

# **Official Transcript of Proceedings**

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UNITED STATES OF AMERICA  
NUCLEAR REGULATORY COMMISSION  
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS  
(ACRS)  
+ + + + +  
NuSCALE SUBCOMMITTEE  
+ + + + +  
WEDNESDAY  
NOVEMBER 20, 2019  
+ + + + +  
ROCKVILLE, MARYLAND  
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The Subcommittee met at the Nuclear  
Regulatory Commission, Two White Flint North, Room  
T2D10, 11545 Rockville Pike, at 8:30 a.m., Walter L.  
Kirchner, Chair, presiding.

COMMITTEE MEMBERS:

WALTER L. KIRCHNER, Chair  
DAVID PETTI, Co-Chair  
RONALD G. BALLINGER, Member  
DENNIS BLEY, Member  
CHARLES H. BROWN, JR. Member  
VESNA B. DIMITRIJEVIC, Member

1 JOSE MARCH-LEUBA, Member  
2 JOY L. REMPE, Member  
3 PETER RICCARDELLA, Member\*

4  
5 ACRS CONSULTANTS:

6 MICHAEL L. CORRADINI  
7 STEPHEN SCHULTZ

8  
9 DESIGNATED FEDERAL OFFICIAL:

10 MIKE SNODDERLY

11

12

13 \*Present via telephone

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P-R-O-C-E-E-D-I-N-G-S

8:30 a.m.

CHAIR KIRCHNER: Good morning, the meeting will come together. This is a meeting of the Advisory Committee on Reactor Safe Guards NuScale Subcommittee. I am Walt Kirchner, Chairman of the NuScale Subcommittee.

My colleague member, David Petty, will Co-Chair the meeting with me today. Members in attendance are Ron Ballinger, Joy Rempe, Jose March-Leuba, Vesna Dimitrijevic.

And we expect Dennis Bley, I see a coffee cup, and Charlie Brown will join us, he's stuck in traffic. And Pete Riccardella, I believe is on the line.

MEMBER RICCARDELLA: I'm here.

CHAIR KIRCHNER: Thank you, Pete. Mike Snodderly is the designated federal official for this meeting. The Subcommittee will review the Staff's evaluation of NuScale topical report, TR-0915-17565, accident source term methodology.

Today we have members of the NRC Staff and NuScale to brief the Subcommittee. I also failed to mention that we have former members Mike Corradini and Steve Schultz with us as consultants.

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4           regulations found in Title 10 of the Code of Federal  
5           Regulations Part 7.

6           The Committee can only speak through its  
7           published letter reports. We hold meetings to gather  
8           information and perform preparatory work that will  
9           support our deliberations at a full committee meeting.

10          The rules for participation in all ACRS  
11          meetings were announced in the federal register on  
12          June 13th, 2019. The ACRS section of the U.S. NRC  
13          public website provides our charter, bylaws, agendas,  
14          letter reports and full transcripts of all full and  
15          subcommittee meetings. Including slides presented  
16          there.

17          The meeting notice and agenda for this  
18          meeting were posted there. Portions of this meeting  
19          can be closed, as needed, to protect proprietary  
20          information pursuant to 5 U.S.C. 552(b)(c)(iv).

21          As stated in the federal register notice,  
22          and in the public meeting notice posted to the  
23          website, members of the public who desire to provide  
24          written or oral input to the Subcommittee, may do so  
25          and should contact the designated federal official

1 five days prior to the meeting, as practicable.

2 We have also set aside ten minutes for  
3 comments from members of the public attending or  
4 listening to our meetings.

5 We have not received written comments or  
6 requests for time to make oral statements from members  
7 of the public regarding today's meeting.

8 A transcript of the meeting is being kept  
9 and will be made available on the ACRS section of the  
10 U.S. NRC public website.

11 We request that participants in this  
12 meeting please use the microphones located throughout  
13 the meeting room when addressing the Subcommittee.  
14 Participants should first identify themselves and  
15 speak with enough volume and clarity so that they can  
16 be readily heard.

17 A telephone bridge line has been  
18 established for the public to listen to the meeting.  
19 To minimize disturbance, the public line will be kept  
20 in a listen-in only mode.

21 Today we have an unusual situation so we  
22 want to ask all people on the phone line to mute their  
23 phones. To avoid disturbance, I request that  
24 attendees also put their electronic devices, like cell  
25 phones, in the off or noise free mode.



1                   We'll now proceed with the meeting. Dave,  
2                   would you like to make any comments?

3                   CO-CHAIR PETTI: No comments.

4                   CHAIR KIRCHNER: Okay, thank you. And  
5                   I'll call on Carrie Fosaaen of NuScale to begin  
6                   today's presentation. Thank you, Carrie.

7                   MS. FOSAAEN: Thank you.

8                   CHAIR KIRCHNER: Please proceed.

9                   MS. FOSAAEN: Good morning, I'm Carrie  
10                  Fosaaen from NuScale Licensing. I wanted to just  
11                  express gratitude for you allowing us to present our  
12                  accident source term methodology.

