Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safegards
SHINE Subcommittee

Docket Number: (n/a)

Location: teleconference

Date: Wednesday, May 18, 2022

Work Order No.: NRC-1960 Pages 1-110

NEAL R. GROSS AND CO., INC.
Court Reporters and Transcribers
1716 14th Street, N.W.
Washington, D.C. 20009
(202) 234-4433

DISCLAIMER

UNITED STATES NUCLEAR REGULATORY COMMISSION'S ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The contents of this transcript of the proceeding of the United States Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, as reported herein, is a record of the discussions recorded at the meeting.

This transcript has not been reviewed, corrected, and edited, and it may contain inaccuracies.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

+ + + + +

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

SHINE SUBCOMMITTEE

+ + + + +

OPEN

+ + + + +

WEDNESDAY

MAY 18, 2022

+ + + + +

The Subcommittee met via Video Teleconference, at 8:30 a.m. EDT, Ronald Ballinger, Chairman, presiding.

SUBCOMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman

VICKI M. BIER, Member

CHARLES H. BROWN, JR. Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member

DAVID A. PETTI, Member

JOY L. REMPE, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

DENNIS BLEY

KEN CZERWINSKI

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

ALSO PRESENT:

MICHAEL BALAZIK, NRR

JEFFREY BARTELME, SHINE

JOSH BORROMEO, UNPL Branch Chief, NRR

MIKE CALL, NMSS

ELIJAH DICKSON, NRR

JAMES HAMMELMAN, NMSS

CATHERINE KOLB, SHINE

JEREMY MUNSON, NMSS

ALEXANDER NEWELL, SHINE

TRACY RADEL, SHINE

MICHAEL SALAY, RES

JOSEPH STAUDENMEIER, RES

DEREK WIDMAYER, ACRS

TABLE OF CONTENTS

P R O C E E D I N G S

8:30 a.m.

CHAIR BALLINGER: Good morning again, this is a meeting of this SHINE Subcommittee of the Advisory Committee on Reactor Safeguards. I'm Ron Ballinger, Chairman of the Subcommittee.

ACRS Members in attendance, I won't need to go through the list, it's the same as yesterday. We're basically reconvening the meeting that we started yesterday.

Today we'll cover Chapter 13 and we'll have a closed session after Chapter 13's open session on Chapter 13 and any other areas that we might need to discuss as we mentioned yesterday.

Today's meeting is held in person and also over Teams so Teams people, be careful, mute yourself at all times unless you're making a comment. If you make a comment, please state your name and make your comment when we go to public comments so the court reporter will know who you are.

Again, the transcript of the meeting is being kept. What else do we need to be careful about? That's about it. SHINE folks are up. Josh, do you want to make a comment of any kind?

MR. BORROMEO: I don't have any.

CHAIR BALLINGER: So, I think we're ready to roll. Is it Tracy that's going to present?

MS. RADEL: This is Tracy Radel. I'm going to go over the accident analysis today.

Some of this will look familiar and it was presented at an earlier ACRS session but I thought it was good to go over the methodology again, refresh that before diving into the specific accident scenarios in the closed session.

SHINE applies a SHINE-specific risk-based methodology similar to the guidance described in NUREG 1520. This is applied to both the DRH and the irradiation facility and the radioisotope production facility for consistency of the safety analysis across the facility.

The SHINE safety analysis is developed based on the following major steps. First, identification and systematic evaluation of hazards at the facility. This is generally done through HAZOPS and failure of the effects analyses.

Then the confluence of identification of potential accident sequences that would result in the next topical concerns.

The identification of safety-related controls, other controls as well as administrative

controls, the identification of programmatic administrative controls that intrude on the availability and reliability of the identified safety systems, and then the assessment of radiological and chemical consequences for the postulated accident sequences given the applied controls, demonstrating that all scenarios are within acceptable limits.

CHAIR BALLINGER: This is Ron Ballinger. Failure modes and effects analysis is a term of art. Did you folks actually perform an FMEA, a formal one?

MS. RADEL: We have performed an FMEA and it depends on the system that we're looking at so the appropriate method was chosen based on the system. So, those were either HAZOPS, FMEAs, or what-if checklists and scenarios.

CHAIR BALLINGER: Thank you.

MS. RADEL: The acceptance criteria here, acceptable risk is achieved by ensuring the event is highly unlikely or the consequences are below the SHINE safety criteria.

The SHINE safety criteria is listed here and they include an acute worker dose of 5 rem or greater, an acute dose of 1 rem or greater to any individual located outside the owner-controlled area.

An intake of 30 millirem or greater of

uranium to any individual located outside the owner-controlled area, and acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could lead to irreversible or other serious long-lasting health effects to a worker.

Or it could cause transient health effects to any individual located outside the owner-controlled area. Criticality where fissionable material is used, handled, or stored with the exception of the solution vessel or loss of capability to reach safe shutdown conditions.

The relevant accident categories were identified using the ISG-augmenting NUREG-1537. Hazard evaluations identified potential initiating event consequences and controls. These were HAZOPS, FMEAs, or what-ifs.

The hazard evaluations also identify the SHINE-specific accident types such as those in tritium or the neutron driver. The hazards that are identified through the different analyses are summarized in the process hazard analysis.

DR. BLEY: May I interrupt you? This is Dennis Bley. yesterday you folks talked about a lot about how unique this facility is and I think I

certainly agree with you.

Why, then, is just using the guidance for more standard facilities a complete enough search for accident categories?

MS. RADEL: So, we evaluated the systems for all failures, we identified that through the HAZOPS, FMEAs and what-if checklist, going through each component, its potential failure mechanisms.

We didn't limit ourselves in any way in those hazard evaluations, just as we were rolling those out into accident categories, we defined the accident categories. Those were all rolled up into NUREG-1537.

DR. BLEY: That helps, I'll let you go ahead but one thing we hinted at I think a while back, and I don't recall seeing anything that clarifies it, when you did this search using HAZOPS and other approaches, did you identify specific important human actions?

I was looking for a table of those and how you identified them.

MS. RADEL: There are administrative controls identified within the safety analysis and we can go into some specific ones when we get into closed session and go through the different accident

scenarios that were evaluated.

DR. BLEY: I would appreciate that and I'm oversimplifying what you just said back to me, it would seem strange to me if every important human action was handled strictly by administrative controls but I can wait for the closed session to hear about that.

MS. RADEL: The process hazard analysis is a summary documenting the most significant events from the HAZOPS, FMEAs and what-if checklist, and this is documented in the SHINE safety analysis report.

The risk index for each potential unmitigated accident sequence is provided there. This is the likelihood times the consequence. It identifies engineered and administrative controls.

The risk for the controlled event is generated using the revised likelihood and consequence of the controls in place. And this then results in a comprehensive list of the safety-related controls for the facility, both administrative and engineered.

MEMBER REMPE: This is Joy Rempe and I had a question. First of all, I appreciated the Staff identifying some of the documents where we could find more details about the actual approach to perform the safety analysis.

Again, you can answer this in the closed session but I was curious about -- I also appreciate you guys providing those documents when I requested them, or at least access to them.

But what I didn't see was much about the sensitivity analysis you did that were referenced by the Staff that gave them confidence.

At a high level here in the open session, could you talk about extensive the sensitivity analyses were and what assumptions were found to have the most impact in the results?

MS. RADEL: Just for clarification, are you talking about sensitivity analyses related to activity assertions or just in general?

MEMBER REMPE: All of the safety analyses, and when we get to the closed session I can even point out which sequences the Staff referenced where they talk about the sensitivity analyses gave them confidence in the results.

But what I'm curious about is I thought a long time ago I had seen something from you guys saying that you did do some sensitivity analyses.

I don't think you did a full-fledged uncertainty analyses where you propagated the uncertainties through the analysis and you can confirm

that, but I didn't see anything that suggested that.

But if you did some sensitivity analyses I'd like to know what parameters were evaluated. I think the Staff thought you picked the most conservative values but was that the only scenario where you did sensitivity analyses, the criticality?

I know for sure we mentioned it with respect to the design basis accidents that you analyzed. So, anyway, I just am curious if you could give us any sort of insight about how extensive your sensitivity analyses were?

MS. RADEL: Yes, we did perform extensive uncertainty analysis on the different parameters related to reactivity, coefficients, productivity coefficients, the delayed neutron fraction.

Pretty much all of the different reactivity parameters used benchmark cases to come up with 95 percent confidence intervals for those important parameters.

When we went to go evaluate specific scenarios within the accident analysis that were more along the lines of the releases, due to failures in the process boundaries or events that resulted in radiological release, the approach there was to take a very conservative bounding approach to each of the

different parameters in the scenario.

And you'll see that we get into the specific scenario descriptions within the closed session. But for each of the different aspects of the accident, we looked at what would be the most bounding situation even if they were quite a bit outside of the normal operating parameters.

MEMBER REMPE: Let me repeat back to you, and I think I heard you say you only did sensitivity analysis with respect to the criticality events that you evaluated.

And then with respect to the release fractions, you didn't really do sensitivity analysis, you just looked at it and said we're taking some bounding assumptions. Is that true?

MS. RADEL: Yes, that's correct. Just to clarify the reactivity insertions, those were not reaching criticality but similar.

MEMBER REMPE: Reactivity insertions, thank you for the correction. Could you tell us what parameters you found to be the most sensitive in the sensitivity analyses you did?

MS. RADEL: In the closed session we can go into the uncertainties that we found on those and the values themselves and talk through those numbers.

Both are proprietary.

MEMBER REMPE: Thank you.

MS. RADEL: This slide here shows the different accident categories for the radiation facility and the radioisotope production facility as well as external events.

These categories primarily come out of NUREG-1537 but extend as I said, we do have facility-specific events and that covers anything that didn't fit into those accident categories. So, we make sure that everything is captured.

MEMBER MARCH-LEUBA: This is Jose. In the open session, can you describe at a high level what the MHA was?

MS. RADEL: Yes, the maximum hypothetical accident was a release into the TSV off-gas cell. So we release out of that gaseous support system for the irradiation units.

MEMBER MARCH-LEUBA: So, it would be similar to a large-break LOCA in the gas area of the TSV?

MS. RADEL: It is a break of the system.

MEMBER MARCH-LEUBA: There is no loss of coolant. So, it's a big break on top of the vessel. Thank you.

MS. RADEL: Here are the risk matrices that we have for development of those numbers within the SHINE safety analysis, so on the upper left there are the consequence categories.

These are similar to 1520 but we do adjust the very routine low-consequence and intermediate consequence to align with the SHINE safety criteria. So, low consequence meets the SHINE safety criteria.

On the right there is the severity, we do either require that events are highly unlikely, essentially preventing the event or of low consequence with the mitigations in place. We do not have an intermediate unlikely event being acceptable.

The other values there, the highly unlikely is defined as 10 to the -5 per event per year.

MEMBER DIMITRIJEVIC: This is Vesna Dimitrijevic, let me ask you a question on the risk matrix. Can you hear me?

MS. RADEL: Yes.

MEMBER DIMITRIJEVIC: You have adjusted this medium Category 4 to be unacceptable and modified risk matrix but in that case you don't really need the three categories.

You can simplify your calculation by

combining the 2 and 3 vote in likelihood and consequences.

Once when you make this change, the vision to the three categories is not necessary. So, I'm not sure, why did you decide to make this change?

MS. RADEL: That's a very good point, that we wouldn't need that extra row. At one point early on in our development, we did have that four box as being acceptable. But we made a change later on to require that you reach either highly unlikely or below the SHINE safety criteria.

MEMBER DIMITRIJEVIC: I just want to point out that in that case, you just overly complicated the vision. The vision could be significantly simplified if you wanted do it this way. I just wanted to make that point.

MS. RADEL: Thank you.

MEMBER REMPE: Just a little curiosity question, why not call not unlikely anticipated operational -- I always kind of stumble on it when I read the phrase not unlikely.

MS. RADEL: Yes, that language does come out of NUREG-1520 so we just used the consistent language there.

MEMBER REMPE: Thank you.

MEMBER MARCH-LEUBA: Plus it also has 10 to the -2. This is 10 to the -4.

MEMBER REMPE: Not unlikely.

MEMBER DIMITRIJEVIC: They did forget to say likely.

MEMBER REMPE: This one comes across.

MEMBER MARCH-LEUBA: Once every 10,000 years is not likely.

MEMBER PETTI: There should be better alignment.

MEMBER DIMITRIJEVIC: Less unlikely.

MEMBER REMPE: I like that, Vesna, I like less likely, something like that, anyway.

MS. RADEL: Next slide. Here we have the guidance that we used as far as the failure frequency index numbers, the failure probability index numbers and the duration index numbers that were used to get the likelihoods.

Safety-related controls that are credited for the prevention or mitigation of accidents are either engineered controls, and this can be either active or passive, and then specific administrative controls.

Programmatic administrative controls are also implemented to assure the safety-related controls

can perform their intended functions, and defense in-depth controls are also identified.

They're not credited but they provide additional margin for risk reduction.

MEMBER KIRCHNER: This is Walt Kirchner. Are there any categories that stand out -- it's a variation on Dennis's question I think -- in your accidents where a significant release is prevented by administrative or programmatic controls?

MS. RADEL: We do not have any scenarios where operator action is credited to mitigate the event once it has occurred. So, we've relied on engineer controls for all of that mitigation once an accident has initiated.

There are some administrative controls to contribute the prevention for reduction in source term prior to new events occurring that are outlined. For the most part, though, the controls are engineered controls.

MEMBER KIRCHNER: That's the answer I was looking for. Thank you.

MEMBER DIMITRIJEVIC: Let me ask you a question to follow this. You have open attachments, you credited the elevating likelihood, so you have a serious screen based on assumed operator actions,

right?

MS. RADEL: They're accredited operator actions, yes, for prevention of scenarios.

MEMBER DIMITRIJEVIC: So, you're looking in scenarios, not just initiating events? They actually operator actions with each other, complementing to the screening out some scenarios, right?

MS. RADEL: Yes, that's correct.

MEMBER DIMITRIJEVIC: My question is why you didn't really consider that to be an operator action, do you know what I mean? You varied the likelihood of this.

MS. RADEL: Those are operator actions that are accredited are specifically called out in the safety analysis and those accredited operator actions are treated differently in the procedures and noted that those are accredited actions.

They are given a low value as far as the reduction in likelihoods. The administrative controls get less credit in the likelihood evaluation because we know that humans are less reliable than the engineered controls.

MEMBER DIMITRIJEVIC: My point was that you just responded to the previous question saying

that no operator action crediting mitigating but that's a true statement because they're buried in the prevention.

So, they operator action credited in preventing the accidents.

MS. RADEL: Correct, in preventing. In mitigation, once the accident has begun, the statement made was that operator action is not credited after the event has occurred to mitigate the consequences.

MEMBER DIMITRIJEVIC: Thanks.

MS. RADEL: Radiological consequences are determined for members of the public and control room operators.

The process includes calculation of radiological inventories, definition of accident-specific material at risk, transport of the radionuclides, which leads to development of the accident source term, and then the convergent radiological dose.

Worker and public doses are generally calculated over a 30-day period. The exception here is tritium and the tritium confinement boundary which uses a 10-day interval for the accident duration.

Conservatisms are applied to the dose analysis. There's a list here of some of them that

apply across all scenarios. The bounding TSV power history and operational cycle is used.

We used minimum nuclide decay times, times to transport nuclides out of the process systems are neglected so the inventory is immediately transferred out of the tank or the piping into the confinement if there is a breach involved in the scenario.

Compensation is conservative and limited as far as radionuclides depositing on the walls versus remaining airborne. Non-credited filtration is neglected.

