noble gases assumed released from target solution
  - Release is complete and instantaneous
  - Isolation dampers in the RVZ1 exhaust ductwork downstream of the final filters also automatically close, but no credit is given to these dampers in the analysis

Inadvertent Nuclear Criticality in Radioisotope Production Facility

- This section of the accident analysis discusses
  - Where special nuclear material (SNM) may be located and where accidental criticality is possible
  - Implemented controls that reduce the likelihood
- Six main process areas involve handling SNM
  - Receipt of uranium and dissolution of metal
  - Dissolving uranium oxide in sulfuric acid
  - Transfer of target solution to TSV in IF
  - Transfer to RPF and extraction processes
  - Cleanup of irradiated solution
  - Conversion of uranyl nitrate to uranium oxide

Inadvertent Nuclear Criticality in Radioisotope Production Facility

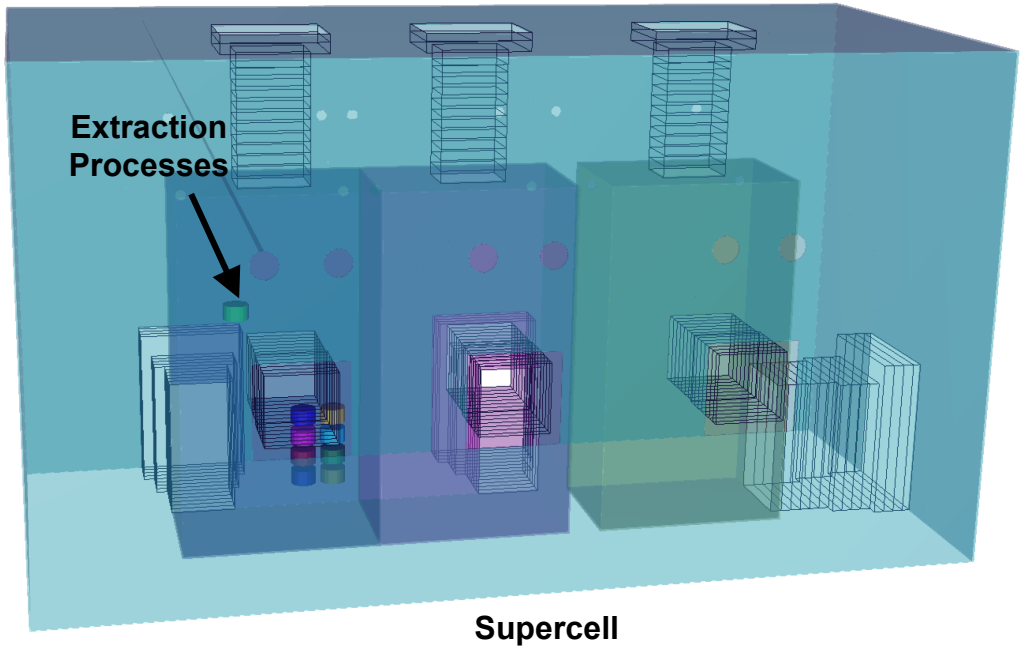
- Preliminary evaluation of scenarios that could lead to inadvertent nuclear criticality were evaluated during ISA
  - Leaks in piping or process equipment, accumulation of material, vessel overflows, and misdirection of material are some of the potential scenarios
  - Engineered controls and administrative controls have been identified based on this evaluation to ensure that each identified scenario is highly unlikely (see PSAR Table 13b.2.5-1)
- As described in Chapter 6b, nuclear criticality safety evaluations (NCSEs) will be performed with detailed design
  - Demonstration of double contingency protection will be made for each process
- SSCs to ensure criticality is highly unlikely will be safety-related

Radioisotope Production Facility Fire

- Fire initiating events have potential to damage safety-related SSCs within RPF and lead to radioactive release
- Fire events considered in ISA for normal and maintenance operations, within and outside of shielded process enclosures
  - Postulated fires from equipment malfunction, ignition of transient combustibles, loss of material control, propagation, and exothermic chemical reactions
- Assumed that small quantities of combustible materials are located in or near the SHINE processes (e.g., lube oil < 1 gallon)
- Hot cell fire detection and suppression system (HCFD) and facility fire detection and suppression system (FFPS) provide controls to reduce fire consequences
- Most limiting fire scenario determined to be a fire affecting the Mo eluate hold tank within the supercell