13                  And I did want to make you aware that due  
14                  to unfortunate circumstances one of our main  
15                  presenters is on the phone in Corvallis, so we will  
16                  try to work with that, but we have prepared a backup  
17                  plan should the phone connection not facilitate that.  
18                  So we are prepared in the eventuality.

19                  I would like to introduce our team  
20                  starting with Gary.

21                  MR. BECKER: Good morning, Gary Becker,  
22                  Regulatory Affairs Counsel.

23                  MR. OSBORN: Jim Osborn, NuScale  
24                  licensing.

25                  MR. GUINN: Paul Guinn, Radiological

1 Engineering, NuScale.

2 MS. FOSAAEN: All right. And Paul will  
3 kick us off with the accident source term overview,  
4 the nonproprietary portion of the meeting.

5 MR. GUINN: Thanks. This is the NuScale  
6 presentation of the NuScale accident source term  
7 licensing topical report.

8 I'm Paul Guinn, we made the introductions  
9 already. Mark Shaver is on the phone in Corvallis.  
10 All the rest are here.

11 And my mouse has quit. There we go.  
12 Okay, there was a problem with the document we have  
13 opened up right now.

14 (Off record comments.)

15 MR. GUINN: Can we proceed without it?

16 CHAIR KIRCHNER: For those on the phone  
17 line, we're just trying to get our view graph  
18 presentations up on the screen. So there's just a  
19 slight pause here.

20 MR. GUINN: Okay, I think we're back up  
21 and running. We're ready? Yes.

22 The intended purpose of this presentation  
23 is to provide a general overview of the NuScale  
24 accident source term methodology. Areas of discussion  
25 are principally based on methodology positions for

1 which NRC approval was specifically requested.

2 Additional expanded detail, including  
3 company proprietary information, will be available for  
4 presentation in closed session.

5 Provided here are some of the acronyms  
6 used throughout this presentation, which may not be  
7 common knowledge. It might be helpful to keep this  
8 slide separated and available for reference throughout  
9 the presentation.

10 Starting off, the high level unique  
11 calculation methodologies within the accident source  
12 term topical report include the atmospheric dispersion  
13 methodology, or core damage source term, containment  
14 aerosol transport and removal method, post-accident  
15 pH, temperature dependent pH, iodine spike design  
16 basis source term noted as the DBST and the  
17 environment qualification dose within the containment  
18 and bioshield envelope.

19 Additionally, calculation methodology is  
20 for industry standard design basis accident  
21 evaluations are contained within the accident source  
22 term topical report. Including rod ejection accident,  
23 fuel handling accident, main steam line break, steam  
24 generator tube failure and small line break accident  
25 methodologies.

1 A central distinction within the accident  
2 source term methodology is the classification of the  
3 core damage event as beyond design basis. This  
4 distinction is consider a requisite to use, of the  
5 core damage event as the maximum hypothetical event  
6 for control room and offsite dose evaluation. But not  
7 to use the core damage event as a design basis  
8 accident requiring radiation environment equipment  
9 qualification.

10 Instead of using core damage event for  
11 chapter equipment qualification purposes, it is used  
12 for Chapter 19 equipment survivability evaluation.  
13 Where any required functional capability of equipment  
14 in the core damage accident is assured.

15 Chapter 19 equipment survivability has  
16 been historically demonstrated through reference to  
17 the Chapter 3 environmental qualification evaluation.  
18 But for our application, the Chapter 19 survivability  
19 evaluation references the core damage accident as a  
20 standalone, beyond design basis event, a part from  
21 Chapter 3, design basis accident bases.

22 Another unique aspect of the methodology  
23 is implementation of a bounding surrogate design basis  
24 source term which is used as the bounding design basis  
25 accident to meet design basis requirements, such as

Chapter 3 environmental qualification. But is also evaluated alongside the core damage event to demonstrate acceptable radiological consequences for a maximum hypothetical accident.

DR. SCHULTZ: When you say, used in conjunction with or alongside of, can you elaborate on that as to how specifically that is done?

MR. GUINN: Sure. They are both evaluated. The iodine spike design basis source term is evaluated as a Chapter 15 accident. As the bounding design basis accident for the traditional, what would be fulfilled by the Reg Guide 1.183 Appendix A LOCA in previous applications?

And the core damage accident is evaluated and reported in the same space, in Chapter 15. Though it is not given as a design basis accident, it's held as a beyond design basis accident so that although they are reported together in 15 for evaluation against control room, offsite dose, acceptance criteria, then the core damage accident is not evaluated against, or for equipment, environmental qualification in Chapter 3.

So not alongside in Chapter 3, that's where they split, but they're alongside each other in Chapter 15.

1 DR. SCHULTZ: And alongside means you're  
2 going to compare one against the other and look for  
3 the maximum or --

4 MR. GUINN: No, we evaluate both and --

5 DR. SCHULTZ: And demonstrate that each  
6 are satisfactorily addressed by the design?

7 MR. GUINN: That's right.

8 DR. SCHULTZ: Okay. And when you say  
9 unique design basis event, iodine spike source term,  
10 there are certain unique features, but the iodine  
11 spike source term is a common approach in the Reg  
12 Guide 1.183, correct?