There are a few filters in the system on the outlet of the hot cells that are credited but for the most part, the filtration within the facility is not credited and is neglected.

MEMBER MARCH-LEUBA: This is Jose, how do you treat the confinement, is it leakage tray or does it fail completely?

MS. RADEL: The confinement has a defined leak rate, each confinement has its own defined leak rate and we make sure that it meets that leak rate prior to going into operation.

MEMBER MARCH-LEUBA: So, you use the nominal leak rate that you assume for confinement?

MS. RADEL: We use a leak rate based on

each confinement so it's not the same for, say, the IU cell versus the hot cell boxes. It's defined based on which --

MEMBER MARCH-LEUBA: How is that leak rate estimated?

MS. RADEL: It's defined in the requirement specifications for the different facility components and then it is tested as part of the testing as a safety function of that component.

MEMBER MARCH-LEUBA: That's what I wanted to hear. After you build the facility you will test for the leak rate?

MS. RADEL: Yes.

MEMBER KIRCHNER: Just to follow on, then, those numbers are used for the leak path factor? This is Walt Kirchner.

MS. RADEL: Yes, there's a leakage rate that is used within the radiological dose calculations. That leakage rate was determined in those calculations applied to the equipment. It was chosen based on what we knew that type of equipment could achieve.

And it was specified for the equipment when we tested prior to starting operations.

MEMBER REMPE: This is Joy, I just wanted

to add a comment here.

I really like the level of effort you went to with this Fowski report, where you actually went through and estimated leak factors and airborne release fractions based on available data and actually applied the model to other facilities.

I thought that was a level of effort I had not yet seen with some of the other folks that are coming in with unique facilities. And so I think a kudos are in order.

MEMBER PETTI: So, Tracy, just a question, a clarification, on the condensation. For tritium, do you assume it's HT or HTO?

MR. NEWELL: This is Alex Newell, the criticality safety lead.

For transports of material in that model, it's assumed to be a gas but because the dose conversion factor for tritiated water is higher, we assume that the tritium exits the facilities in that form.

MEMBER PETTI: So, you don't condense it, per se, you assume it's a gas that doesn't condense and then from a dose perspective, you use the higher dose conversion of HTO.

MR. NEWELL: That's correct.

MEMBER PETTI: Thanks.

MS. RADEL: And we use chi over Q values that are the 95th percentile. This diagram here shows the process of determining the radiological consequences, it's a lot to go through so adjust it. If you have any questions, I can answer those now.

MEMBER PETTI: Tracy, I don't recall from when I read this the statistical adjustment. What is that in the inventory?

MS. RADEL: We did do a statistical analysis on the source terms. It was a very small effect and it's another conservatism that I didn't list in the previous slide because there's a small change from the other ones.

MEMBER PETTI: Thanks.

MS. RADEL: The last slide here is just touching on the hazard chemicals.

Chemical hazards of licensed material, hazardous chemicals interacting with licensed material and hazardous chemicals from licensed material are evaluated in the SHINE safety analysis.

Chemical consequence assessments demonstrate that consequences meet the SHINE safety criteria for the public and the workers. And here the workers include both the RCA worker and the control

room operator.

DR. BLEY: Tracy, it's Dennis Bley. Could you go back one slide to your diagram? Down in the bottom, you're doing a public dose and a control room dose. Where do you do the workers who are not in the control room?

(Simultaneous Speaking.)

MS. RADEL: We do evaluate the RCA worker dose also within our dose calculation but it was not something that was required from a regulatory perspective to be provided. But we do calculate it and verify that it is all for below the SHINE safety criteria.

DR. BLEY: Thank you, and I guess you're showing what's required on here so I get that. I'm glad you're doing the other one though, because it's got to be higher.

MS. RADEL: It depends on the scenario, there are some scenarios where because it's more of an external release, the control room dose is higher than the RCA workers' dose. But the residence time within the RCA is fairly short versus the time spent in the control room.

DR. BLEY: Thank you.

MS. RADEL: The PAVAN computer code is

used to perform consequence analysis for the public and nearest resident and consistent with the accident analysis overall, we used the 95th percentile chi over Q values.

MEMBER PETTI: What code was used to do the control room dose?

MS. RADEL: Can you repeat the question?

MEMBER PETTI: What code was used to do the control room dose?

MR. NEWELL: We did the wrong calculation. This is Alex Newell, we used ARCON96 to calculated the control room chi over Q.

MEMBER KIRCHNER: This is Walt, just one question on chemical exposures. I'm trying to think of the different chemicals you're using in your processes. Do you have any ground-huggers, so to speak?

In other words, heavier than air, chemicals that are part of your processes?

MS. RADEL: As far as those that are mixed with licensed material, they were evaluated in Chapter 13. I would have to look through the list, it's provided in the closed session slides.

I don't know offhand of any that were denser than air that presented that.

MEMBER KIRCHNER: Because using the chi over Q doesn't make any sense if you've got chemicals that are heavier, that's the plume-type model. If you've got something like CO2 that's a ground-hugger, then it's a different kind of analysis to account for that.

MS. RADEL: I don't believe any of the ones on that list provided at the end of the closed session slides were of that type but we can verify that.

MEMBER KIRCHNER: That's good, thank you.

MEMBER BIER: Are you done with your comments on this slide, Tracy?

MS. RADEL: This is the last slide, so if you have any questions for open session here, we can take those. Otherwise, we are clear for the closed session.

MEMBER BIER: This is Vicki Bier.

I wanted to go back to the risk matrix slide if you can, I forget what number that is? I wanted to raise an issue, not because I object to using risk matrices but I just want to illustrate a potential limitation of them.

Which is that in theory, there can be events in the green area that are worse than events in

the orange area, and that's because these bounds are I don't want to say broad but they're not points.

So, if you take, for example, the middle row and the dividing line between green and orange, you could have an event in the green that is close to the maximum dose, say, close to 100 rem for facility staff.

But a low likelihood within that range of likelihood -- sorry, high likelihood within that range of likelihood. So, high on both so you can think of it as being in the upper right-hand corner of the green square, I wish we could annotate these.

And then in the orange square, the one with the rating of 4, you can get an event that is very low within the range of dose, say, 6 rem to facility staff and low within the band of likelihood, just barely above 10 to the -5.

And so the green event could be almost a factor of 15 worse than the orange event. Like I said, I don't object to using risk matrices as long as people realize they're a pretty coarse tool.

But if interested, a colleague wrote a paper on the pitfalls of risk matrices so I could share that with the DFO to pass onto you if people are interested, et cetera.

MS. RADEL: That's a very good point.

MEMBER BIER: It's just awareness.

MS. RADEL: That would be great.

Some of the challenge here is defining the likelihoods for the facility that has not been built before, is new, and so we take conservative approaches to assigning those likelihood numbers and take a more simplified approach here.

Any other questions?

CHAIR BALLINGER: Is that the end, Tracy

MS. RADEL: Yes, it's the end of the open session presentation.

CHAIR BALLINGER: Thank you, if there are any comments from the Members before we switch to the Staff? Hearing none, okay, there is the Staff presentation. Who is the presenter?

MR. DICKSON: Hi there, this is Elijah Dickson along with my colleagues, doing the Chapter 13 accident analysis. Can you hear me okay? They're in the meeting room.

CHAIR BALLINGER: Yes.

MR. DICKSON: Good morning, ACRS Chairman and Members, my name is Elijah Dickson, I'm a reactor scientist in the Office of Nuclear Reactor Regulation, Division of Risk Assessment, Radiation Protection and

Consequence Branch.

I'd also like to thank Tracy for providing the overview of SHINE's perspective of SHINE's accident analysis. I'll be presenting, along with my colleagues our review of the Chapter 13 accident analysis today. Next slide, please.

This was very much a team effort with technical expertise across several NRC organizations. My organization, the Office of Nuclear Reactor Regulation, NRR, but also the Office of Nuclear Materials and Safety Safeguards, NMSS, and Office of Research.

I'll be presenting with Mike Call today, who will be presenting on the SHINE safety analysis, and James Hammelman, who will be presenting specifically on the SHINE chem safety analysis.

Jeremy Munson you heard yesterday speaking in regard to criticality safety, Kevin Quinlan is our meteorologist, Mike Salay from Office of Research performed or helped review iodine transport calculations and models.

Can we move on to the next slide, please?

Contents. I'll provide a little bit of a background of our review and approach to Chapter 13 accident analysis. We'll then dive right into the

SHINE SSA or SHINE safety analysis. I'll speak to SHINE's design basis accident analyses, and again, right on into chem safety by James Hammelman.

We'll discuss what we learned from our audits and confirmatory analyses, and then finally present to you our evaluation findings and conclusions. Next slide, please.

Here's a list of regulatory requirements and commitments, 10 CFR 5034 of course has SHINE present to us a final safety analysis report for our review.

5036 and the technical specifications help us identify safety-related structured systems and components for SSCs, their safety limits, and their limiting safety system settings, or LSSSs, as well as those limiting conditions of operations.

And in doing the dose analyses, we particularly focus on the first three criterion of the technical specifications, Criterion 1 includes tech specs for instrumentation that are used to detect and indicate in the control room significant abnormalities or degradations in the reactor coolant pressure boundary.

Criterion 2 are process variables, design features and operational restrictions that are initial

conditions of a design basis accident or transient, that either assumes failure or presents a challenge to the integrity of the primary fission product barrier.

And then lastly, Criterion C are structures, systems, and components that are part of the primary success pathway, which actually actuate or mitigate design basis accidents that either assume a failure or present a challenge to the integrity of the primary fission product barrier.

We also have commitments that are similar to those of 10 CFR Part 70. Jeremy Munson today discussed what those commitments were.

Again, you can find those in Tech specification 5.8.3, these are or additional reporting requirements since much of the SHINE facility is very similar to a materials facility.

Next slide, please. I'll go over the regulatory guidance. There is quite a bit of regulatory guidance that the Staff needs to utilize to perform such a review. The two primary guidances are NUREG-1537 Parts 1 and 2.

Part 1 is the guidance preparing reviewing applications for licensing of non-power reactors. This is the format and content guide. Part 2 is the standard review plan and acceptance criteria that the

Staff use.

Now, for both Part 1 and Part 2 of NUREG-1537, there is final interim staff guidance that's specific to the licensing of radioisotope production facilities and for aqueous homogeneous reactors.

Part 2 was leaned on heavily for the information regarding our standard review plan and our acceptance criteria. Onto Slide 6, please?

Continuation of regulatory guidance, NUREG-1520 is the standard review plan for fuel cycle facilities and licensing applications. This was the primarily guidance that NMSS utilized which then points you to NUREG-1513 which is the integrated safety analysis guidance document.

This document was utilized to review SHINE's safety analysis summary.

NUREG CR-6410, the nuclear fuel cycle facility accident analysis handbook, was very important for the Staff in performing the material of risk calculations and transport of the source term through the systems into the environment.

It provides methodologies to do those calculations and then the last three bullets are in regard to reviewing atmospheric and meteorological characteristics of the site.

NUREG CR-2858, Regulatory Guide 1.145, and NUREG 6331.

MEMBER PETTI: Elijah, just a question. You've got mixed guidance here coming from the non-power reactors in the fuel cycle. Was there any case where the guidance was contradictory or was it just more additive or explanatory?

This is kind of a unique case but it would be interesting to know if you guys had to make some decisions, like one guide said X and the other guide said anti-X and you had to figure out what made sense.

MR. DICKSON: From the Staff perspective, I think we all probably have a short list of where there might be some contradictions between the different guidances. But working together, we're able to work through some of those types of issues.

There's a couple areas of the guidance that could use some improvement, or rather, clarification really.

MEMBER PETTI: That's good to know because I think we're going to see cases where we have unique facilities and the rules may not always fit and separate so nicely into these buckets.

MR. DICKSON: One, for instance, is the use of dose conversion factors that align with the

regulations, that's something that can probably be spelled out pretty clear in regulatory guidance.

There's a whole host of dose conversion factors, or DCS, out there that are designed for different ICRP methodologies, the different tissue-weighting coefficients, and in pointing licensees to the ones that we've endorsed that meet the regulation, it's very much like a nuanced issue that we see every couple of years, for instance.

That could be an improvement for the guidance.

DR. BLEY: Elijah, I'm following up on that. How or who on Staff keeps track of these sorts of things so they either end up in interim Staff guidance or in changed guidance documents?

MR. DICKSON: There's a Branch in the Division of Advanced Reactors who are owners of our guidance and I believe they keep track of these items as they seek to update the guidance that are updated every 10 years.

DR. BLEY: So, each of you, as you were saying, found different things. I assume each of you then passed those onto the people who own the guidance?

MR. DICKSON: Yes, that's definitely the

intent.

MEMBER MARCH-LEUBA: Basically, there is a suggestion box at the door and you fill in a card or you're supposed to know the guy and go to lunch with him? How does that work?

MR. DICKSON: There's no suggestion box. I know the guys that work on these things and I tell them these things when I come across them.

But usually, when we do develop important documents like this, we do go through internal convergence processes and that's also a time where things like this get identified when we go through those internal concurrence processes between divisions and offices.

MEMBER MARCH-LEUBA: Just to give you a heads-up, on Slide 12 I was going to ask you to go over the acceptance criteria, the 1 to 5 rem that you guys chose.

And the SCR goes into very good detail of all the criteria you could have used and I would like you to explain what your rationale was and is related to this.

There is a little confusion that you can pick A, B, or C and it's up to the licensee to pick. So, I'll ask you the question again on Slide 12.

MR. DICKSON: Great.

MEMBER HALNON: Elijah, this is Greg Halnon, I want to look at this list of guidance, all but 1 to 20 years-plus old, some are 40 years old. Part of me says that's great because they're tried and true and if they needed a revision, there would be one.

However, part of me thinks there's got to be some learnings that we've had over the last 40 years that would assist in this review that would require some updated guidance.

Is there other ISGs or other more contemporary guidance that you're using behind the scenes here?

MR. DICKSON: For a facility such as SHINE or these non-power reactor facilities or NPUFs, NUREG-1537 is and the ISGs are the primary guidance that we have right now.

As you mentioned, the meteorological guidance, that's tried and true calcium plume atmospheric dispersion modeling. But there's other NUREGs that we utilized too for iodine transport and things of that nature.

And I think Mike Salay could probably talk to those types of guidance, too, that could get

wrapped up into these as well. But you asked for other documents to be used, there are a handful of other ones as well.

MEMBER HALNON: For instance, which one of these or what other Reg Guides were used for the ARCON96? That's just recently been updated in 4.28.

MR. DICKSON: I guess that would have to get captured in an update to NUREG-1537.

MEMBER HALNON: I didn't see it in there, I'll look.

DR. BLEY: Speaking of NUREG-1537, this is Dennis again, my understanding of interim Staff guidance is the Staff does that because there isn't time to go through the process of updating NUREGs or SRPs or whatever.

But the interim Staff guidance for this one is 10 years old. How long do things live as interim Staff guidance? I've seen some in the electrical area that live forever.

Maybe somebody wants to comment?

MR. DICKSON: I don't have a good answer for that one but I know that we are actively working on updating this guidance.

(Simultaneous Speaking.)

MR. BALAZIK: I'm sorry, Elijah, to cut

you off. This is Mike Balazik, Project Manager for SHINE. Yes, we plan on incorporating the ISG into NUREG-1537.

The ISG, and I'll use the term relatively recent, we developed that guidance for the review of production facilities, which wasn't captured in 1537.

We do plan on incorporating it within the overall 1537 guidance.

DR. BLEY: Thank you.

MEMBER KIRCHNER: Just one question, Mike. Someone made the suggestion of a suggestion box or something but in the end doesn't RES own the engineering division, own the responsibility for the Reg Guide, updating and such?