Radioisotope Production Facility Fire

- The design basis fire is assumed to occur during radiological process operations
  - A fire occurs inside of a supercell enclosure in the extraction portion
  - Hot cell fire detection is activated, alerting operations personnel
  - The hot cell ventilation is automatically isolated by the detection system interface
  - Hot cell fire suppression is not credited, but would be activated automatically or manually
  - Due to the thick radiation shielding of the cell, fire damage is limited to the hot cell interior
- RVZ1 exhaust filters release
  - HEPA filters remove 99% of particulates
  - Charcoal adsorbers remove 95% of halogens
- Dose consequences
  - Worker: 0.58 rem TEDE
  - Public: <0.001 rem TEDE (site boundary)



Accidents with Hazardous Chemicals Produced from Licensed Materials

- This DBA category evaluated hazards associated with chemicals produced from licensed material or that could affect the safety of licensed material
  - Chemicals are generally stored and used in small (<1000 lb) quantities
- Chemicals are used to support a variety of operations, including molybdenum extraction, target solution production, target solution cleanup, and waste processing
- The following potential initiating events were analyzed that could lead to releases of hazardous chemicals produced from licensed materials:
  - Failure of tanks/vessels, including associated piping components, due to mechanical failures
  - Failure of tanks/vessels due to fires inside and outside of tank vaults/cells
  - Exothermic chemical reactions
  - Spills of hazardous chemicals during handling
  - Unstable degradation products involving TBP and nitric acid

Accidents with Hazardous Chemicals Produced from Licensed Materials

- The analysis assumed that postulated IEs impact the entire inventory in a single location
- 11 chemicals were identified for further analysis based on toxicity, potential dispersibility, and inventory
  - Five factor formula used to determine source material released to environment
  - Releases modeled using EPIcode
- For the nitric acid and n-dodecane release scenarios, the bubble-tight dampers are credited with reducing the leak-path factor to 0.1
- Calculated chemical dose concentrations for the nearest resident are below PAC 1, 2, and 3 levels and worker concentrations are below PAC 2 values

Red Oil Event Prevention Features

Process	Expected Temperature	Nitric Acid Concentration	Organic Controls	Vent
UREX – extraction	25°C	0.1 M	Solvent control program to monitor impurities and TBP degradation products and replace solvent periodically	Hold and collection tanks ventilated Vents sized per DNFSB/TECH-33
UREX – scrub	25°C	0.3 M		
UREX – strip	50°C	0.01 M		
UREX – wash	50°C	0.01 M		
UREX – cold feed/flush	25°-50°C	1 M		
DNFSB/TECH-33 Recommendations	**<130°C**	**<10 M**	**Minimize impurities and degradation products**	**Size for potential red oil production**

Red Oil Event Prevention Features

Process	Expected Temperature	Nitric Acid Concentration	Organic Controls	Vent
Uranium metal dissolution	85°C	*Control not used*	Admin controls to prevent introduction Backflow prevention	Vents sized per DNFSB/TECH-33
Uranyl nitrate preparation	100°C	0.1 M		
UNCS evaporation	108°C	*Control not used*	Sampling prior to transfer	
UNCS denitration	*Control not used*	*Control not used*		
Liquid waste evaporation	108°C	Acids Neutralized		
DNFSB/TECH-33 Recommendations	**<130°C**	**<10 M**	**Minimize impurities and degradation products**	**Size for potential red oil production**

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Advisory Committee on Reactor Safeguards
Radiation Protection & Nuclear Materials Subcommittee
Meeting on SHINE Construction Permit Application

Chapter 13b Radioisotope Production Facility Accident Analysis

Mary Adams, Kevin Morrissey, James Hammelman, Technical Reviewers
U.S. Nuclear Regulatory Commission

September 22, 2015

Regulatory Basis and Acceptance Criteria

- Regulatory Requirements:
  - 10 CFR 50.34, “Contents of applications; technical information,” Paragraph (a), “Preliminary safety analysis report”
  - 10 CFR 50.35, “Issuance of Construction Permits”
- Acceptance Criteria
  - NUREG-1537 and ISG, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria”

Purpose of the Review

- To provide reasonable assurance that the proposed design of the SHINE Radioisotope Production Facility (RPF) has incorporated adequate capabilities and features to prevent or mitigate potential accidents and protect the health and safety of the public and workers