13 MR. GUINN: Correct.

14 DR. SCHULTZ: So, what specifically, when  
15 you say it's unique, what's unique for NuScale here?

16 MR. GUINN: Right. Yes, we'll unpack all  
17 of that in a couple of slides too.

18 DR. SCHULTZ: Okay.

19 MR. GUINN: We have dedicated specifically  
20 to iodine slide.

21 DR. SCHULTZ: That's good.

22 MR. GUINN: Yes.

23 DR. SCHULTZ: Thank you.

24 MEMBER BLEY: Paul, can you explain, at  
25 some level, the difference between testing

1 requirements and any other requirements between a  
2 equipment that in the normal sense is qualified, but  
3 in your sense is going to be subject to the  
4 survivability evaluation. Are there differences in  
5 reliability requirements testing, anything else?

6 MR. GUINN: Well, I think if I understand  
7 the question correct it's basically the difference  
8 between Chapter 3 equipment qualification criteria and  
9 Chapter 19 survivability --

10 MEMBER BLEY: And what follows after that,  
11 say during operation?

12 MR. GUINN: Right. I think the central,  
13 the main distinction is that when we port it to  
14 Chapter 19 space we are assuring it for the functional  
15 duration for which it's required.

16 And in traditional Chapter 3 environmental  
17 qualification space you follow a standard that  
18 prescribes just standard set chunks of time. But I  
19 don't want to speak out of term either. If we do have  
20 someone in Corvallis who has anything to add to this  
21 --

22 MEMBER BLEY: Let me specialize the  
23 question a little more. In qualification, you set  
24 criteria and you test against them but you kind of  
25 test until you pass and then you're qualified.

1 Will there be similar kinds of  
2 qualification testing for equipment survivability  
3 evaluation or how will you evaluate its ability to  
4 survive for the times and specific conditions that are  
5 required for each accident in Chapter 19?

6 MR. GUINN: Right. Multiple methods. We  
7 allow, and again, Corvallis please --

8 (Off microphone comment.)

9 MR. GUINN: Yes, Scott Weber, correct me  
10 if I need it, but I believe the answer is in 19 we do  
11 specify that we can assure reliability by testing for  
12 equipment survivability or analysis or other methods.  
13 Scott, does that sound accurate?

14 MR. WEBER: Yes. This is Scott Weber in  
15 Corvallis. That's right. Paul, that's accurate.

16 The equipment survivability assessment,  
17 it's first based off a comparison to do equipment  
18 qualification doses since there is still an equipment  
19 qualification program. And because the functional  
20 durations for survivability may be different than  
21 those for equipment qualifications.

22 The survivability assessment looks  
23 specifically at what function would be served during  
24 a severe accident. So, the first comparative is  
25 against equipment qualification doses. And if the EQ



1 dose is higher, that would also confirm survivability.

2 If the survivability dose is higher, than  
3 we are committed to some form of additional analysis  
4 or testing. Although the testing that is performed  
5 would not be held to the same IEEE standard as the  
6 equipment qualification testing would be because it's  
7 for beyond design basis in Chapter 19. It would be  
8 under a less stringent testing standards.

9 MEMBER BLEY: But there would be testing  
10 under adverse environments.

11 MR. WEBER: Yes, if necessary. There is  
12 equipment where the environment is more severe than  
13 for equipment qualification, then testing one of the  
14 methods that might be used to show survivability.

15 MEMBER BLEY: Thank you. Well, your  
16 answer leaves me a little confused. Will you, even  
17 though you're not using it in Chapter 3, you will be  
18 going through a qualification program similar to what  
19 people would do if they were using it in Chapter 3 but  
20 maybe at different times?

21 You keep saying there will be  
22 qualification testing, and I'm not quite sure what  
23 that means.

24 MR. WEBER: Right. So the survivability  
25 program, so much, does not exist at this time. That's

1 something that is shown at the application phase,  
2 similar to equipment qualification.

3 Yes, during a testing program similar to  
4 EQ is an option, but also because we are in Chapter 19  
5 design basis space, there is an option to do modeling  
6 or some other assessments that demonstrate  
7 survivability. Potentially using existing information  
8 from vendors.

9 So, testing is certainly one option that  
10 would be performed to demonstrate survivability but  
11 it's not the only option.

12 MEMBER BLEY: Is one of the options simply  
13 expert judgment?

14 MR. WEBER: No.

15 MEMBER BLEY: Or will you need some kind  
16 of testing?

17 MR. WEBER: There needs to be --

18 MEMBER BLEY: Okay.

19 MR. WEBER: -- some basis beyond judgment.

20 MEMBER BLEY: Thank you.

21 MR. GUINN: If there is nothing more we'll  
22 head to the next slide. This figure provides an  
23 overview of dose evaluation products that are provided  
24 by the accident source term topical report.