Do they have an inbox for all these observations that your Staff makes when they conduct a review like this and they see potential discrepancies or alternatives or confusion.

Does that get fed back in some formal manner to engineering?

MR. BALAZIK: This is Mike Balazik again, no, our licensing branch actually has the responsibility for NUREG-1537 and we would collect lessons learned from the SHINE review. We've collected them from past RTR license renewal efforts.

That's how we would incorporate any changes into 1537. So, I would say it's not necessarily a formal process on providing feedback.

MEMBER KIRCHNER: I may have misspoken. I guess now, more correctly, engineering owns the Regulatory Guide process, if I understand it correctly, and you're mainly using a NUREG as the basis for your reviews.

So, you own, so to speak, 1537?

MR. BALAZIK: Yes, sir, and the ISG.

MEMBER KIRCHNER: Thank you.

MR. DICKSON: Any other questions on regulatory guidance? If not, we can move on to Slide 7.

SHINE presented to us two types of safety analysis for Staff review, the first being the SHINE safety analysis or SSA, the purpose of the SHINE safety analysis is systematic analysis of facility processes used to identify and evaluate facility hazards associated with the processing and possession of licensed material.

And then the design basis accident analysis, the purpose is to evaluate the design and performance of structures, systems, and components of the facility with the objective of assessing

radiological consequences resulting from operation of the facility.

Onto the next slide --

MEMBER REMPE: This is Joy, I'm going to reiterate a question that Member Ballinger has raised several times during this review.

In the SE, the Staff indicated they found the application of SHINE's design criteria that was discussed in Chapter 3 reflect the design features of the safety-related SSCs.

And it goes on about, which include redundancy, environmental qualification, et cetera. What gives you that confidence? Because we haven't seen your review of the design criteria.

Can you elaborate? We're kind of wondering about that issue.

MR. DICKSON: Yes, I can definitely do that. For Chapter 13, the primary purpose is to assess radiological consequences.

And the primary design criteria that SHINE presented to us that assesses radiological consequences is Design Criterion 6, which is their control room habitability criteria.

We assume, and the guidance tells us to assume, that all other design criteria are being met

when we're performing the dose analyses.

I don't know if I answered your question or not but we don't go line by line in the Chapter 13 analyses, or design criteria by design criteria in the Chapter 13 analyses.

We're specifically assessing radiological consequences.

MEMBER REMPE: That helps, thank you.

MEMBER KIRCHNER: When you do that, then at least in the DBA category, now you only can credit those SSCs that are safety-related?

MR. DICKSON: That's right, and I have a slide on that. I receive some questions from the ACRS and I've littered responses in this presentation and that was one of the questions that I received, what is the design basis accident methodology from a very high level?

And I'll talk about that. If there's no other questions, we can go onto Slide 8, and I believe this will be Mike Call. Mike Call, are you on the line?

MR. CALL: Yes, can you hear me?

MR. DICKSON: Yes, thank you.

MR. CALL: Good morning, this is Mike Call, as mentioned earlier, I'm in the Office of NMSS

and was involved in the SSA review.

As you can see on the slide here, and has been mentioned and presented by SHINE earlier, this is one of their approaches to evaluating the accidents for the facility.

In terms of our approach to looking at that in NUREG-1537 and particularly, the ISG augmenting the NUREG, it recognizes that the ISG methodologies and performance criteria and implementation of management measures is an acceptable approach to demonstrating and ensuring safety, though it does allow for alternatives if those are also demonstrated to ensure adequate protection of safety as well.

So, that was the approach. It's very general in terms of how the ISG approaches that. So, those methodologies are captured in Subpart 8 or Part 70 and NUREG-1520, which typically apply to your fuel cycle facilities.

As was discussed earlier by SHINE, they make use of these types of methodologies but there are differences. We can probably go into a little bit more of that if necessary.

You can see some of the differences listed here such as terminology. Instead of having an ISA,

an integrated safety analysis, they have a SHINE safety analysis.

In terms of Part 70, you typically have things referred to as items relied on for safety or IROFS, which can be administrative or engineering.

And for SHINE they have what they refer to as safety-related controls and similar to Part 70 management measures, they have what they call programmatic administrative controls or reliability management measures.

In terms of the content, the purpose, the function, the kinds of analyses, they're very similar. Another interesting point for folks to be aware of, in Part 70 for ISAs there is an ISA summary that is submitted on the docket and reviewed by Staff.

In this case the SSA summary itself was not a docketed report but is something that will remain as a licensee-controlled or Applicant-controlled document. Next slide, please.

As mentioned before, there are great similarities to the approaches in Part 70, Subpart H, and NUREG-1520. And so in that regard, the Staff made use of 1520 which also refers to NUREG-1513 in terms of providing guidance for the conduct of ISAs.

Such things are to evaluate the criteria

for the types of techniques used in the hazard analysis to ensure that those are reasonable and appropriate, also looking at the safety criteria, looking at the application of the outcomes of the analyses for the accidents in applying safety-related controls and management measures or programmatic, administrative controls.

With there being some alternatives, I think a good example again with the terminology as well as the approach, another example would be the safety criteria, which SHINE mentioned before.

In NUREG-1537 it recognizes that for ISAs, those criteria are typically in 7061 as the limiting high-consequence events to be highly unlikely and what the criteria in terms of radiological chemical dose for what constitutes a high-consequence event or an intermediate-consequence being limited such that it's unlikely, and what those consequence criteria are.

You'll see some of what SHINE has used. There are some differences there, for example, the radiological criteria is actually lower than what's in 7061 for what 7061 would look like as an intermediate consequence.

And so looking at the alternatives using what's in Part 70 and 1520 is a starting point and

evaluating the acceptability of the alternatives to demonstrate adequate safety.

Again, in consideration of this being a Part 50 application instead of a Part 70, though there are similarities to Part 70 facilities that have been noted previously.

And some of the licensing approach, for example in Part 70 there aren't tech specs whereas we have tech specs for this application.

And the SSA and its implementation are a key element of the SHINE safety program and in looking at the SSA and doing the SSA review, a major objective is ensuring that the SHINE safety program is adequate.

In doing so, the Staff did a broad review considering the method for the SSA and the safety program.

And then in terms of its implementation, we do a narrow or vertical-slice reviews selecting certain accident types to understand and evaluate the implementation of the method, ensuring it's done consistently with the definition of the method that SHINE has provided.

And that ensures the program will be adequate.

And part of that, also, another key

element is having these reliability management measures in the programs, the programmatic administrative controls, which are the programs that establish an maintain those management measures.

If we can go to the next slide.

Just a little further elaboration on that, in terms of looking at the method, the Staff looked to ensure that the method identified and evaluated facility hazards and identifying credible accident sequences, including providing definitions of what credible is as well as the definitions for the other likelihood categories they've evaluated.

And again, assessing the radiological and chemical consequences and likelihoods for, first, as an uncontrolled or unmitigated accident and then identifying and applying the safety-related controls to either prevent or mitigate the accidents to meet their safety criteria.

And then ensuring that reliability management measures were identified that ensure the safety-related controls will be available and reliable to the extent evaluated in the safety analysis.

And then having the programs for establishing and maintaining those measures. And looking at how they defined those, the definitions,

the programs, the techniques, are they appropriate, do they make sense for the way they've been used?

And that all figures into the Staff's evaluation.

And then, of course, looking at the implementation of that method to ensure that also is reasonable and it will provide the confidence that the method is being used appropriately and ensuring the facility's operation will be conducted in a way that ensures health and safety.

Again, just stressing that the SSA, the method and the implementation are an important element of the safety program.

And for that to be effective, the SSA needs to reflect the as-built, as-operated facility and demonstrate that it ensures the health and safety of the public and personnel.

And with programs that SHINE has, some of which are captured in the tech specs in the administrative program section of the tech specs such as Tech specification 5.5.1. for the nuclear safety program in maintaining the accident analysis, which includes identifying appropriate safety controls and programmatic administrative controls.

That ensures the safety program would be

effective and adequate to demonstrate and ensure the facility operations maintain public health and safety. If there are no questions, this was my last slide and I can turn it back to Elijah.

MR. DICKSON: This is Elijah Dickson.

Before jumping into the DBA analyses, I'd like to highlight some important facility design and operational features that we considered in characterizing radiological risks and ultimately the impact to public health and safety.

The target solution itself is a low-enriched uranium in the form of uranium sulfate, which is held in the target solution vessel, or TSD. During operations, the target solution is close to ambient pressure and temperature.

In the system, the primary system boundary acts as the primary fission product boundary and this is defined as the TSV, the TSV dump tank, the target solution off-gas system, or TOGS, the associated components such as piping and valves, all of which are seismically qualified.

Within the irradiation unit, the target solution is irradiated in a subcritical assembly by neutrons produced by fission neutron source.

After irradiation the target solution is

then processed in the radioisotope production facility for extraction to purify molybdenum-99 and other medical isotopes.

Radioactive waste is then processed and/or converted into solid waste for shipment to an offsite disposal facility. They utilize a typical design philosophy with defense in-depth and multiple barriers, redundancy and diversity with their systems.

Each of the accelerators are independent from each other so there's not knock-on effects between the accelerator.

All design basis accidents which are tripped by the target solution vessel reactivity protection system, or TRPS, results in immediate safe shutdown condition of the target solution within the TSV dump tank, which is favorable geometry.

The lightwater pool has sufficient capacity to passively handle decay heat following the trip and the nominal source term is quite small when you compare this to other Part 50 facilities.

Their materials at-risk source term or their safety basis source term is conservative given aggressive modeling assumptions. With that, I'd like to go onto the next slide.

MEMBER MARCH-LEUBA: Just a moment, this

is Jose. When you stick to the reactivity protection system, we are I wouldn't say concerned.

We are curious about how we're going to calibrate the sensors that are used to trip the TRPS system, especially because there is no plan to have a calorimetric calibration of the power sensors. Do you have an idea of how do we know we're tripping the correct power?

We are still reviewing the technical specifications and they have three LSSSs there, one that protects against high power, one that protects against a power average, and then for startup, there's another LSSS that makes sure that when you're doing your 1 over M type filling, you don't fill too quickly.

Again, we're still reviewing those tech specs but for the most part, we've reviewed a tremendous amount of design calculations and they have most certainly exercised MCMP to basically do a proof of principle that their system works.

So, we reviewed MCMP calculations, we looked at their geometries, the physics --

MEMBER MARCH-LEUBA: My question is more mundane, about actual operating experience. When you place an interim detector outside the vessel, wherever

the interim detector is, you end up having milliampere signal coming out of it.

And you have to be able to convert those milliamperes into a power level. So, that's with calibration of the detectors and the way we do it typically in power-plants is we have a calorimetric operation.

You have the floor that goes into the core and the T, and that gets you power. We don't have that here, so maybe the question will be in the closed session for SHINE.

MR. DICKSON: They've done a lot of calculations that show that it works. I think we're going to also have to lean heavily on startup physics testing as well to ensure the calculations they do do actually predict what is actually happening.

And that's another part of the conversation that we'll have later on in this presentation. I'll see that Joe Staudenmeier as well has his hand up, who did the Chapter 4 analysis of transience.

Joe, do you have something to say?

MR. STAUDENMEIER: Yes, we've had discussions with SHINE about calibration, I don't want to say how they are doing but I'm not sure if it could

be said in the open session.

But they have ways that they're looking at calibrating the detectors themselves and then the power level in the irradiation unit. But I think it maybe should hold off to the closed session and SHINE could address it or we could address it based on the discussions we've had with them.

MEMBER MARCH-LEUBA: The Staff has followed up on these. Go ahead.

MS. RADEL: This is Tracy with SHINE. As part of the Chapter 7 review, we did provide a detailed response to an RAI related to how we will calibrate the flex detectors.

Due to the design of the system, the calometric method is not easy and straightforward and we've had significant uncertainty on it.

And given that we had a liquid system, we're using a method that's been used by liquid systems in the past, which is looking at the particular fission product isotopes in the solution.

So, running the system, getting the profile of what was run power-wise, and then measuring I believe it's four or five different fission product isotopes and using that to calibrate the advanced detectors.

MEMBER MARCH-LEUBA: I would be glad to wait until Chapter 7 or the closed session but since you were volunteering, you are going to mention the isotopes online as you start that?

Or is this a batch production after you run it for X number of days?

MS. RADEL: It would be a batch, it would be post-irradiation as it's been through the super cell.

MEMBER MARCH-LEUBA: That has the potential of being very accurate, except, again, we go back to the first cycle and we have to be very conservative that we can be off by 20 percent with our estimate until we do the calibration.

Thank you, I'll wait until Chapter 7, that sounds like a good approach.

MEMBER REMPE: Is this response to an RAI something recent? Could you have follow-up with our DFO, Chris, and make sure that he makes us aware of it. Because perhaps we already are but I'm curious about your response.

CHAIR BALLINGER: That SHINE response was just submitted back in April so it might be good to follow up.

MEMBER REMPE: Thank you.

MEMBER MARCH-LEUBA: This is Mike Balazik. Dr. Rempe, I'll send that over to Chris after the meeting.

MEMBER REMPE: Thank you.

MR. DICKSON: If there are no other questions we can move onto Slide 12.

The design criteria and the radiological acceptance criteria, as I've mentioned before, SHINE has one particular design criterion, Design Criterion 6, that is specific to Chapter 13 analyses for the control room.

It's your typical control room habitability criteria where a control room is provided from which actions can be taken to operate the irradiation unit safely under normal conditions and perform the required operator actions under postulated accidents.

This is similar in effect to General Design Criteria 19 for power reactors.

Now, for the acceptance criteria or dose acceptance criteria, there are two, the first one being the criterion, which typically referred to members of the public.

It's where radiological consequences to an individual located at the unrestricted area following

the onset of a postulated accidental release of licensed material would not exceed 1 rem total effective dose equivalent, or TEDE, for the duration of the accident.

Then the second is the acceptance criteria for the control room operator where radiological consequences to the worker do not exceed 5 rem TEDE during an accident. I know there's a couple questions on this.

MEMBER MARCH-LEUBA: I have a new one, I'll give you fair warning. On the SCR, the SCR goes into some detail on all the criteria that could have been chosen. Could you give us a summary of why this one was the proper one?

MR. DICKSON: Currently, in the rule, in the CFR, there's no accident dose criteria for NPUF-type facilities and so in writing the SCR, we provide a bit of background as to what has been selected in the past.

Typically, that has been some variation of the Part 20 actual public dose criteria. Around the time that SHINE had come in for their operating license, the NRC had published for rulemaking the NPUF rulemaking, which selected a regulatory dose criteria of 1 rem TEDE and SHINE came to us and utilized this

as the value for their accident dose criteria.

And it's effectively based off the EPA PAG manual for protective action guidelines and guidances for radiological instances.

MEMBER MARCH-LEUBA: Thank you, now we have it on the record. When you read the CR, it looks a little confusing that there are so many criteria that one can choose from but there is a method to the madness.

MR. DICKSON: Over time, over the last 30-some-odd years that we've been licensing these things, in some place you'll see 100 millirem, in some places you'll see 500 millirem, and then somewhere in between.

So, we tried to flesh that out in the SCR. If we need to do a little work on further explaining it, we can.

MEMBER MARCH-LEUBA: I'm happy with your explanation, thank you.

MEMBER KIRCHNER: If I remember, historically, a part of this 1 rem ties to the protected action guidelines, the idea that these NPUF facilities would not, in effect, exceed that 1 rem for actuation of full-blown emergency planning requirements.

MR. DICKSON: That's right. We use the 1 rem for other regulatory actions as well as, for instance, for decommission facilities the 1 rem is used. It's something that we've used before.