- Accident analyses defines the facility hazards that need to be protected against and that help support the establishment of the design basis

Review Procedures and Technical Evaluation

The staff performed a review of the technical information presented in Section 13b of the SHINE PSAR, as supplemented by responses to RAIs and information from other PSAR sections, to assess the sufficiency of the accident analysis for SHINE's RPF in support of the issuance of a construction permit. **The staff's evaluation included review of the following:**

- The ISA Team and team makeup used to perform the accident analysis
- The hazard evaluation process used to identify credible hazards and support the defining of the design basis of the facility
- The ISA methodology used to create accident sequences, estimate likelihoods and consequences, designate possible controls, and estimate the risk to workers and the public
- The identification and analysis of possible credible accident scenarios was complete
- The identification of safety controls (engineered safety features) needed to prevent or mitigate accidents to acceptable limits could provide reasonable assurance of safety

RPF Processes

- Receipt of uranium metal and dissolution
- Dissolving uranium oxide in sulfuric acid
- Transfer of solution to target solution vessel
- Transfer of irradiated solution back to the RPF
- Processing of irradiated solution via UREX
- Molybdenum extraction and purification
- Conversion of uranyl nitrate to uranium oxide

Accident Event Types

- Maximum Hypothetical Accident (MHA)
- External events
- Chemical accidents
- Radiological accidents
- Criticality accidents
- Fires

Maximum Hypothetical Accident (MHA)

- MHA bounds all radiological accidents

- For the RPF, the release of all inventory stored in the Noble Gas Removal System tanks is the MHA

External Events

- Seismic
- Tornado/High winds
- Aircraft impacts

Chemical Accidents

- Tank/vessel failures
  - Inside cells/vaults
  - Outside cells/vaults
- Exothermic reactions
- Handling errors

Radiological Accidents

- Loss of containment
  - Tanks/vessels
  - Pipes
- Overfills
- Mishandling
- Equipment malfunctions

Criticality Accidents

- Geometry changes
- Concentration
- Transfers
- Blockage of lines
- Loss of power
- Transients
- Dissolution

Fires

- Operational fires
- Maintenance fires
- Fire in other areas
- Lightning
- Leaks
- Vessel/tanks

Accident Analysis Review

- Reviewed postulated accident scenarios that are representative of the range of events that are possible in the RPF portion of the operating facility
- Reviewed the safety systems and defense in depth features of the design provided for the accident sequences
- Reviewed the design features needed for the prevention and mitigation of potential accidents

Review areas deferred to Operating FSAR

- Applicant needs to demonstrate that all accident sequences meet performance requirements
- Applicant needs to provide detail on safety controls and their safety functions needed for demonstration of acceptable risk for all accidents
- Applicant needs to provide detailed likelihoods and consequences for all accident sequences
- Applicant needs to provide detailed information on the management measures needed to support availability and reliability of safety controls.
- Applicant needs to provide the expected content of technical specifications for RPF safety related controls and detailed technical specification data
- Applicant needs to provide specific human actions versus the generic actions credited in the PSAR to prevent or mitigate accidents

Evaluation Findings and Conclusions

- The applicant has proved reasonable information on the performance and methodology used to evaluate accidents in the RPF portion of the SHINE facility

- The applicant has proposed and analyzed a set of accidents that should be representative of the possible range of events that may happen in an the RPF areas of the SHINE facility

- The analyzed set of accidents provides insights into the types and number of safety systems and safety features needed for the facility. The potential accidents might be prevented or mitigated by administrative controls, engineered safety systems, and trained personnel actions

- The staff concludes that the proposed preliminary accident analysis of the RPF and the preliminary safety design, including the engineered safety features, should, with reasonable assurance, protect the health and safety of workers and the public

Evaluation Findings and Conclusions (cont.)