25 Accident source term topical report

1 content is shown within the hashed selection in this  
2 figure.

3 Some source term applications, which are  
4 extraneous to the topical report, are included in the  
5 upper right portion of the figure. Now, this is  
6 because this is intended to be a recurring figure  
7 throughout today's presentations and will be reused  
8 later to outline wider application of source terms in  
9 NuScale licensing submittal areas.

10 CO-CHAIR PETTI: So, just a question. I  
11 want to ask something outside the blue dash.

12 MR. GUINN: Okay.

13 CO-CHAIR PETTI: The one percent failed  
14 fuel, are you going to talk about that later?

15 MR. GUINN: Yes.

16 CO-CHAIR PETTI: Okay. I just need to  
17 understand what that is.

18 MR. GUINN: Sure.

19 DR. CORRADINI: And the 500 is an upper  
20 bound per 1.183?

21 MR. GUINN: That's correct. Yes.

22 DR. CORRADINI: Okay.

23 MR. GUINN: Yes.

24 MEMBER BLEY: And everything we'll hear  
25 today is based on a single module, is that right?

1 MR. GUINN: Yes, that's right. Yes.

2 MEMBER BLEY: Okay.

3 MR. GUINN: Okay. Well, for the purposes  
4 of the accident source term licensing topical report,  
5 the rod ejection accident methodology shown there, as  
6 it's described, assuming the results of a failure 1  
7 fuel assembly in order to establish a methodology.  
8 Any future application of the methodology would use  
9 the event specific amount of damage fuel.

10 DR. CORRADINI: Maybe I don't, I'm a  
11 little confused. But there is an assumed fuel failure  
12 fraction to compute the iodine spike. Is it not, I  
13 thought it was the same one percent, am I  
14 misremembering?

15 PARTICIPANT: No.

16 MR. GUINN: I'm sorry, the rod ejection,  
17 does this pertain to rod ejection?

18 DR. CORRADINI: I'm looking at your  
19 diagram; I'm looking at the arrow that TS PCA + 500x.

20 MR. GUINN: Right.

21 DR. CORRADINI: 500x of what fraction of  
22 fuel, because if I understand the physics behind it,  
23 I've got some failed fuel. There is steam there, as  
24 soon as I have the event, water comes back in and  
25 releases iodine into the coolant, activity of 500

1 times whatever it was.

2 So, what's the failed fuel fraction  
3 assumed in that box?

4 MR. GUINN: So, you see it cascades out of  
5 the, and really it would help, you know, I think we  
6 talk more to the origin. The bounding fuel isotopics  
7 in a later presentation --

8 DR. CORRADINI: Okay. Fine.

9 MR. GUINN: -- this figure is kind of  
10 unfortunate.

11 DR. CORRADINI: If we're going to get to  
12 it, that's fine.

13 MR. GUINN: Yes, definitely we are.

14 DR. CORRADINI: Okay, fine.

15 MR. GUINN: And actually, in the later  
16 day's presentation we have, I think a lot more detail  
17 on failed fuel fraction. Okay.

18 Next year is the summary of software  
19 program utilized in the accident source term  
20 methodology. ORIGEN-SCALE, ARCON96, RADTRAD, are all  
21 applied in all evaluated radiological events.

22 NRELAP5 is applied for evaluated primary  
23 coolant loss, design basis accidents. MELCOR,  
24 STARNAUA, the NuScale pHT Code and MCNP are applied in  
25 the core damage event evaluation.

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1 Moving on to one of the specific unique  
2 radiological evaluation methods within the accident  
3 source term topical report. One of these is the use  
4 of an adapted.

5 ARCON96 methodology for calculation of  
6 offsite dispersion factors chi over q values. ARCON96  
7 methodology is typically only applied for onsite chi  
8 over q evaluation.

9 Typical atmospheric dispersion evaluations  
10 for large light water reactors apply the PAVAN  
11 dispersion model for offsite chi over q values.

12 In the NuScale accident source term  
13 methodology, the applicability of existing guidances  
14 evaluated and the use of PAVAN is demonstrated to  
15 result in non-physical chi over q's for the small  
16 NuScale site boundary.

17 Application of ARCON96 methodology is  
18 demonstrated to be conservative for the design. PAVAN  
19 was developed for longer distances for offsite doses  
20 as existing plants whereas ARCON was developed for  
21 shorter distances of the control rooms at existing  
22 plants.

23 NuScale site boundary is more comparable  
24 to control room distances than offsite distances at  
25 existing plants.

1           Additionally, the ARCON96 methodology is  
2           adapted to incorporate some unique conservatism  
3           consistent with the Reg Guide 1.145, PAVAN  
4           methodology. Detail of these additional conservatisms  
5           is available for discussion in the closed session.

6           DR. SCHULTZ: Paul, let me know if you're  
7           going to cover this in the closed session or perhaps  
8           later, but for control room habitability, to you use  
9           ARCON or do you use NARCON?