Onto slide 13, please? This slide was developed in response to the one question that was received from the ACRS last week.

The methodology in assessing design basis accident radiological consequence analyses, they're generally divided into six parts, where you select bounding design basis accidents.

These design basis accidents really are categories, you'll do a number of different analyses within each of these categories. You'll then derive applicable accident source terms for each of the DBAs.

You identify the major safety-related structures, systems, and components, or SSCs, intended to mitigate the radiological consequences, you estimate fission product release characteristics to the environment, you review the meteorological characteristics of the site location.

That's a very important step in the process. And then you finally compute radiological consequences for each of the bounding DBAs within those categories.

Generally, the Staff does not accept DBA analyses that credit facility features that are not safety-related, are not covered by the technical specifications, do not meet single-failure criteria reliability offsite power.

Now, I believe Dr. Rempe had asked a little bit about the purposes of all of these different nuances, and the purpose is to ensure the reliability of the system is performing its safety function to protect the public health and safety for special events.

And so we can hang our hat on the final dose results when we consider all of these different aspects in the dose analyses if they need them. They are in fact meeting their intended safety purpose function.

Onto Slide 14, please. Here is a list of the SHINE design basis accidents, which are consistent with the interim Staff guidance. The SHINE facility is unique but NUREG-1537 and the ISG were helpful in developing and reviewing the DBAs applicable to SHINE.

It assisted both them and us and looking at as many reasonable possible failures and combinations, failures of combinations of SSCs, to understand as many radiological health consequence

results as possible.

They did analyze classic accidents, such as accessory activity, reduction of cooling, power oscillations. What they did, SHINE and Tracy had mentioned that they did identify facility-specific events, which are typically tritium events.

For the most part, the failure reactive systems do not lead to an uncontrolled upset of the target solution, which would then cause a release to the environment.

Accidents such as reactivity insertion and power oscillations for their particular design intend to be self-correcting. And more importantly, the LSSSs and the tech specs have been set to protect the primary system boundary.

We will talk briefly about their maximum hypothetical accident in a few short slides. If we go onto the next slide it just has --

DR. BLEY: Elijah, before you do that, this is Dennis.

We're returning in this case and probably in some others to the maximum hypothetical accident or the maximum credible accident, depending on which approach people want to take.

If one has a convincing case that their

MHA is truly the maximum and even so, it's acceptable, why do they need to go beyond that?

I think one reason, I'm going to maybe help you out or maybe give you something to correct, is having some confidence in the completeness of the remainder of these 2 through 12 on this list gives confidence that the MHA is in fact what it claims to be.

You talk about that a bit and I'm thinking not only of this application but of how the idea might be applied in future applications.

MR. DICKSON: I think assessing the facility from multiple different aspects is important. It's hard to say that you can use one accident category and then a number of bounding assumptions to then say that you are covered, completely covered.

And so I do think, and I agree with you, that looking at the other accidents helps identify things that you may not have realized before when you do your initial design of the facility.

We see for advanced reactors where they utilize their PRAs to then make design modifications as they're going through the design process. It's I think similar in this case.

You may have a blind spot when you're

going into this and if you look at it from other aspects, you might identify that you may need an extra safety-related system to handle something you weren't aware of.

DR. BLEY: Thanks, I just wanted to hear the Staff's take on that idea.

MEMBER MARCH-LEUBA: This is Jose. Is this question or topic related to what we're talking about with the lack of or incompleteness of all the regulations, they're not specific or exactly what you're supposed to do?

And it's something maybe you can talk to your friend on this in the suggestion box for future applications. It will be nice if I'm designing an NPUF next year, I get money from Wall Street to do it, that I know what the Staff expects from me.

MR. DICKSON: Yes, SHINE looked at the interim Staff guidance and this list of accidents is effectively there.

They went and they reviewed them and if we saw an application which is one accident and then looked at our own guidance and said where is the rest of them, it's a high bar to say --

MEMBER MARCH-LEUBA: Your recommendation, as one member of the Staff, is to do an MHA to cover

your bases, but then again, run through a spectrum of accidents to make sure you recover?

MR. DICKSON: Yes, I'd definitely recommend that. I think in the case of SHINE, they did all of these analyses and then they identified what their MHA was, it was kind of the other way around.

MEMBER MARCH-LEUBA: If you think about it, the MHA in SHINE does not cover tritium releases, for example.

MR. DICKSON: No, it doesn't.

MEMBER MARCH-LEUBA: It's different. You can even think it's a different facility, one is a TSV, one is tritium purification.

MR. DICKSON: Right, that is the nuance, and I'm glad you brought that up, about the SHINE facility. They have the fission-product-based accidents and then they also have a lot of tritium on site too.

They went and performed the analyses to understand the consequences of tritium accidents. Go ahead, I'm sorry.

DR. BLEY: Go ahead, you hadn't finished yet.

MR. DICKSON: The guidance itself tells us

that the MHA is a fission-product-based-type accident and that makes sense because the radionuclides that you're most concerned of are those shortlived, highly radioactive nuclides, specifically radioiodine, its affinity for the thyroid can cause high doses.

SHINE also has these handful of tritium events and their tritium events do result in high doses too but tritium has a 12.3-year half-life and the actions that you need to take to respond to a tritium accident are much different than the actions that you would need to take to respond to an upset of the core.

In the sense of protecting public health and safety, do you need to evacuate, not evacuate? Things of that nature.

And so I might be going a little bit off track here, they were able to utilize I think it was the very, very last item here, the facility-specific events who identified those tritium accidents.

And I think that's a good thing.

DR. BLEY: Elijah, that was following up a little bit on Jose here. You mentioned that they use the set from NUREG-1537 but when SHINE talked, and I liked what they had to say, they used the HAZOPS and the failure mode effects analysis ahead of that to

look for things that might not be in that list.

I think that's pretty important, you didn't talk about that, you held onto that list. Can you talk about that idea a little bit?

The thing you're sensing is some of us are very concerned about how to be as complete as possible when looking at new facilities.

MR. DICKSON: When SHINE initially came in, they had their SHINE safety analysis and performed that HAZOP. And that was primarily the NMSS side of the house that performed that review.

That information then did feed into the NUREG-1537 and ISG analyses, it did feed into it.

It made a pretty complete story I think at the end of the day, since for the DBAs, we're effectively assessing structures, systems, and components and we're not necessarily assessing the actual processes that humans are interacting with the systems themselves.

Two important aspects looked at two different perspectives, and it's all combined in Chapter 13.

DR. BLEY: That covers what I wanted to hear.

MEMBER HALNON: This is Greg, back to the

process question about going beyond other things besides the MHA, isn't that required or necessary to determine the scope of safety-related equipment and what's covered by tech specs and that sort of stuff?

MR. DICKSON: Yes.

MEMBER HALNON: So, we just can't stop at one accident, you need to do the full scope so you know what other instruments and what other safety-related equipment needs to be there.

MR. DICKSON: Yes.

MEMBER HALNON: I just wanted to make sure I was thinking right.

MR. DICKSON: Yes. If there's no other questions, we can move on. This is just the rest of the identified applicable DBAs and then Slide 16, we can now talk about the material risks and accident source terms.

CHAIR BALLINGER: This is Ron Ballinger.

I've been searching for a time where we could take a break, a convenient time, and this looks like about as good as any. So, what I would like to propose is that we take a break until 10:20 a.m., which would be 15 minutes from now.

Let's take a break and come back at 10:20 a.m. by the clock on that computer. Thank you.

(Whereupon, the above-entitled matter went off the record at 10:04 a.m. and resumed at 10:20 a.m.)

It is 10:20 a.m., time to reconvene. So, let's pick it up where we left off.

MR. DICKSON: This is Elijah Dickson again. We are on Slide 16, design basis accident analyses, material risk accident source terms.

SHINE presented to us two types of materials at risk, the first being the fission-product-based source term, which is their safety basis source term derived for the target solution vessel. They also produced another fission-product-based source term for the primary closed cooling system as well I believe.

The other primary source term is the tritium source term which is based off of maximum quantities for the facility were used by an individual irradiator assembly.

The NRC Staff reviewed the applicability of the safety basis SHINE calculation documents which were used to derive their material at risk source term. We looked at validation calculations for their reactivity solution system, the ways the estimated neutron fluids target solution burnup over the length

of the target solution recovery.

They utilized two primary codes or classic codes to perform these calculations, the first being the Los Alamos National Lab, Monte Carlo N-Particle 5 code, MCMP5 Version 1.6. MCMP5 was used to compute neutron flux spectrums and cross-sections with the target solution as well as the PCOS.

SCALE was then used with a code developed by Oak Ridge National Lab, it's the standardized computer analysis for licensing evaluation code. Specifically, the ORIGEN-S was used to perform the depletion calculations.

MEMBER MARCH-LEUBA: Is there a reason you're using MCMP5 instead of 6, and is 5 still supporting and getting updates?

MR. DICKSON: I'm not sure, I use MCMP5 myself. I do believe they're onto just MCMP now, they don't have a number after it.

MEMBER MARCH-LEUBA: I think it's MCMPX?

MR. DICKSON: MCMPX was discontinued I believe a number of years ago. They wanted to the MCMP6 but then I think the latest version is just MCMP at this point.

MEMBER MARCH-LEUBA: The question is if they're planning to use a frozen version but if Los

Alamos finds a bug in the implementation, then that does not get translated into the flux version. Then, on the downside, if you use a version that changes daily with updates, then you never verified and validated it.

So, maybe this is a good compromise.

MR. DICKSON: MCMP5 has been around for a long time, it's certainly a tried and true version of the code.

MR. MUNSON: This is Jeremy Munson, I could make a quick comment on that.

MEMBER MARCH-LEUBA: Go ahead.

MR. MUNSON: I just wanted to say what really matters is not just the code version but the cross-section data that the code is validated with or when it's validated, its area of applicability.

And known issues within the area of applicability within the codes should come out in the wash whenever you do the bias determination calculation. That's part of the reason why we do the validation.

So, regardless of which version of the code they're using or which cross-section library they're using, as long as they're operating or doing their calculations within their area of applicability

as established by the validation report, they should be okay.

MEMBER MARCH-LEUBA: I don't have any problem with MCMP, either version, it's one of the gold-standard goals on this. Go ahead.

MR. DICKSON: The last bullet is they utilized NUREG CR-6410 to compute accident-specific source terms utilizing the so-called five-factor formula. Now, onto Slide 17.

Verified operational assumptions in deriving their material at risk, for each of these they did include margin.

The corresponding fission product power, their license fission product power with additional margin, irradiation times per cycle, total time lengths between irradiations, extractions between irradiations, and then the length of target solution recovery were the primary parameters in which they derived their material at risk.

We find that they used the most aggressive usage of their target solution that would effectively fit within their licensing basis.

They maximize cycle lengths, minimize downtime lengths in their calculations and neglected the evolution of iodine from their material at risk,

which is something that Mike Salay will be talking about in a few slides.

The calculations do include the effects of fission transmutation, activation, and decay. They assessed burnup assesses different radionuclides' peak at different burnups.

We asked some questions in regard to making sure they're capturing peak radionuclides of interest such as Item 131.

We find there's very large margin between their material at risk and normal operations and the Staff finds the conservative assumptions and treatment of uncertainty to justify the material at risk to be acceptable.

On Slide 18, the material at risk transport and mitigation, this is Slide 1 of 3, we reviewed NUREG CR-6410 which has a process in which you developed a so-called leak path factors.

Leak path factors were developed for each scenario where you identify major safety-related SSCs intended mitigate radiological consequences and estimate fission product release characteristics to the environment using those leak path factors.

These factors include physical processes such as control volumes, volumetric flow rates,

leakages through gaskets, differences in pressure, barometric breeding, and other removal processes.

We find they did conservatively assume with the five-factor formula damage ratios to be one, that means that any piping or tank or piece of equipment that breaks fully breaks and releases that material.

Nothing is upheld within it and that airborne release fractions, or ARFs, vary by accident but typically, they assume an airborne release fraction of one for most scenarios.

The leak path factors are generally organized into four leak path combinations for the entire facility.

You have the release location, the initial confinement, leakage into the surrounding building and then subsequently to the environment.

The Staff finds their leak path factors are generally consistent with the methodology of NUREG CR-6410. With that, if there's no questions --

MEMBER KIRCHNER: This is Walt Kirchner, I have a question. When you get to the actual finish of construction and you begin pre-op testing and everything, do you go back and look at things? I'll pick on one.

The removable shield blocks, I imagine these also have gaskets that help form the primary confinements. Do you do leakage testing of those and then check back to your calculations to see that the appropriate leak path factors has been used, it's been conservative?

MR. DICKSON: Yes, so that was something the Staff did focus on, where these gaskets were used to effectively establish confinement. We did a review of their design calcs and they use first principles in computing analytical leak values.

Once they do startup and testing, they'll be able to actually then measure leak rates and then those would be utilized in their design calcs.

MEMBER MARCH-LEUBA: Are you done, Walt?

MEMBER KIRCHNER: Yes.

MEMBER MARCH-LEUBA: I mentioned this in an earlier meeting, that they calculated those rates to the public are very close to limits, they're within 20, 30 percent of the limit because we used a very conservative calculation method and assumptions.

Now, the danger here is that when we actually test the as-built facility, we find the leak factor through one of those gaskets is 25 percent larger than we assume, and then we're over the limit.

So, I'm just warning that if feels that using an extremely conservative calculation method with extremely assumptions is a good thing to do until you get bitten by it.

So, let's just make sure that when the facilities are built, the gaskets are placed properly and they test within the numbers.

It's not a question, just a comment.

MR. DICKSON: I understand, thank you. Are there any other questions?

With that, we'll move on to Mike Salay's presentation, the iodine evolution calculations that he had done in his assessment for material at risk transport mitigation.

MR. SALAY: Can you hear me?

CHAIR BALLINGER: Yes.

MR. SALAY: Hi, I'm Mike Salay from Office of Research, I reviewed iodine release and transport and a few other things related to non-iodine release and transport and this slide highlights some of the relevant effects of iodine evolution.

These aren't specific to SHINE but rather, generic to aqueous fission systems and even other fluid systems that postulate an accident which evolved iodine can leak into the environment.

It also applies to other walled housing, it can leak and perhaps in other systems. So, in general scenarios you have normal operation with fission and an accident by evolution without fission.

And fluid systems behave different than typical solid LWR fuel since some radionuclides are mobile, readily mobile. And these effects are often not considered in dose analyses.

So, iodine that evolves from solution to gas space, it can leak to the environment and specifically it can be a significant contributor to flow and the release rate generally depends on geometry, temperature, and flow.

But there are many other factors and internal flow within the fluid, but one thing that needs to be realized is this process is always occurring. Iodine evolves from solution during normal operation and if you account for these effects, it has implications for how much can be released during the accident.

If evolution is fast relative to your decay constant, it can deplete the inventory available for release to the environment, so this results in some effective reduction in your MAR by evolution.

And these effects aren't accounted for

inventory codes such as SCALE or MCMP. And on the other hand, if your evolution is slow relative to decay, your evolving radionuclide will decay before evolving.

I'm focusing on Iodine 131 since it's typically the most dose-significant isotope. And so this effectively limits on your airborne release fraction and the combination of the two.

And these effects are shown in the figures on the right and these effects are basically common to all fluid systems, not just aqueous systems.

If your volatile radionuclides evolve and are sequestered during normal operation, they're no longer available for release from the main irradiation facility or in this case, reactor to reactor, during an accident.

Although, of course, wherever the location where the radionuclides are sequestered can be another radiation source.

Again, the figures on the right illustrate some of these effects on Iodine 131 behavior and the competition between decay and evolution during operation in a postulated accident scenario using some simplifying assumptions.