Accordingly, SHINE has met the requirements of 10 CFR 50.35 for issuance of a construction permit:

- The applicant has proposed and analyzed a set of accidents that should be representative of the possible range of events that may happen in an operating facility and support the determination of the design basis for the facility
- Further more detailed technical, design, or analysis information may be reasonably left for later consideration in the FSAR to support operation of the facility
- The proposed facility and its structures, systems and components can be constructed without undue risk to the health and safety of the public and workers

Discussion

Consequence Calculations

- Dose to workers in facility
  - — Worker evacuation time assumed
- Doses to member of the public at the site boundary
- Uses 10CFR Part 20 for Dose limits
  - — 5 rem workers
  - — 0.1 rem off-site doses

Examples of Safety Controls

- Facility structure
- Process tanks and piping
- Hot cell structure
- Robust tanks and vessels
- Conduct of operations program
- Facility shielding
- Radiation area monitoring system
- Production facility biological shield system
- Noble gas removal system
- RVZ1 /RVZ2 systems (confinement/filtration)
- Radiological integrated control system
- Safe geometry overflow/radioactive drain systems
- Tank level detection
- Fire protection Program

Examples

Accident

Excessive time of process solution in the evaporator, creating increased concentrations and temperatures that promote formation of unstable compounds(e.g., reactions between nitric acid, Tri-Butyl Phosphate (TBP), and related decomposition products) that accumulate over time, resulting in an explosion

Controls

• Solvent Residence Time	Preventor
• Conduct of Operations Program	Preventor
• Process Tanks and Piping	Mitigator
• Zone 1 Ventilation	Mitigator
• Zone 2 Ventilation	Mitigator
• Radiation Area Monitoring System	Mitigator

Examples

Accident

- Operational fires where the fire is initiated inside the irradiation cell (irradiation cell is locked and closed)

Controls

• Combustible Loading Limits	Preventor
• Fire Protection Program	Preventor
• Irradiation Unit (fire rated)	Mitigator
• Dampers (Irradiation Cell) (Bubble type)	Mitigator
• Zone 1 Ventilation	Mitigator
• Zone 2 Ventilation	Mitigator
• Subcritical Assembly System (robust, includes TSV,	
• dump/hold up tanks)	Mitigator
• Process Vessel Vent System (filtered vent release)	Mitigator

Examples

Accident

- Dump Tank piping Leak Into the Irradiation Cell

Controls

- TSV Integrity, TSV Dump Tank Design (includes pipes and valves) — Preventor
- TSV Dump Tank Design (Dump Tank integrity, includes pipes and valves) — Preventor

- Shielded Pipe Trenches (includes sumps) — Mitigator
- Dampers (Irradiation Cell) (Bubble type) — Mitigator
- Irradiation Unit (cell confinement) — Mitigator
- Radiation Monitoring System in Irradiation Facility (monitors cell and cooling water) — Mitigator
- Zone 1 Ventilation — Mitigator

SHINE Response to ACRS Subcommittee Members' Questions

Catherine Kolb, SHINE

September 22, 2015

Structural Design

- The SHINE facility is designed per ASCE 7-05 for determining rain-on-snow surcharge loading:
  - SHINE is located in a 25 psf snow region, per ASCE 7-05, Figure 7-1
  - A rain-on-snow surcharge load of 5 psf is required only for locations where snow loading is 20 psf or less, but not zero, per ASCE 7-05, Section 7.10
- A snow load of 30 psf is used for the SHINE structural design for conservatism (since the dividing line between 25 psf and 30 psf is located in the same county as the facility in Figure 7-1)

Fire Hazards

- SHINE structures, systems, and components (SSCs) are designed in accordance with IEEE 384-2008, “Standard Criteria for Independence of Class 1E Equipment and Circuits”
- SHINE will perform the following as part of detailed design:
  - Evaluate the locations of Target Solution Vessel (TSV) Reactivity Protection System (TRPS) and Radiological Integrated Control System (RICS) components with respect to fire area designations
  - Ensure that electrical and control system train separation, including the consideration of fire hazards, is performed in accordance with applicable IEEE and NFPA standards

Nonsafety-Related Systems

- SHINE determined that those systems designated as nonsafety-related in PSAR Table 3.5-1 will be designed such that their operation or failure will not have an adverse impact on any safety function.
- Classifications of systems will be verified during detailed design when additional documentation is developed and when the final safety analysis is completed.
- Additional detail for the nonsafety-related systems will be placed in the FSAR based on the final design.

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Open Discussion Items

September 22, 2015

ACRS Subcommittee Follow-up Items

- Combined snow/rain load
- Designation of RVZ3 as non-safety-related for construction
- Treatment of defense-in-depth design against common mode failures
- Designation of FVZ4 as non-safety-related for construction
- Evaluation of facility chilled water supply and distribution system
- Fire area evaluation
- Irradiation facility accident analysis
- Review methodology
- Design changes