10          MR. GUINN: We use NARCON, yes.

11          DR. SCHULTZ: NARCON is used for, so, with  
12          that modification for NARCON, as described in your  
13          documentation, that's what's used for all events?

14          MR. GUINN: Correct.

15          DR. SCHULTZ: Okay, thank you.

16          MR. GUINN: Next. Another unique aspect  
17          of the accident source term topical report is the core  
18          damage source term methodology.

19                 This is comprised of a mechanistic source  
20          term evaluation consistent with the recommendations of  
21          a 2012 nuclear energy institution position paper on  
22          small modular reactor source terms and a 2011 Sandia  
23          National Labs report, which was prepared to develop  
24          accident source terms using state of the art modeling.

25                 The mechanistic source term approach

1 involves derivation of core damage source term data  
2 from a range of Level 1 PRA severe accident scenarios.

3 A unique licensing approach is submitted  
4 for the core damage source term wherein the core  
5 damage event is treated as a beyond basis event that  
6 is evaluated in concert with a bounding surrogate  
7 design basis source term to constitute a bifurcated  
8 approach to the traditional maximum hypothetical  
9 accident evaluation.

10 DR. CORRADINI: Maybe this is not the  
11 place for the answer but the question I, there was a  
12 position, I can't remember all the positions, one of  
13 the positions was, this was a severe accident source  
14 term, staff came back and said, it's a source term.  
15 It's not severe, it's not design basis, the source  
16 term we're using for core damage.

17 What is so crucial about you identifying  
18 it continually as a severe accident source term? I'm  
19 a bit confused.

20 MR. GUINN: Yes. I believe we kind of set  
21 that up in one of the very first slides here where we  
22 named it as a central distinction. And maybe I failed  
23 to really drive home the critical or crucial nature of  
24 it.

25 But our belief is it's necessary in order



1 to implement the division that we do for the Chapter  
2 3, Chapter 19 separation of core damage event, that we  
3 rule it as a beyond design basis event. We understood  
4 it to be necessary.

5 And, Gary, do you have anything to add in  
6 on that?

7 MR. BECKER: Yes. Gary Becker. So, the  
8 primary reason for our distinction there, between a  
9 design basis event and a beyond design basis event, is  
10 because the rule for environment qualification, 10 CFR  
11 5449, requires that equipment, safety related  
12 equipment and certain non-safety related equipment, be  
13 qualified for the most severe design basis event.

14 DR. CORRADINI: Thank you.

15 MR. GUINN: Okay.

16 CHAIR KIRCHNER: I guess this is a point  
17 where I'll just make a statement. If you look at 10  
18 CFR 52, the maximum hypothetical accident, by its very  
19 nature, is beyond design basis of event for a large  
20 PWR or PWR as well.

21 So, it seems that your distinction is more  
22 about equipment survivability then a deviation from  
23 the regulations. The regulations are pretty clear  
24 that you assume a severe accident in developing your  
25 MHA and then go from there in doing your dose

1 assessment and consequence analysis.

2 MR. BECKER: If I may add to that. Gary  
3 Becker again. We don't think it's that clear that  
4 it's a beyond design basis event for the existing  
5 fleet.

6 In particular, if you look at the  
7 definition of safety related, it ties the SSCs, system  
8 structures components, that are used to mitigate the  
9 event that's evaluated for offsite release. The SSCs  
10 that are used to mitigate that event as classified as  
11 safety related.

12 So it actually infers, in our opinion,  
13 that the traditional so called design basis LOCA,  
14 which is a core melt, core damage event, we believe  
15 that the inference to that is typically true as a  
16 design basis accident.

17 MEMBER BLEY: No, no.

18 CHAIR KIRCHNER: Correct.

19 MEMBER BLEY: That was bothering me.

20 CHAIR KIRCHNER: You design a large -- the  
21 large PWR LOCA is a design basis event that does not  
22 result in core damage. And that's how you design the  
23 systems. That's the determination for all the systems  
24 that are safety related in a large LWR.

25 That has nothing to do with, or I think it

1 confuses the issue when you say that, for the large  
2 existing fleet, that that's a design basis, that's  
3 not, that's beyond design basis.

4 MEMBER BLEY: Yes. When I was reading  
5 this I was struck by the same thing, Walt. And Mr.  
6 Becker, you're here as counsel so you're digging into  
7 things we don't often dig into and maybe we don't need  
8 to, but there's a dichotomy in the regulation.

9 I mean, the event isn't a core damage  
10 event but when you look at equipment qualification and  
11 the like you assume you've had one and you have to  
12 survive that. And that's where their arguments get  
13 kind of fuzzy. And I don't think they're important to  
14 our findings, so we can avoid that.

15 MR. GUINN: Okay. Let's see. Sandia  
16 National Lab reports report SAND2011-0128 provides a  
17 set of radionuclide groups considered for the core  
18 damage source term. This reflects current state of  
19 the art knowledge of severe accident progression.

20 Release fractions here refers to the  
21 perception of total core isotopic inventory for a  
22 given chemical group released to containment in a core  
23 damage scenario.