The key assumption here, one of the main

assumptions is that evolution removal constant from solution during accident scenario is the same as that during operation. And this may not necessarily reflect actual scenarios.

The top figure shows the buildup of Iodine 131 inventory during normal irradiation and the release history by evolution during the postulated accident scenarios. The solid curve shows the build up for stationary iodine in which there's no evolution, the red curve.

For iodine that is evolving with the removal constant, that's 10 times greater or one-tenth that of the radioactive decay constant.

The longer-dash curve shows evolution of airborne release fractions for these two evolution cases, and the short-dashed curve shows the combined effects of the reduction and inventory and release fraction. And this timescale is about 70 days.

Xs simply represents the fractional inventory relative to the steady state inventory, the equilibrium inventory with no evolution. Xe represents the fractional equilibrium inventory due to loss by evolution.

And this is essentially the equilibrium Xs or Exs at the end of time. The bottom figure shows

the same effects but for equilibrium conditions way beyond 70 days and as a function of the evolution removal coefficient.

The solid red curve shows the effect on equilibrium inventory, the green longer-dashed curve shows the airborne equilibrium release fraction, and the blue shorter-dashed curve shows the combined effect of the two.

These figures show the combined effect of inventory reduction airborne release fraction for evolution limit overall releases during an accident scenario starting from initial equilibrium inventory, and the reduction is greatest when the evolution rate constant is much larger or smaller than that of the radioactive decay constant.

And I'll point out these curves are just focused on evolution rates. They neglect the iodine partitioning so these effective ARF airborne release fraction values kind of provide an upper limit on releases based on transfer condition alone for the simplified assumption.

So, one can generally say it's conservative to neglect these evolution rate effects. However, it can be difficult because of the complex and related behavior between aqueous speciation,

liquid gas partitioning evolution rate to demonstrate that a given evolution rate provides a bounding release.

Some NUREGs related to ion behavior often focus on equilibrium iodine partitioning rather than the evolution rate.

The effective evolution removal coefficient, lambda E, in addition to depending on typical mass transfer behavior including diffusion through water and gas and recirculation of fluid, and the surface to volume ratio, it also depends on the iodine speciation in solution.

Depending on pH and the concentration of all iodine isotopes in solution, some fraction of the iodine will be in volatile I2 form, molecular iodine form, whereas some other fraction will not be volatile.

It is only the volatile I2, the volatile iodine, that is subject to evolution of the gas phase partitioning, and so the effective partitioning and evolution rate depends on volatile iodine fraction.

In other words, since only the volatile iodine evolves, the aqueous chemistry model affects the effective iodine evolution rate on that second curve, where you are on the X axis on that bottom

figure.

So, the chemistry model predicts that all of the iodine solution is in volatile form, the effective equilibrium partitioning and evolution rate to match is the nominal partition coefficient and nominal evolution removal coefficient.

Conversely, if the chemistry model predicts a low fraction of iodine in solution, the effect of equilibrium gas to liquid ratio and evolution rates are correspondingly reduced.

So, uncertainties in the chemistry model and the volatile iodine fractions lead to uncertainties in the effective evolution removal coefficient.

DR. CZERWINSKI: I've got a question, this is Ken Czerwinski. Maybe you'll do this a little bit later but can you give some information on this speciation model that you're using?

MR. SALAY: This is the B model, it's NUREG 5950 and this completely neglects that. I'm just saying these are effects that --

DR. CZERWINSKI: I understand what you're saying, where the speciation is going to drive the formation of the volatile iodine species and that would be the species of concern for that isotope.

MR. SALAY: Yes, so it's whether you have I2 or I-. I haven't put anything together for that, it's more detailed, but of course it could be provided in the future, unless SHINE provided some info.

I'll go on and I can provide more information in the future about the different models as needed.

CHAIR BALLINGER: Can that information be provided in the closed session?

MR. SALAY: I haven't prepared anything for that but I can.

CHAIR BALLINGER: I'm just trying to capture what we should do with that.

MEMBER PETTI: It's also fair to say SHINE didn't go into this level of detail. They made much more conservative --

MR. SALAY: They did not consider the transport effects but the speciation in partitioning, they did include this based on the NUREG-5950 iodine evolution pH control model.

MEMBER REMPE: And that was in the FAI report? They actually, I thought, did a pretty good job.

MEMBER PETTI: As I recall, they did.

MR. SALAY: Another effect is the higher

surface to volume ratio configuration such as bubbles and froth. You can also enhance the effective iodine evolution rate.

Although, for this illustrative example, it considered the same evolution rate coefficient for accident conditions as for normal operating conditions. The evolution rate for an actual analysis should consider the actual geometry.

And generally, one would expect that the evolution rate during an accident, postulated accident, would be lower than that during operation, namely because your power generation can result in more bubbles, it can result in more natural circulation, all of which enhances the evolution rate.

And I don't know if I mentioned it, but uncertainties in your volatile iodine fraction lead to uncertainties in your effective coefficient. Anyways, next slide, please.

MEMBER KIRCHNER: Before you go on, it seems to me for the duty cycle, the operational cycle that SHINE is using one could construct a composite curve that includes some conservative assumptions and then come up with a curve available for release as a function of that multi-day duty cycle that they're operating on.

MR. SALAY: Yes, it seems like a comment but, yes, I think you can account for the reduction due to --

MEMBER KIRCHNER: So, this would take into account the phenomena you identify on the previous-view graph but you would have an effective total I2 inventory available for release that then could be used in a bounding calculation?

MR. SALAY: Yes, I think I agree with what you're saying and this simplified illustration just considered a constant irradiation, it didn't consider any cycling.

And one thing that I forgot to say in the previous slide was because of these interrelated effects, it can be difficult to justify that you have a bounding value for these rate effects.

So, this slide lists some of the evolution transfer analyses. As mentioned before, iodine evolution analyses can provide an estimate of reduction in dose-significant inventory during operation and the airborne release fractions.

For situations where any potential reduction in inventory is not credited and all iodine is assumed to be released to the gas, there's really no need for an iodine evolution calculation because

you're just assuming everything is released and nothing is reduced during operation.

So, it of course did not conduct or analyze any of those situations. One could to estimate conservatism, instead it just went through the simple generic analysis that I showed on the previous page.

Other than that, I did look at some specific scenarios that focused on higher-consequence scenarios that involved iodine evolution.

The specific one they looked at was the scenario in which the iodine was released to a pool and started the calculation, went through just walking through the process of release.

It was getting to the point where I was getting a lower considering rate effects, I was getting a lower airborne release fraction, substantially so, than SHINE.

And so even though like I said, the analysis didn't actually continue, given that it was clear that our analysis would have lower release fraction than SHINE and airborne release fractions that are released to the environment in a lower dose, it wasn't clear whether it would be useful to continue to perform the analyses, especially considering this

was feeding into the NRC confirmatory calculation track thread.

And I also provided feedback on the technical basis used to derive the pressure during flow rates, including non-iodine airborne release fractions and phenomena used to calculate the leak path factors that were used in the confirmatory calculations.

And this is more of a high level review that references the equation but I don't think I calculated anything except for perhaps the flows rates.

And if there are no other questions, I'll turn it back over to Elijah.

MR. DICKSON: This is Elijah Dickson, we'll move to Slide 21. I'll briefly talk about atmospheric dispersion or meteorology. SHINE developed short-term atmospheric dispersion factors, or chi over Qs, using traditional calcium plume diffusion methodologies.

The chi over Qs were developed for both the offsite public location and the control room receptor. The chi over Q values were computed for specific time periods following the event from 0 to 2 hours, 0 to 8 hours, 8 hours to 24 hours, 1 to 4 days,

and 4 to 30 days.

They conservatively assume a ground-level release, the chi over Q values are 95th percentile chi over Q values and they utilize the NRC computer code PAVAN, which implements regulatory guidance from 1.145.

And there should be another bullet here in regard to ARCON2 I think for computing chi over Qs for the control room. Onto Slide 22 if there's not questions, radiological consequences.

We find that they computed radiological consequences to be consistent with the regulations so in terms of total effective dose equivalent, which is defined in 10 CFR 50.2 then of course in Part 20, 20.1002.

They utilize the appropriate dose conversion factors to compute committed effective dose equivalent, which is the internal doses, that's utilizing DCS for Federal Guidance Report 11.

And then for external exposures, they utilized dose recursion factors from Federal Guidance Report 12. As Tracy had mentioned earlier this morning, for the fission-product-based accidents, they utilized a time duration of 30 days.

For the tritium accidents they utilized a

time of 10 days. Receptor locations assume no personal protective equipment or protected actions. The individual effectively just stays put for that period of time.

Any questions here? Onto Slide 23, we'll just very briefly go over SHINE's maximum hypothetical accidents since the intra-Staff guidance tells the Staff to identify and focus on it.

Their MHA can be found in Subsection 13A2.2.7, it's under the design basis accident category of mishandling or malfunctioning of equipment.

Their most limiting scenario is the failure of the target solution vessel off-gas system, or TOGS, pressure boundary, resulting in the release of off-gases into the TOG cell.

So, effectively, what they assume is a break in the TOGS line in the upward section of the TOGS lower in conjunction with the complete blockage of piping in the process vessel ventilation system, or heat PVVS.

This is effective multiple failures. The blockage in the PVVS system creates the back-pressure when the nitrogen purge system clicks on, pressurizing the TOGS cell, and then the source term or material at

risk then leaks out of the TOGS cell.

No credit is taken for deposition of radioiodine in this analysis. Onto Slide 24. As Mike Salay had mentioned in the two or three previous slides, they utilize a material at risk of 100 percent of the halogens and noble gases.

Interim Staff guidance asks us to identify the safety controls for the accidents and those would be the primary confinement boundary, ventilation radiation monitors and nitrogen purge system, ventilation isolation mechanisms, and then for a brief time, a hold-up in the radiological event Zone 1E exhaust section.

For the calculated doses for the MHA scenario, they computed a control room operator dose of 1.94 and then for their MHA to a member of the public is 0.727 rem.

The Staff finds these results are acceptable since they're within their design accident dose criteria for the control room as well as the 1 rem TEDE out at the site boundary.

MEMBER PETTI: Elijah, I'm just a little confused. There is another scenario in the SHINE accidents that products slightly larger public dose, slightly lower worker dose. Why isn't that the MHA?

MR. DICKSON: I'd have to look at my notes. Are you referring to one of the tritium accidents?

MEMBER PETTI: Yes.

MR. DICKSON: I had mentioned before in our discussions that the MHA is a fission-product-based source term and the guidance asks us to assess the MHA as a fission-product-based source term. Now, SHINE, because they have a lot of tritium on site, they did perform other accident analyses with tritium.

And I think the important distinction between the two is that tritium does have quite a bit longer half-life than Iodine 131 does, and the immediate radiological threat to the public post-accidents is truly Iodine 131.

Because that's how you start to set up -- these calculations set up the reasons for protected actions, right?

So, if you have this quick, fast-acting accident with Iodine 131, you'd be taking protected actions that are much different than those with the release of tritium with the 12-year half-life.

These doses, I'd like to mention, are 50-year committed doses so the intake is integrated over

a 50-year period upon intake. It's not just an immediate 2 rem per se but it is dose that would be integrated over 50 years.

So, the tritium dose takes a lot longer to effectively deposit this material into the human body. Again, that all goes into how you consider protected actions.

DR. BLEY: I want to ask you, and again, this is probably not fair to ask you, maybe somebody else wants to comment on it, you confirmed the calculations for control room operators and I'm a little curious.

The NRC is charged with protecting public health and safety, I guess that's why we look at the control room operator, because the control room operator would be important to protecting public health and safety.

We don't seem to require looking at worker risk. Now, SHINE did, they told us they looked at people outside the control room.

I'm thinking back to the 1960s and the NRC didn't look at environmental effects until there was a court case that said you've got to do that, you're putting reactors out there and they may have environmental effects.

We're putting reactors out there that could have worker effects and they've been pretty benign so far, but we're also looking at new systems with very interesting chemistry and chemical risk for workers.

Why doesn't the NRC concern introduce with worker risk itself rather than just the worker risk that affects the public?

MR. DICKSON: I do not have an answer for you on that.

DR. BLEY: I didn't think you would, I was hoping somebody else would jump in. I'm interested in pursuing this because we have some systems that could be introducing, this system in particular, much higher worker risk than we've seen before.

Nobody from Staff wants to help?

MR. BORROMEO: This is Josh Borromeo, Chief of the Nonpower Production Utilization Facility Licensing Branch. For this SHINE review we evaluated the items and regulations that are put forth to this facility, the same type of facility.

The question you're asking I think is a more broad question that is beyond the scope of this review but we certainly understand it and we can bring that back to the suggestion box that we were talking

about before.

DR. BLEY: I think you'll be hearing more about this on other kinds of applications but it just seems reasonable to me, and I see a parallel with the environmental effects way back 60 years ago, that we ought to be protecting the workers as well as the public.

MEMBER PETTI: My view on this, because I bumped into it in a fusion application, some of the regulators, we allow more dose to workers because we assume they're basically saving lives in the public so we allow them to have a dose limit that's greater than the general public.

What if you don't have a significant dose to the public, then what's the right limit for the worker? This is more in the reactor context. It's an interesting discussion that can come up when the relative risks are talked about and buried in the actual ghost numbers that are allowed.

MEMBER SUNSERI: This is Matt and what I hear you all talking about is there are limits for workers and there is the ALARA rule for the general design of the facility.

MEMBER PETTI: The question is, is that numerical number the right number when you've got a

facility that has, say, not significant offsite risk? What is the right number? Is ALARA applied?

MEMBER HALNON: This is Greg, I think we need to keep in mind this is a maximum hypothetical accident perspective. There's no safety-related actions that need to be done to mitigate real accidents.

And it's hard to determine worker dose when they can be evacuated and brought to low-dose areas or even the control room for that matter or sent offsite.

So, it would be very accident-specific, operator-action specific, if somebody was stuck in one position to operate one thing in a location.

So, it would be really difficult in my mind to come up with an occupational dose for a hypothetical accident that operators don't have to be there. Now the control room is different because they have to be there and that's why you calculate their dose.

CHAIR BALLINGER: I guess I don't see an issue because a combination of the worker dose limits and ALARA, don't they just basically solve the problem?

MEMBER MARCH-LEUBA: I think the

difference between the public and the employee is mostly in DOE space you are monitored or you're not monitored.

If you are not monitored, you better not give me more than 100 millirem because I'm not carrying out a dosimeter, I don't know how much you're giving me.

Whereas, if I carry a dosimeter I can get up to 5 rem and many studies suggest that 5 rem is the limit where no adverse effects are seen. So, I have high confidence that I'm not getting more than 5 rem if I'm monitored and I'm an employee.

Whereas, if I'm a member of the public, you're telling me I'm getting 100 rem but I don't know.

MEMBER HALNON: I did an analysis operating large light-water reactors and they needed to put missile shields on the containment entry at post-accident, and the dose rates were too high for them to do that. So, that was a problem.

So, there is very localized effects that could occur that would cause a problem but you have to engineer your way out of those types of things.

MEMBER KIRCHNER: Just an observation, there's a footnote in 5034 on the acceptable doses,

quote, unquote, acceptable doses, for purposes of doing your design basis accident consequence analysis.

But the footnote makes it very clear that the 25 rem and the other values that are cited are not, what shall I say, expected or acceptable. I haven't memorized the footnote.

These are limits, the presumption is that the Applicant will demonstrate in the consequence analysis that there's significant margin to that 25 rem.

Or in this case, I would hope for the 5 rem for the control room operator, if someone came up, an Applicant with an analysis that said -- I'll make up a number -- it's 4.6 rem, after this DBA I suspect the Staff would look very hard at that and say, this is not -- although it meets the, quote, unquote, 5 rem requirement, this is questionable and would probably result in further review by the Staff as to the acceptability of that DBA.

DR. BLEY: There have been bits of history floating around here, very historically, the reason worker risk was allowed to be higher was because it was assumed they were voluntarily there, where the public wasn't voluntarily nearby.

We have evolved so that the guidance looks

at the control room operator. I maintain that still because, Greg's right, they have to be there is one thing but the other thing is we need them to protect the public so it's still a public risk effect.

To Walt's last statement, the SHINE folks weren't required to look at the operators outside the control room. They did. NRC Staff is just reporting on what the guidance requires and that's the operators in the control room.

And as we move to new technologies where the chemical risk to workers might be much higher than the radiological risk, we don't have anything to fall back on but perhaps OSHA.

So, I think it's something the NRC should be thinking about.

MEMBER KIRCHNER: I agree with you, Dennis.

MEMBER REMPE: Dennis, you're not questioning heroic actions by workers, which is where that would fall, but you're also saying, hey, we need something for chemical releases to also address what is required for their heroic actions?

Because the workers are there, if they could be evacuated they would be but if they need to be there it's because they're needed for heroic

actions, right?

DR. BLEY: I won't see that written down anywhere, Joy. They often do heroic actions and if you read some of the really detailed explanations of what happened at Chernobyl, not Chernobyl, well, yes, there too but Fukushima, it's very clear that they do that.

But they're also there and, I don't know, if you have a chemical kind of problem you might get exposed before you can get out of there. So, I think it's something that's just been a gap and ought to be considered.

MEMBER KIRCHNER: Dennis, this is Walt again.

Not to belabor this, but if I remember correctly, with GDC19 for the power reactors, one also has to look at toxic chemicals, not necessarily what we are talking about here where the toxicity is a result of the production operation but from when you look at siting and offsite external hazards, typically, am I not correct, toxic exposure is part of the analysis for GDC-19.

DR. BLEY: You might well be correct, I don't remember. I'd have to look it up.

MEMBER KIRCHNER: It's like things in the

siting where they look at release of chlorine gas and then do a dispersion calculation to see what the control room and operators would be exposed to.

DR. BLEY: And the focus is on the control room operators again, Walt.

MEMBER KIRCHNER: I agree with you.

(Simultaneous Speaking.)

DR. BLEY: And all those external kind of events.

MEMBER REMPE: Could this come back to the GDCs that we haven't reviewed yet may not be adequate?

DR. BLEY: I think the Staff can go on with their presentations.

MEMBER KIRCHNER: I would observe, though, that 10 CFR 53 draft does address this matter but it doesn't distinguish control room from the other onsite worker status.

MR. DICKSON: This is Elijah Dickson. Are you ready for me to keep presenting?

CHAIR BALLINGER: Go ahead.

MR. DICKSON: I'd like to mention that part of control room habitability for power reactors do look at chemical effects, there's some guidance on that. Off the top of my head, I don't remember it but that is assessed in some fashion for operating power

reactors.

And then for events that actually do occur, Part 20 does take into effect the actual occupational dose limits and ALARA practices that would be practiced by SHINE Staff there to ensure that doses would be maintained below the actual occupational dose limits of Part 20.

Onto Slide 25, just very briefly, a quick discussion in regard to the technical specifications. The interim Staff guidance asked us to take a look at the tech specs and we are still reviewing the tech specs as a whole.

We believe we'll have a presentation for you in the future on them. The LSSSs for protecting the primary system boundary are all set to protect the primary systems boundary under a variety of conditions.

So, the first three of them themselves protect against power excursions and boiling events within the target solution vessel, others protect against hydrogen buildup with the TOGS.

The second bullet, we'd like to discuss here that we asked them to revise Tech Spec 3.4 to include an LCO for the primary confinement boundary. That way, they can ensure the primary confinement

boundary is performing its safety function.

When they do start operations, they're limiting themselves to 85 percent power in the tech specs and Tech Spec 5.8.4 states that SHINE will conduct startup testing in accordance with the startup testing program and it will submit a startup report to the NRC within six months of completion of the startup testing activities.

And so this will allow us to assess that work that we've done and the work they've done up to startup and see how well their analyses will predict actual operations.

I have nothing else on this slide and if there's no other questions we can move on to chemical safety by James Hammelman.

MR. HAMMELMAN: Good morning, my name is Jim Hammelman, I'm a senior chemical process engineer at NMSS, Division of Fuel Management. I'll be discussing the Staff's review and evaluation of SHINE's chemical hazards analysis for their planned medical isotope production facility.

The primary objective of the NRC review was to examine SHINE's identification and evaluation of chemical hazards to the public. Review was focused on those chemical hazards that are under NRC's

regulatory jurisdiction and the review will support the Staff's licensing decision under the requirements of Part 50.

The Staff evaluated the design against the chemical safety criteria presented in Section 3.1 of the FSAR. These criteria are more restrictive than those identified in the ISG that augments NUREG-1537.

Next slide, please. The Staff reviewed the SHINE description of the processes, the equipment, the facilities used for irradiated material processing that are presented in the FSAR.

The Staff noted the small scale of operation, the shielded cells used for irradiated material processing and the controls placed on inventory of toxic and reactive chemicals.

The Staff also reviewed the accident sequences identified and analyzed in the SHINE safety analysis and in the FSAR. The Staff found the identified accident sequences to be reasonable and consistent with the process and facility information presented in the FSAR.

The Staff performed independent analysis of the consequences to offsite individuals from chemical releases identified in the SHINE accident analysis. Staff's independent analysis supports the

SHINE conclusion that public exposure would be minimal.

More specifically, the predictive concentrations are less than those that would produce mild transient health effects.

The Staff reviewed previous analysis of similar operations and found that its conclusions are consistent with those made by the Staff in its 1987 evaluation of the Cintichem facility, which also produced moly-99.

And the conclusions are also consistent with those made by DOE in an IES that it prepared when it was considering medical isotope production. Next slide, please. The Staff also performed independent analysis of the impacts of chemical releases on SHINE plant personnel.

The Staff used near-term fuel dispersion estimates which were developed by SHINE and reviewed and accepted by the NRC Meteorological Staff and information about airflows through the control room.

In this case also, the Staff agrees with SHINE's conclusion that worker exposure would be less than those that would produce irreversible or other serious health effects, which are the criteria listed in Section 3.1.

Next slide, please.

Overall, the chemical safety review concluded that SHINE's process, the facility design features and operational controls provide reasonable assurance that SHINE will meet its chemical safety design criteria presented in Section 3.1 of the FSAR and that public health and safety will be adequately protected from chemical hazards that are under NRC's regulatory jurisdiction.

I'll return the mic to Elijah unless there's any questions?

MEMBER HALNON: This is Greg, just one question, and you can tell me this isn't the right spot.

I understand when we look at the radiological portion and we have to respond to accidents and do certain things, we look at the chemical portion and we see a chemical spill or something and we have to do certain things.

When we put those two together, is there any conflict in mitigative actions that have to be taken or should be taken where we're responding to a chemical problem and there's a radiological issue that we have to deal with, or anything vice versa?

Does that make any sense?

MR. HAMMELMAN: I think what I hear you asking is if there's concurrent accidents, radiological and chemical --

MEMBER HALNON: Is there any conflict in the actions?

MR. HAMMELMAN: Yes, and I guess it's that concurrent accidents were not analyzed. So, I suppose in some cases there might be but there's nothing that's obvious to me.

In the case of the chemical analysis, we were just focusing on what happens if the worker stays in place for a little while before he evacuates. There was no chemical response other than flee, for the workers.

DR. BLEY: I want to expand on Greg's question a little. I think that's something you folks ought to really have on your ticket to look at when you review or spot-check the procedures later on in this process.

We've seen several events occur at operating nuclear reactors where a fire, as it evolved, has led to other situations and the fire procedures were kind of written independently of the other emergency procedures and the operators got in a bit of a bind because the fire procedures took away

people that were urgently needed for what else was going on in the control room.

So, looking through the procedures to see if you get in places where your staffing for either side of these events is challenged is something that needs to be done at some point in time. It's not here, it's later.

MEMBER HALNON: And Dennis, even going further, I was thinking more of the co-located facilities that we may be looking at down the road where the chemical hazard is much greater, or maybe even worse than the radiological hazard.

DR. BLEY: Good point.

CHAIR BALLINGER: This is Ron Ballinger.

I'd like to reiterate that, I know of a bunch of cases, non-nuclear, where a lack of knowledge of issues in a facility by, in particular, fire people, offsite fire brigades and stuff, has resulted in serious injury or death because of that lack of knowledge.

And so we're getting into, as Dennis said, technologies that the Staff will be reviewing where that interface between chemical technology and nuclear technology may be more evident.

MEMBER REMPE: In listening to this, I'm

wondering if after we finish this review we ought to do a lessons learned letter on insights that could affect other applications.

And again, it doesn't have to be the same math but I think we have to make a list and look at and do such a letter in addition to the SHINE review.

CHAIR BALLINGER: We've certainly had enough conversation that we ought to probably think about that.

MR. MUNSON: This is Jeremy Munson, I would just add that to a degree we do consider things like that in criticality safety. For example, in moderator-controlled areas or in areas where you're primarily relying on moderation control.

Fire-fighting requirements in the event of a fire are limited to mists, they can't do solid streams.

We consider things like whatever suppression agent they're using and the fire suppression system whenever we do the safety evaluations in terms of what type of moderation they provide, reflection, things like that.

So, to a degree we do do that in criticality.

CHAIR BALLINGER: But that's an internal

evaluation onsite with people that have been trained. A lot of times, where the problem occurs is when you get an external interaction with the public where they need help in an area where they haven't been trained or don't have any knowledge of.

That's where the problem arises. Anyway, that's my story and I'm sticking to it.

MR. DICKSON: This is Elijah. I can move on to Slide 30, onto some confirmatory analyses.

The Staff did audit SHINE safety basis calculations, we performed a sampling of reviews of their safety basis calculations and documents to verify modeling assumptions, methodologies used, and input values used for the design basis accident analyses.

We did perform some confirmatory analyses in areas where we felt that it was prudent. We did perform simplified target solution inventory calculations using MCMP and ORIGEN, and we did confirm their results for the most part.

We performed, as Mike Salay had discussed, transport calculations using simplified iodine evolution and transport models. Our meteorologist did assess the meteorological data and confirmed chi over Q factors.

And then lastly, radiological consequences were confirmed using the NRC SNAP/Rad Trad Code Version 4.0 to confirm their MHA dose results.

MEMBER REMPE: This is Joy and I appreciate you adding this slide. In the SC, there were some inferences that there was some confirmatory calculations, but I appreciate the additional detail here.

Tell me, is there a file system that if four years from now somebody wants to look up what the Staff did, it's connected to your SE in the process that you did for this evaluation?

MR. DICKSON: No, to answer your question there is not a central file. For many Staff analyses we do have folders that we maintain and keep results in there.

Specific to SHINE, though, a lot of the information that we utilized, the confirmatory analyses needed to be destroyed after we were using it.

We have an agreement with them to review certain documents and whatnot on their portal. And so some of those calculations do need to be destroyed afterwards. But for a lot of other calculations such as power reactors, we do maintain files for that type

of work.

MEMBER REMPE: Thank you for your response, although I'm not sure it leaves me happy.

MR. DICKSON: It's a matter of there's proprietary information that we're reviewing and specific information with regards to the design and only certain Staff need to review those types of need-to-know-type analyses.

Onto Slide 31 if there's no other questions, this slide will be handled by both Mike Call and myself. Mike, would you like share your evaluations and findings and conclusions?

MR. CALL: Sure, this is Mike Call in the NSSA group review for NRC. Based on the review, as was explained in earlier slides, the Staff was able to make the findings you see here.

The NSSA method is an acceptable method and supports the adequate identification of capabilities and features to prevent or mitigate the accidents and then protect the health and safety of the public and workers.

And it provides reasonable assurance that SHINE has identified accidents as required for prevention and mitigation, and they have established appropriate safety-related controls.

I'll turn that back to you, Elijah.

MR. DICKSON: I'll start by saying that for a facility such as SHINE, we did find the margin to safety is large and that there are few credible accidents that can sufficiently damage the system that would result in a major release of radioactive material to the unrestricted area.

With that, the Staff found reasonable assurance that SHINE meets the siting criteria for public health and safety and that we also found reasonable assurance that the control room habitability requirements for radiological consequences have also been met.

And with that, that concludes our presentation today and we can field any other questions you may have.

CHAIR BALLINGER: Questions from Members? Okay, this is a break-point session. After this, which I'm sure will be after lunch, will be the -- yes, I'm getting there. So, now it's time for public comments.

If there are members of the public that are out there that would like to make a comment, you may have to use star 6 to unmute your phone, or if you're logged in through Teams, please state your name

and make your comment, please.

Hearing none, I think we're at an end to this part of the session pretty much conveniently, or whatever, just like yesterday. We don't have enough time to switch before lunch to go into the closed meeting because I think we're going to have longer discussions than a half an hour.

So, I think we will recess the meeting until 1:00 p.m., thank you very much.

(Whereupon, the above-entitled matter went off the record at 11:28 a.m.)

Chapter 13 – Accident Analysis (Open Session)

TRACY RADEL, VICE PRESIDENT OF ENGINEERING

SHINE Safety Analysis (SSA) Methodology

- SHINE applies a SHINE-specific, risk-based methodology similar to the guidance described in NUREG-1520, Standard Review Plan for Fuel Cycle Facilities License Applications, in the development of the detailed accident analysis.
  - This methodology is applied to both the irradiation facility (IF) and the radioisotope production facility (RPF) for consistency of the safety analysis across the facility.
- The SSA is developed based on the following major steps:
  - Identification and systematic evaluation of hazards at the facility
  - Comprehensive identification of potential accident/event sequences that would result in unacceptable consequences, and the expected likelihoods of those sequences
  - Identification and description of safety-related controls (i.e., structures, systems, components, or specific actions) that are relied on to limit or prevent potential accidents or mitigate their consequences
  - Identification of programmatic administrative controls that ensure the availability and reliability of identified safety systems
  - Assessment of radiological and chemical consequences for postulated accident sequences to demonstrate compliance with acceptable limits

Acceptance Criteria

CONSEQUENCE ANALYSIS METHODOLOGY

- SHINE Safety Criteria:
  - An acute worker dose of 5 rem or greater total effective dose equivalent (TEDE).
  - An acute dose of 1 rem or greater TEDE to any individual located outside the owner controlled area.
  - An intake of 30 milligrams or greater of uranium in a soluble form by any individual located outside the owner controlled area.
  - An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could lead to irreversible or other serious, long-lasting health effects to a worker or could cause mild transient health effects to any individual located outside the owner controlled area.
  - Criticality where fissionable material is used, handled, or stored (with the exception of the target solution vessel).
  - Loss of capability to reach safe shutdown conditions.