24 Here are the, in this table, provided the  
25 fission product release information generated from our

1 MELCOR simulations for the five example surrogate  
2 accident scenario cases. This fission product  
3 released information for each of the example surrogate  
4 accident scenario cases as compared to the values from  
5 Reg Guide 1.183 and the release fractions from the  
6 Sandia report. SAND2011-0128.

7 DR. SCHULTZ: So, Paul, just on your  
8 previous slide, the selection of the median release  
9 fractions, to utilize those as you're going to call  
10 the representative from the events that you've  
11 analyzed, that's your decision to use the median?

12 MR. GUINN: Correct.

13 DR. SCHULTZ: It's not part of Sandia's  
14 methodology, it's an evaluation and a determination  
15 that NuScale has made to select the representative  
16 source term.

17 MR. GUINN: Well, it's a choice we may  
18 have, I believe it's informed by a choice that Sandia  
19 made. I believe it's -- their method was to select  
20 the median release fractions from their spectrum of  
21 events. And we --

22 DR. SCHULTZ: Oh, okay. Thank you.

23 MR. GUINN: Yes.

24 CO-CHAIR PETTI: So, can we just go back.

25 MR. GUINN: Sure.

1 CO-CHAIR PETTI: I was wondering why the  
2 median releases for the halogens, and it's probably  
3 the one that stands out, why is it lower than the  
4 Sandia results?

5 MR. GUINN: Well --

6 CO-CHAIR PETTI: Give me the simple  
7 answer.

8 MR. GUINN: Yes, I do have some  
9 information right here --

10 CO-CHAIR PETTI: I think because the  
11 alkaline earths are about the same number, tellurium  
12 to the less.

13 MR. GUINN: In general we expect the  
14 fractions to be different because we're a different  
15 design from the --

16 CO-CHAIR PETTI: So it has to do with the  
17 design of, the thermal hydraulics of the accident --

18 MR. GUINN: That's right.

19 CO-CHAIR PETTI: -- that's getting into  
20 the containment. Because these are releases into  
21 containment, correct?

22 MR. GUINN: Yes. Yes, yes.

23 CO-CHAIR PETTI: Yes.

24 MR. GUINN: And I think it's been pointed  
25 out in previous meetings that our core profile is

1 quite a bit different than the typical. I mean, it's  
2 half height PWR fuel but it's a small core.

3 It's tall and narrow and you just don't  
4 see the overheating that you do in a standard core, in  
5 a much lower power density. Several design aspects  
6 would influence this.

7 DR. CORRADINI: I guess we're going to get  
8 to it eventually because the Staff did audit  
9 calculations with their version, and I'm curious about  
10 their direction relative to the SAND report too, to  
11 see if they saw also really smaller amounts. I  
12 thought they did, I just can't remember. I think  
13 we'll talk about that in closed session anyway.

14 MR. GUINN: Right. Yes, I think if you  
15 refer to the transcript of the Chapter 19 closed  
16 session meeting that you had earlier this year, I  
17 think that that would probably give you a lot of  
18 answer you're seeking.

19 DR. CORRADINI: Okay.

20 MR. GUINN: Next. Here is the RADTRAD  
21 model nodalization figure that's shown. This is the  
22 example of our core damage event model.

23 It shows some of the dose receptor  
24 locations for licensing based core damage event  
25 scenario. And you can also see the redundant control

1 room inhabitability systems in this figure.

2 Emergency bottle air spray up in the upper  
3 right and the filtered ventilation air supply in the  
4 center right. C is for compartment, P is for pathway.

5 Hang on. Another unique aspect of the  
6 topical report is our explicit request of approval for  
7 the included aerosol transport and removal  
8 methodology.

9 Specific unique methodology choices  
10 include the selection of thermal hydraulic data from  
11 the surrogate accident scenario with minimum time to  
12 core damage onset. And the assumption of no maximum  
13 limit on iodine to contamination factor for natural  
14 deposition of iodine aerosol.

15 We credit natural deposition aerosol  
16 removal mechanisms including sedimentation,  
17 diffusiophoresis, thermophoresis. And the key output  
18 of our aerosol transport evaluation is removal  
19 coefficients for input to downstream RADTRAD, dose  
20 transport modeling.

21 More detail discussion of these positions  
22 is available in closed session.

23 During the review of the accident source  
24 term topical report and error was identified in the  
25 STARNAUA software. Here is provided a timeline of

1 corrective actions and the extent evaluation.

2           Upon discovery of the error during  
3 evaluation of RAI 9224, an extent evaluation was  
4 immediately initiated. Staff was notified and a  
5 notification letter was sent to the code vendor.

6           Subsequently, the extent evaluation was  
7 completed to identify the issue as limited to a post  
8 processing model recently added to the historically  
9 reliable NAUAHYGROS code by the most recent developer.

10           The following completion of the extent  
11 evaluation, a manual post processing workaround was  
12 developed to provide independent alternative manual  
13 calculation of the required removal rate outputs. As  
14 well as to demonstrate that the sensitivity suite of  
15 co-generated results could also additionally be used  
16 to check for the air.

17           All of our original code benchmarking was  
18 re-performed to use the alternate validation. The  
19 software quality assurance documentation, based on  
20 these benchmarks, was revised and the code was  
21 rededicated through commercial grade dedication.

22           The key takeaway here is that the STARNAUA  
23 application leverages mature and robust NAUAHYGROS  
24 basis for containment aerosol transport analysis.  
25 There are extents limited to an output post-processing



1 model added in the software version of STARNAUA.

2 Principle NAUA-based outputs are  
3 unaffected by the error observation. And if NAUA-base  
4 code subroutines were wrong this would have been  
5 evident in the initial benchmarking.

6 As it were, we re-performed the  
7 benchmarking to implement the error workaround and  
8 confirm good co-behavior again. We do have more  
9 detailed discussion --

10 CHAIR KIRCHNER: Did you take a note, yes,  
11 you're going where I'm going. When we get to the  
12 closed session, can you talk about sensitivity  
13 analysis of STARNAUA and your overall calculations and  
14 that impact and address what you saw in the error of  
15 how it changed the answers for your sample problems.

16 MR. GUINN: Sure, yes. Do we need to  
17 repeat that for those with us in Corvallis?

18 Did we get that there because maybe that's  
19 something to look up and we can refer to? I don't  
20 know that we have a lot of presentation material on  
21 our sensitivity analysis work.

22 We have a lot of presentation material on  
23 the benchmarking, rededication work and some of the  
24 code mechanics discussion. But I know that we do have  
25 information on sensitivity analysis and the spec, and

1 yes, we will be able to get into that. Yes.

2 DR. SCHULTZ: Did the code vendor do some  
3 of this evaluation of extent of condition or was this  
4 --

5 MR. GUINN: No.

6 DR. SCHULTZ: -- within NuScale?

7 MR. GUINN: No. This was a NuScale  
8 effort.

9 DR. SCHULTZ: All told so, but the error  
10 had to do with some post-processing application that  
11 NuScale had developed?

12 MR. GUINN: No, that the code vendor had  
13 developed.

14 DR. SCHULTZ: And so the, but you took it  
15 on to do the full extent evaluation to make sure that  
16 there wasn't anything else that you could identify in  
17 the code?

18 MR. GUINN: That's correct.

19 (Off microphone comments.)

20 MEMBER REMPE: Sorry. I have to practice  
21 with the new mics.

22 Anyway, as you prepare for this upcoming  
23 answer to Walt's question, I was very curious on why  
24 you picked some of the sensitivity parameters that  
25 would seem to be well known so your logic on some of

1       them and then what was most important is of course of  
2       interest.  Thanks.

3               MR. GUINN:  Okay, I'll make a note of  
4       that.

5               CHAIR KIRCHNER:  Just for the record at  
6       this point, the error, you're pointing to the error  
7       was in the post-processing.

8               MR. GUINN:  Correct.

9               CHAIR KIRCHNER:  So not the basic physics  
10      that the model of the aerosol removal or not removal?

11              MR. GUINN:  Yes.  None of these deposition  
12      velocity --

13              CHAIR KIRCHNER:  Okay.

14              MR. GUINN:  -- no.

15              Okay, next up here, topical report also  
16      explicitly requested approval.  They included, post-  
17      accident, temperature dependent, pH calculation  
18      methodology.

19              Methodology evaluates acids and bases,  
20      shows in conservatively based upon the NUREG-5950  
21      guidance for evaluation of post-accident containment,  
22      liquid coolant chemistry.

23              Calculated pH is then shown by NUREG-5950  
24      methods to result in negligible estimated iodine re-  
25      evolution.  This position simplifies the analysis

1 without an impact to conservatism over calculated  
2 results.

3 The negligible impact was confirmed in a  
4 sensitivity analysis provided in response to an  
5 earlier RAI. Which I have the accession number if  
6 you're interested.

7 Another unique aspect if the iodine spike  
8 design basis source term. This was the slide I  
9 referred to earlier on. Where we would unpack it.

10 The iodine spike design basis source term  
11 is a bounding surrogate for all primary coolant loss  
12 design basis accidents to be addressed as the limiting  
13 design basis accident for radiation environment  
14 equipment qualification inside the containment and  
15 area immediately above the containment.

16 Some of the conservative treatments of the  
17 source term include instantaneous homogeneously mixed  
18 release of 100 percent of radionuclide inventory,  
19 radionuclide concentrations at tech spec limits with  
20 an additional iodine spike, and conservative leakage  
21 treatment where its volume dependent for doses  
22 evaluated outside the containment.

23 Activities assume leak at the Reg Guide  
24 1.183 leakage rate. And for doses evaluated inside  
25 containment, no activity leakage is assumed. The

1 source is bottled up for the full qualification  
2 period.

3 Recently added methodology position for  
4 which approval was requested was for the environmental  
5 qualification accident dose calculation methodology.  
6 As mentioned in the previous slide, the iodine spike  
7 design basis source term is applied as the bounding  
8 surrogate source term for all primary coolant loss  
9 DBAs for purposes of equipment qualification within  
10 the containment and the under bioshield region  
11 immediately above the containment.