Process Hazard Analysis and Accident Sequence Development

- Identification of relevant accident categories
  - Relevant accident categories as identified in the interim staff guidance (ISG) augmenting NUREG-1537 are carried forward
  - Hazard evaluations identify potential initiating events, consequences, and controls that may be applied
  - Hazard evaluations also identify SHINE-specific accident types (e.g., tritium, neutron driver)
- Process hazard analysis (PHA) for internal and external events
  - Identify accident sequences based on the hazard evaluation results and the ISG guidance
  - Estimate a risk index for each potential unmitigated accident sequence (likelihood x consequences)
  - Identify engineered and administrative controls for those sequences which have an unacceptable risk
  - Evaluate controlled risk indices crediting risk reduction from controls
  - Develop list of safety-related controls

Process Hazard Analysis and Accident Sequence Development

- IF accident categories:
  - Maximum hypothetical accident (MHA)
  - Insertion of Excess Reactivity
  - Reduction in cooling
  - Mishandling or malfunction of target solution
  - Loss of off-site power
  - External events
  - Mishandling or malfunction of equipment
  - Large undamped power oscillations
  - Detonation and deflagration in the primary system boundary
  - Unintended exothermic reaction other than detonation
  - System interactions
  - Facility-specific events (i.e., neutron driver assembly system [NDAS], tritium purification system [TPS], and heavy load drop events)
- RPF accident categories:
  - Critical equipment malfunction
  - Inadvertent nuclear criticality
  - RPF fire (i.e., carbon delay bed fire, carbon guard bed fire)
  - Hazardous chemicals (e.g., uranium uptake)
- External event accident categories:
  - Seismic event
  - Severe weather (e.g., tornado, high winds, heavy snow, lightning)
  - External flooding events (i.e., probable maximum precipitation)
  - External fire events (e.g., vegetation, natural gas, vehicle fires)
  - Transportation accidents (e.g., aircraft impact, chemical truck accident)
  - Flooding events internal to the IF and RPF
  - On-site chemical/gas releases (e.g., spills)
  - Fire events internal to the IF and RPF are evaluated on a fire area basis

SHINE

Risk Matrix Development

Consequence Category	Facility Staff	Offsite Public
High Consequence 3	RD > 100 rem CD > PAC-3	RD > 25 rem 30 milligrams sol U intake CD > PAC-2
Intermediate Consequence 2	5 rem < RD ≤ 100 rem PAC-2 < CD < PAC-3	1 rem< RD ≤ 25 rem PAC-1 < CD ≤ PAC-2
Low Consequence 1	Accidents with lower radiological and chemical exposures than those above	Accidents with lower radiological and chemical exposures than those above

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

Likelihood Category	Likelihood Index (T)	Event Frequency Limit	Risk Index Limits
Highly Unlikely	1	Less than 10^{-5} per event, per year	T ≤ -5
Unlikely	2	Between 10^{-4} and 10^{-5} per event, per year	-5 < T ≤ -4
Not Unlikely	3	More than 10^{-4} per event, per year	-4 < T

SHINE

Likelihood Evaluation

Failure Frequency Index Number (FFIN)	Based on Evidence	Based on Type of Control	Comments
-6	External event with freq. < 10^{-6}/yr	N/A	If initiating event, no controls needed.
-5	Initiating event with freq. < 10^{-5}/yr	N/A	For passive safe-by-design components or systems; failure is considered highly unlikely for robust passive engineered controls: 1. Whose dimensions fall within established single parameter limits or that can be shown by calculation to be subcritical including the use of the approved subcritical margin, 2. That have no credible failure mechanisms that could disrupt the credited design characteristics, and 3. Whose design characteristics are controlled so that the only potential means to effect a change that might result in a failure to function would be to implement a design change.
-4	No failures in 30 years for hundreds of similar controls in industry.	1. Exceptionally robust passive engineered control (PEC), 2. Two independent active engineered control (AECs), PECs, or enhanced specific administrative control (SAC)	Rarely can be justified by evidence. Further, most types of single control have been observed to fail.
-3	No failures in 30 years for tens of similar controls in industry.	A single control with redundant parts, each a PEC or AEC	None
-2	No failure of this type in the facility in 30 years.	A single PEC	None
-1	A few failures may occur during facility lifetime.	1. A single AEC 2. Enhanced SAC 3. Redundant SAC	None
0	Failure occur every 1 to 3 years.	A single SAC	None
1	Several occurrences per year.	Frequent event, inadequate control	Not for controls, just initialing events.
2	Occurs every week or more often.	Very frequent event, inadequate control	Not for controls, just initialing events.

Failure Probability Index Number (FPIN)	Probability of Failure on Demand	Based on Type of Control	Comments
-6	10^{-6}		If initiating event, no control needed.
-4 or -5	10^{-4} - 10^{-5}	1. Passive engineered control (PEC) with high design margin. 2. Inherently safe process. 3. Two redundant controls more robust than a simple AEC, PEC, or enhanced SAC.	Can rarely be justified by evidence. Most types of single controls have been observed to fail.
-3 or -4	10^{-3} - 10^{-4}	1. Single PEC 2. Single AEC with high availability	None
-2 or -3	10^{-2} - 10^{-3}	1. Single AEC 2. Enhanced SAC 3. SAC for routine planned operations	None
-1 or -2	10^{-1} - 10^{-2}	A SAC that must be performed in response to a rare unplanned demand.	None

Duration Index Number (DIN)	Average Failure Duration	Duration in Years	Comments
1	> 3 years	10	
0	1 year	1	
-1	1 month	0.1	Formal monitoring to justify indices < -1
-2	A few days	0.01	
-3	8 hours	10^{-3}	
-4	1 hour	10^{-4}	
-5	5 minutes	10^{-5}	

SHINE

Safety-Related Controls

- The types of safety-related controls that are credited for prevention and/or mitigation of accident sequences are:
  - Engineered controls (active or passive), identified as safety-related structures, systems, and components (SSCs); and
  - Specific administrative controls (e.g., procedural controls).
- Programmatic administrative controls are also implemented to assure that safety-related controls can perform their intended functions.
- Defense-in-depth (DID) controls may also be identified that are not credited in accident sequences but provide additional margin for risk reduction.

SHINE

Accident Analysis and Determination of Consequences

METHODS

- Radiological consequences are determined for members of the public and control room operators
- Process includes:
  - Calculation of inventories
  - Definition of accident-specific material at risk (MAR)
  - Transport of radionuclides
  - Development of accident source terms
  - Conversion to radiological dose
- Generally, worker and public doses are calculated over a 30-day interval
  - The scenario resulting in the release of tritium into the tritium confinement boundary uses a 10-day interval because it is expected that tritium recovery can be accomplished in this time frame

SHINE

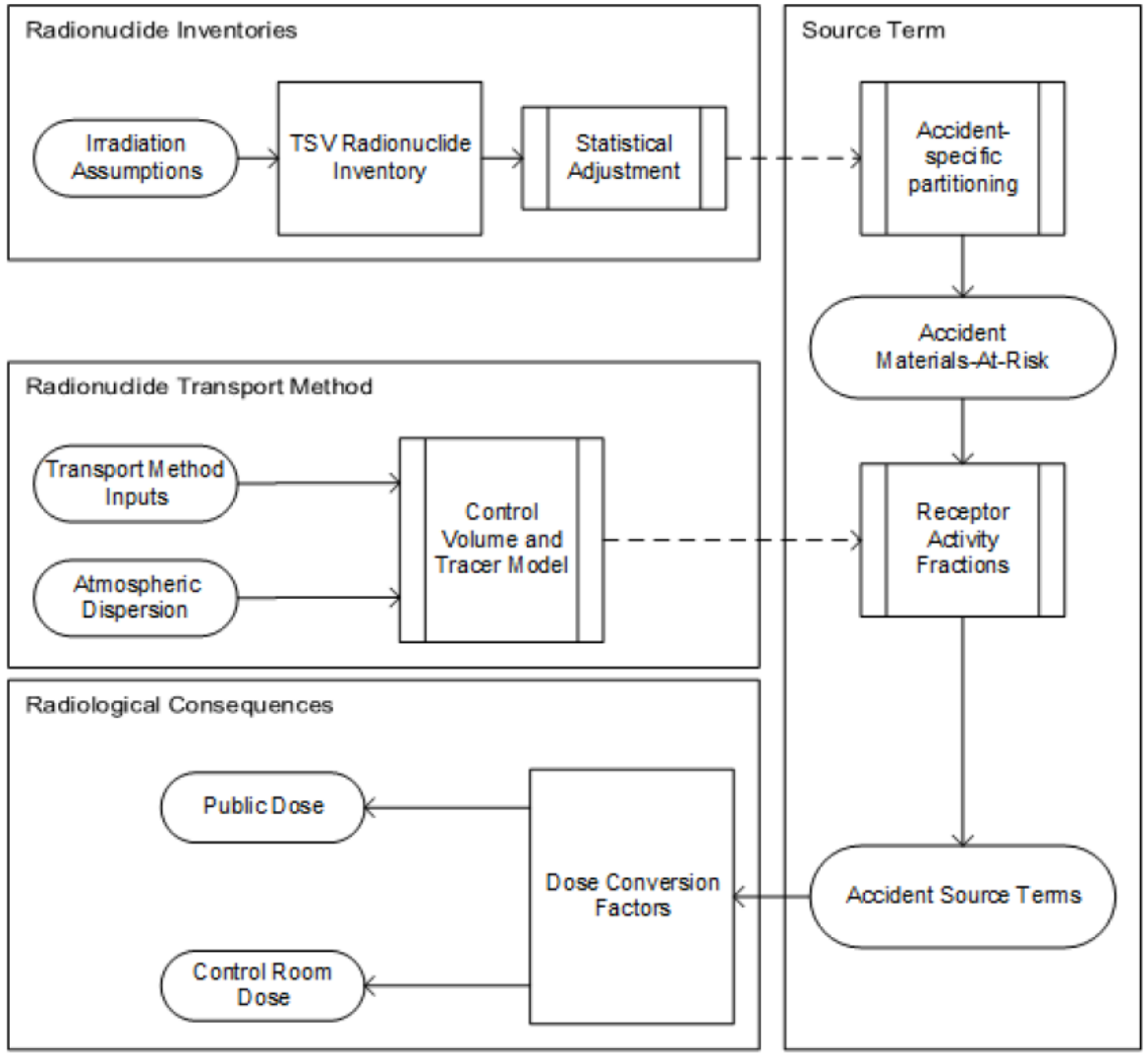
Accident Analysis and Determination of Consequences

CONSERVATISMS

- Conservatisms applied in the dose analysis include, but are not limited to:
  - Bounding TSV power history and operational cycle
  - Minimum nuclide decay times
  - Times to transport nuclides out of process systems are neglected
  - Condensation conservatively neglected
  - Non-credited filtration neglected
- Atmospheric dispersion values (χ/Q) used are 95^{th} percentile

SHINE

Accident Analysis and Determination of Consequences



Hazardous Chemicals

- Chemical hazards of licensed material, hazardous chemicals interacting with licensed material, and hazardous chemical produced from licensed materials are evaluated in the SSA
  - These do not include substances prior to process addition to licensed materials or after process separation from licensed materials
- Hazardous chemical consequence assessment is performed to demonstrate that potential consequences meet the SHINE Safety Criteria for the public and workers (RCA worker and control room operator)
- The PAVAN computer code is used to perform consequence analysis for the public and nearest residence
  - Chemical exposure to both receptors is calculated using the 95^{th} percentile χ/Q values

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Advisory Committee on Reactor Safeguards

SHINE Medical Technologies, LLC
Operating License Application

Chapter 13 - Accident Analysis

Elijah Dickson, NRR/DRA/ARCB
Mike Call, NMSS/DFM/NARAB
James Hammelman, NMSS/DFM/CTCFB
U.S. Nuclear Regulatory Commission

May 18, 2022

Review Team

- Technical Reviewers and Presenters
  - Elijah Dickson – NRR/DRA/ARCB
  - Mike Call – NMSS/DFM/NARAB
  - James Hammelman – NMSS/DFM/CTCFB
  - Jeremy Munson – NMSS/DFM/NARAB
  - Kevin Quinlan – NRR/DEX/EXHB
  - Michael Salay – RES/DSA/FSCB
- Project Managers
  - Michael Balazik – NRR/DANU/UNPL
  - Molly-Kate Gavello – NRR/DANU/UNPL
  - Holly Cruz – NRR/DANU/UNPL

Contents

- Background and Review Approach
- SHINE Safety Analysis
- Design Basis Accident Analyses
- Chemical Safety
- Audits and Confirmatory Analyses
- Evaluation Findings and Conclusion

Background and Review Approach – **Regulatory Requirements and Commitments**

- 10 CFR 50.34, “Contents of applications; technical information,” paragraph (b), “Final safety analysis report.”
- 10 CFR 50.36, “Technical Specifications.”
- 10 CFR 50.40, "Common Standards," paragraph (a).
- 10 CFR 50.57, "Issuance of Operating License," paragraph (a)(3).
- Commitments to 10 CFR Part 70-like requirements.

Background and Review Approach – **Regulatory Guidance**

- NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content," issued February 1996
- NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria," issued February 1996
- "Final Interim Staff Guidance [ISG] Augmenting NUREG-1537, Part 1, 'Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,' for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," dated October 17, 2012
- "Final Interim Staff Guidance Augmenting NUREG-1537, Part 2, 'Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,' for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," dated October 17, 2012

Background and Review Approach – **Regulatory Guidance (Cont'd)**

- NUREG-1520, Rev. 2, "Standard Review Plan for Fuel Cycle Facilities License Applications," issued June 2015
- NUREG-1513, "Integrated Safety Analysis Guidance Document," issued May 2001
- NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," issued March 1998
- NUREG/CR-2858, "PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," issued November 1982
- Regulatory Guide 1.145, Rev. 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," issued February 1983
- NUREG/CR-6331 Rev. 1, "Atmospheric Relative Concentrations in Building Wakes," issued May 1997

Background and Review Approach – **Review Methodology**

SHINE presented two types of safety analyses for the staff to review:

1. SHINE Safety Analysis (SSA)

   Purpose: Systematic analysis of facility processes used to identify and evaluate facility hazards associated with the processing and possession of licensed materials.

2. Design Basis Accident (DBA) Analyses

   Purpose: Evaluate the design and performance of structures, systems, and components (SSCs) of the facility with the objective of assessing the radiological consequences resulting from operation of the facility.

SHINE Safety Analysis - **Overview**

- ISG Augmenting NUREG-1537
  - 10 CFR Part 70 and NUREG-1520 Integrated Safety Assessment (ISA) methods acceptable to demonstrate safety
  - Alternatives are acceptable if adequacy to ensure safety is demonstrated

- SHINE's approach: use of ISA methods with differences

- Differences in terminology but methodology are similar in content, function, analysis (e.g., SSA Summary, safety-related controls)

- SSA Summary: applicant-controlled document; not submitted on docket

SHINE Safety Analysis – **Review Approach**

- Similarities to NUREG-1520 approach
- Evaluate alternatives to NUREG-1520 approach
- Considerations of review: applicable regulatory requirements and unique aspects of facility and the application
- Ensure adequacy of safety program, including SSA method and implementation
  - Broad (horizontal) review – evaluate method and safety program
  - Narrow (vertical) reviews of select accident types – evaluate method implementation
  - Reliability management measures

SHINE Safety Analysis – **Method and Implementation**

- SSA Method:
  - Identify, evaluate facility hazards – HAZOP, FMEA.
  - Identify credible accident sequences, define credible.
  - Assess radiological, chemical consequences and likelihoods.
  - Identify, describe safety-related controls to prevent/mitigate accidents, meet SHINE Safety Criteria.
  - Identify reliability management measures, programs for establishing and maintaining these measures.
- Method Implementation
  - Assess by review of select accident types (facility-specific events, external hazards, equipment mishandling/malfunction, and select events for criticality).
- Safety Program: SSA and SSA method are an important element; effective = reflect as-built, as-operated facility, demonstrate ensures health, safety of public and personnel.

Design Basis Accident (DBA) Analyses – **SHINE Facility Highlights**

- Typical design philosophy of defense-in-depth and multiple barriers.
- Eight independent accelerator-driven subcritical assemblies.
- Seismic qualified design features.
- TSV operates at relatively low power density, temperature and negative pressure.
- All DBAs which trip the TSV reactivity protection system (TRPS) results in an immediate safe shutdown condition.
- Light water pool has sufficient capacity to passively handle decay heat following a trip.
- Nominal source term is small. Material at Risk source term (safety-basis) is conservative, given aggressive modeling assumptions.

DBA Analyses – **Design Criterion and Siting Criteria**

(Presented as NUREG-1537 and the ISG guides staff to review)

Design Criterion 6 – Control room

> "A control room is provided from which actions can be taken to operate the irradiation units safely under normal conditions and to perform required operator actions under postulated accidents."

Siting and Control Room accident dose acceptance criteria:

1. Radiological consequences to an individual located in the unrestricted area following the onset of a postulated accidental release of licensed material would not exceed 1 rem total effective dose equivalent (TEDE) for the duration of the accident, and
2. Radiological consequences to workers [control room operator] do not exceed 5 rem TEDE during the accident.

DBA Analyses – **Methodology**

SHINE DBA analyses are consistent with NUREG/CR-6410 methodology.

Generally divided into six parts:

1. Select bounding design basis accidents;
2. Derive applicable accident source terms;
3. Identify major SSCs intended to mitigate the radiological consequences;
4. Estimate fission product release characteristics to the environment;
5. Review meteorological characteristics; and
6. Calculate radiological consequences from the bounding DBAs.

Generally, the staff does not accept DBA analyses that credit facility features that:

- are not safety-related;
- are not covered by technical specifications;
- do not meet single-failure criteria; or
- rely on the availability of offsite power.

DBA Analyses – **Identified Applicable DBAs**

SHINE DBA analyses are consistent with the ISG.

1. Maximum hypothetical accident (MHA) (Subsection 13a2.1.1);
2. Excess reactivity insertion (Subsection 13a2.1.2);
3. Reduction in cooling (Subsection 13a2.1.3);
4. Mishandling or malfunction of target solution (Subsection 13a2.1.4);
5. Loss of offsite power (LOOP) (Subsection 13a2.1.5);
6. External events (Subsection 13a2.1.6);
7. SHINE MHA - Mishandling or malfunction of equipment (Subsection 13a2.1.7);
8. Large undamped power oscillations (Subsection 13a2.1.8);
9. Detonation and deflagration in the primary system boundary (Subsection 13a2.1.9);
10. Unintended exothermic chemical reactions other than detonation (Subsection 13a2.1.10);
11. System interaction events (Subsection 13a2.1.11); and
12. Facility-specific events (Subsection 13a2.1.12).

DBA Analyses – **Identified Applicable DBAs (Cont'd)**

13. MHA (FSAR Section 13b.1.2.1) (see section 13a2.1.7);
14. Loss of Electrical Power (FSAR Section13b.1.2.2);
15. External Events (FSAR Section 13b.1.2.3);
16. Critical Equipment Malfunction (i.e., Malfunction or Mishandling of Equipment) (FSAR Section 13b.1.2.4);
17. Inadvertent Nuclear Criticality in the RPF (FSAR Section 13b.1.2.5);
18. RPF Fire (FSAR Section 13b.1.2.6); and
19. Hazardous Chemical Accidents (FSAR Section 13b.1.2.7).

DBA Analyses – **Materials at Risk - Accident Source Terms**

- Two types of Materials at Risk (MAR):
  1. Fission-product based
     - "safety-basis source term" derived for the TSV inventory.
  2. Tritium
     - Based on maximum quantities at the facility or used by an irradiator assembly.

- Primary codes and methods used to derive the MAR:
  1. Los Alamos, Monte Carlo N-Particle 5 (MCNP5), version 1.60.
  2. Oak Ridge, Standardized Computer Analyses for Licensing Evaluation (SCALE), version 6.1.2, ORIGEN-S.

- The staff finds these computer codes acceptable for the purposes of developing radionuclide inventories to derive a bounding SHINE-specific MAR.

- Accident-specific source terms are consistent with "five-factor" formula methodologies described in NUREG/CR-6410.

DBA Analyses – **Materials at Risk (MAR) – Accident Source Terms (Cont'd)**

- Verified operational assumptions with additional margin:
  - Corresponding fission power
  - Irradiation time per cycle
  - Total time between irradiations
  - Extraction between irradiations
  - Length of target solution recovery

- Calculation includes effects from fission, transmutation, activation, and decay.

- Assessed burnup and radionuclide peaking.

- Very large margin between the MAR and normal operations.

- The staff finds that the conservative assumptions and treatment of uncertainty to justify the MAR are acceptable.

DBA Analyses – **MAR Transport and Mitigation (1/3)**

- Leak path factors (LPF) developed for each scenario by:
  - Identify major SSCs intended to mitigate the radiological consequences;
  - Estimate fission product release characteristics to the environment.

- Factors include important physical processes such as control volumes, volumetric flow rates, leakage through gaskets, pressure, barometric breathing, and removal processes.

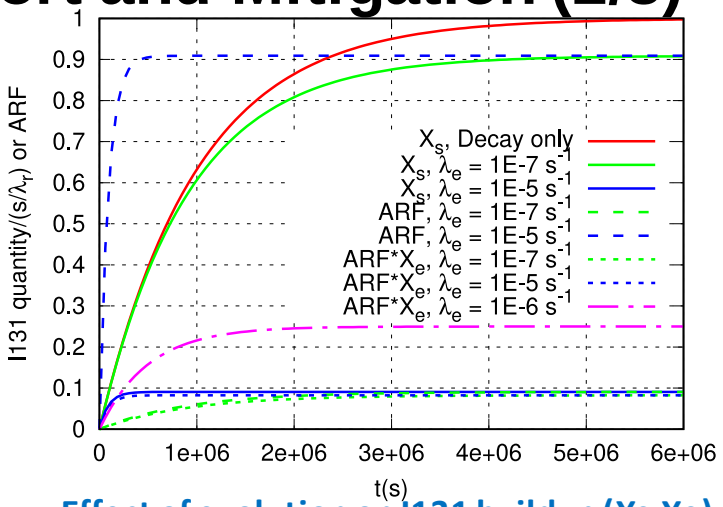
- Five-factor formula "damage ratio" is assumed to be 1 and "airborne release fractions" vary by accident.

- Environmental pathways are:
  - Confinement by IU cell or concrete cell -> IF building -> environment
  - Confinement by glove box -> IF building -> environment
  - Confinement by hot cell -> RPF building -> environment
  - Confinement by concrete vault -> RPF building -> environment

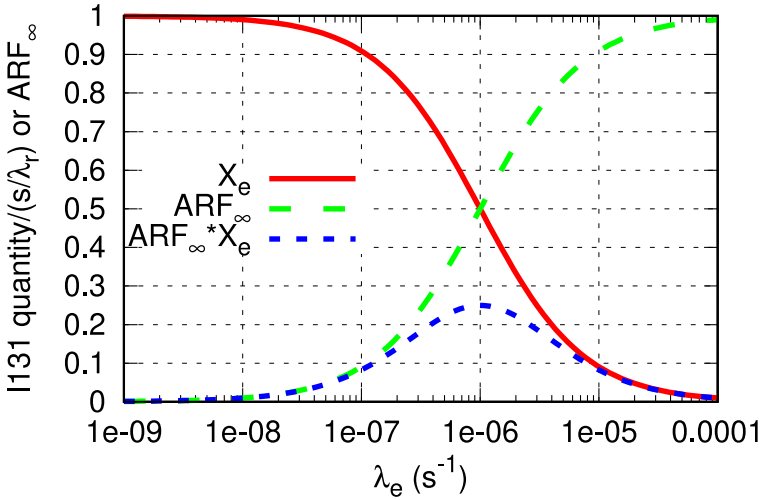
- LPFs are generally consistent with the methods described in NUREG/CR-6410.

DBA Analyses – **MAR Transport and Mitigation (2/3)**

- Iodine that evolves from solution to gas space can leak to environment.
  - Significant contributor to dose.
  - Geometry, temperature, flow.
- Iodine also evolves from solution during normal operation.
- If evolution is fast, it depletes inventory available for release to the environment.
  - Reduction in MAR by evolution not accounted for by inventory codes such as SCALE or MCNP.
- If evolution is slow, most ^{131}I will decay before evolving during an accident.
  - Limit on ARF
- Limit on Xe,MAR*ARF for ^{131}I.
- It is conservative to neglect evolution rate effects.



Effect of evolution on I131 buildup (Xs,Xe) and airborne release fraction



U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

DBA Analyses – **MAR Transport and Mitigation (3/3)**

Evolution and Transport Analyses:

- Evolution transport analysis to compare to SHINE pool release calculation.:
  - All iodine isotopes.
  - Stopped transport analysis after getting substantially lower ARF then SHINE. ARF*LPF
  - Reviewed technical bases for flow and LPF parameters.

- Influence of evolution on combined MAR*ARF (evolution vs decay):
  - Evaluation of the reduction in MAR due to evolution and adsorption during normal operation.
  - Evaluation of the release fraction to gas during a postulated accident scenario, ARF
    - Some time-dependent value between 0 and 1.

- SHINE neglecting evolution/adsorption reduction in evaluation of MAR for accident analysis and assuming an iodine ARF of 1 for many accident scenarios eliminate the need for additional analyses by using most bounding assumptions.
  - Partial general analyses illustrate some of these effects.
  - Geometry and scenario-specific calculations can be used to estimate conservatism.

DBA Analyses - **Atmospheric Dispersion**

- Developed short-term atmospheric dispersion (χ/Q) factors using traditional Gaussian plume diffusion methodology.
- χ/Q were developed at the offsite public and control room receptor.
- Conservatively assumed ground level release.
- Utilized bounding short-term 95th percentile χ/Q values.
- Calculations performed with NRC computer program, PAVAN, which implements the guidance provided in RG 1.145.

DBA Analyses – **Radiological Consequences**

- Consequence results are consistent with the total effective dose equivalent (TEDE) methodology defined in 10 CFR 50.2 and 20.1003.

- Dose-conversion-factors utilized are consistent with regulations.
  - Federal Guidance Report 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion.”
  - Federal Guidance Report 12, “External Exposure to Radionuclides in Air, Water, and Soil.”

- For fission product-based accidents, the accident duration is assumed to be 30 days and tritium accidents for 10 days.

- Receptor locations assume no personal protection equipment or protective actions.

DBA Analyses – **SHINE Maximum Hypothetical Accident**

Mishandling or malfunction of equipment (SHINE Subsection 13a2.2.7)

- Most limiting scenario, “Failure of the TOGS Pressure Boundary Resulting in Release of Off-Gas into the TOGS Cell.”

- Failure of the TOGS portion of the primary system boundary could allow escape of fission product gases or hydrogen into the primary confinement boundary and the radiologically controlled area.

    - TOGS circulates sweep gas during the irradiation cycle, a portion of the iodine is removed by the zeolite beds, and hydrogen and oxygen are recombined by the catalytic recombiners.

DBA Analyses – **SHINE Maximum Hypothetical Accident (Cont'd)**

- MHA MAR uses 100% of the MAR halogens and noble gases.

- Identified the safety controls for this accident to be:
  - Primary confinement boundary;
  - Ventilation radiation monitors;
  - Nitrogen purge system;
  - Ventilation isolation mechanisms; and
  - Holdup volume in the RVZ1e.

- The calculated doses for the MHA scenario are the following:

  - Control Room Operator – 1,940 mrem (1.94 rem)
  - Maximum exposed member of the public – 727 mrem (0.727 rem)

- Staff finds results are within the acceptable limit siting criteria of 1 rem TEDE and the control room operator of 5 rem TEDE.

DBA Analyses –**Technical Specifications**

- SHINE Tech Spec. Limiting Safety System Settings are set to protect the Primary System Boundary.
- Revised Tech Spec. Section 3.4 to include an LCO for the primary confinement boundary.
- SHINE Tech Spec. Limiting Safety System Settings ≤ 85% power averaged over ≤ 45 seconds.
- TS 5.8.4 states that SHINE will conduct startup testing in accordance with the Startup Testing Program (FSAR Section 12.11) and submit a Startup Report to the NRC within 6 months of the completion of all startup testing activities.

Chemical Safety Review

- Evaluated impacts on public health and safety - criteria in 50.40
- Focused on chemical hazards under NRC's regulatory jurisdiction
- SHINE Safety Criteria for public exposure
  - Acute chemical exposure to an individual from licensed material or chemicals produced from licensed material that could cause mild transient health effects to an individual outside the owner-controlled area should be highly unlikely.

Chemical Safety Review (Cont'd)

- Staff reviewed SHINE SSA accident sequences using information about processes and facility details and stated chemical quantity limits. Staff finds accident sequences reasonable.

- Staff reviewed SHINE public consequence calculations, performed independent dispersion calculations. Staff agrees with SHINE conclusion – public exposure would be less than PAC-1 levels which could produce mild, transient health effects.

- Staff noted that public chemical risk conclusion is consistent with other relevant analysis of Mo-99 production operations.

Chemical Safety Review (Cont'd)

- Staff reviewed SHINE worker consequence calculations and performed independent dispersion calculations. While the staff used different analytical methods, the staff agrees with SHINE conclusion – worker exposure would be less than PAC-2 levels, which could produce irreversible or other serious health effects.

Chemical Safety Review (Cont'd)

- Staff review was coordinated with other reviewers, particularly meteorology and SHINE Safety Analysis reviewer.
- Staff concluded that SHINE process, facility design, and control features provide reasonable assurance that the public health and safety will be adequately protected from chemical hazards that are under NRC's regulatory jurisdiction.

Audits and Confirmatory Analyses

- Staff audited SHINE safety-basis design calculations and documents to verify methodology assumptions and input values.
- Select independent confirmatory analyses include:
  - MAR: simplified target solution inventory calculations using MCNP5 and ORIGEN confirmed results presented in design calculations.
  - Transport: simplified iodine evolution and transport calculations.
  - Meteorology: assessed meteorological data and confirmed χ/Q factors.
  - Consequences: confirmatory analyses with NRC SNAP/RadTrad Ver. 4.0 confirmed results presented in the Ch 13 FSAR.

Staff Evaluation Findings and Conclusions

- SHINE Safety Analysis (SSA)
  - SSA method is an acceptable method and supports adequate identification of capabilities and features to prevent/mitigate accidents and protect health, safety of public and workers.
  - SSA provides reasonable assurance SHINE has identified accidents that require prevention or mitigation and established appropriate safety-related controls.

- DBA Analyses
  - Staff found reasonable assurance that SHINE meets the siting criteria for public health and safety.
  - Staff found reasonable assurance that the control room habitability requirements for radiological consequences are met.

Acronyms

DBA	Design Basis Accident
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
HAZOP	Hazard and Operability Analysis
ISA	Integrated Safety Assessment
ISG	Interim Staff Guidance
LCO	Limiting Condition of Operation
LOOP	Loss of offsite power
LPF	Leak path factors
MAR	Materials at Risk
MCNP5	Monte Carlo N-particle 5
MHA	Maximum Hypothetical Accident
NRC	Nuclear Regulatory Commission
RG	Regulatory Guide
RPF	Radioisotope Production Facility
SSA	SHINE Safety Analysis
SSC	Structures, Systems, and Components
TSV	Target Solution Vessel
TEDE	Total Effective Dose Equivalent
TRPS	TSV Reactivity Protection System
TOGS	TSV Off-gas System

Chapter 13 Backup Slides

Startup Testing

- SHINE has a startup testing program to demonstrate operability of their systems, calibrate instrumentation, and make reactor physics measurements to confirm their design calculations.
- The startup testing plan provides an outline of the tests that will be performed.
- The reactor physics tests are support by calculations and the measured results will be compared to the design calculations.