12 Conservative methodology choices  
13 associated with the environmental qualification dose  
14 methodology are as though previously described for the  
15 iodine spike design basis source term. And more  
16 detailed discussion of conservative environment  
17 qualification methodology treatments are available in  
18 the closed session.

19 Also as discussed in accident source term  
20 topical report, Reg Guide 1.183 guidance on the  
21 assumptions for primary coolant initial activity  
22 concentration based on tech spec limits and  
23 assumptions on modeling of iodine spiking in the  
24 primary coolant are determined to be directly  
25 applicable to the NuScale mainstream line break, steam

1 generator tube failure and small line break  
2 calculation methods.

3 These accidents involve standard  
4 configuration of PWR fuel and light water primary  
5 coolant system.

6 NuScale design specifies minimum reactor  
7 pool depth as greater than 23 foot, therefore Reg  
8 Guide 1.183 guidance on iodine decontamination factor  
9 for fuel handling accidents considered conservatively  
10 applicable to NuScale fuel handling accident,  
11 calculation methodology.

12 Secondary coolant activity concentration  
13 is demonstrated to be an order of magnitude less than  
14 primary coolant activity. And therefore an  
15 insignificant contributor justifiably excluded from  
16 accident source term methodology.

17 Since the offsite dose results to  
18 containment shine, shown to be negligible, that was  
19 shown in response to an earlier RAI. RAI 8706.

20 Finally here listed, the accident  
21 radiological design basis events with calculation  
22 methodologies that are given in the accident source  
23 term topical report. As noted, these industry  
24 standard PWR design basis accident methodologies are  
25 determined to be applicable for the NuScale design,

1 standard regulatory guidance is applied.

2 And note that the rod ejection accident is  
3 included as an approved methodology of the accident  
4 source term licensing topical report, even though  
5 application of the methodology is not realized in  
6 design basis space because it's precluded by  
7 subchannel analysis determination that there is no  
8 fuel failure for this accident.

9 And that concludes this presentation. Any  
10 questions?

11 CHAIR KIRCHNER: Any further questions of  
12 NuScale at this point? From the Members? No.

13 Then thank you very much. We'll not  
14 recess but we'll just take a pause for people on the  
15 phone line while we bring up the Staff for their  
16 presentation.

17 (Off record comments.)

18 MEMBER RICCARDELLA: Hello, the line seems  
19 to have gone dead.

20 CHAIR KIRCHNER: No, we're just --

21 PARTICIPANT: We hear you.

22 CHAIR KIRCHNER: We hear you, we're just  
23 on a pause here while the Staff gets ready to make  
24 their presentation.

25 CO-CHAIR PETTI: This is ACRS; can the

1 people on the phone line hear us? Pete, can you hear  
2 us?

3 MEMBER RICCARDELLA: Yes. You were dead  
4 for a while but now you're back on.

5 CO-CHAIR PETTI: Yes, we muted. Okay,  
6 we're ready to start with the NRC presentations.

7 MR. TESHAYE: Good morning, everyone. My  
8 name is Getachew Tesfaye --

9 MEMBER BLEY: You have to speak into the  
10 microphone.

11 MR. TESHAYE: I thought it was on. Okay.  
12 Good morning everyone, my name is Getachew Tesfaye, I  
13 am the NRC project manager for NuScale's topical  
14 report TR-0915-17565, Revisions 3, accident source  
15 term methodology.

16 Topical report describes the general  
17 methodology for developing accident source terms and  
18 performing cross-building design base accidents and  
19 other required accident radiological consequence  
20 analysis to be referenced by the NuScale small module  
21 reactor design certification application and by other  
22 applications that are referenced in NuScale SMR  
23 design.

24 The NRC Staff has completed an advance  
25 topical report safety evaluation, which was submitted



1 to this Committee on October 18 of this year. The  
2 Staff has also closed all first two open items in  
3 Chapters that are affected by this topical report.  
4 Some of the open items merely track the completion of  
5 TR safety evaluation.

6 We have organized today's presentation in  
7 two parts. The first part of our presentation this  
8 morning focuses on the topical report safety  
9 evaluations report.

10 The second part of our presentation this  
11 afternoon will address the chapters that are impacted  
12 by the accident source term methodology. Those who  
13 are presenting are sitting here with me and the rest  
14 of the technical staff is in the audience.

15 With that, I ask Michelle to get started.

16 MS. HART: Good morning, I'm Michelle  
17 Hart. I was the lead reviewer for the topical report  
18 as far as the accident source term methodology.

19 With me I have Jason White, who is going  
20 to present the atmospheric dispersion aspects of it  
21 and then I will present the rest of the aspects that  
22 are included in the topical report.

23 The topical report is the methodology for  
24 the NuScale SMR design for the accident dose analyses.  
25 And it includes development of the source terms.

1 As they described earlier, they also have  
2 some, a couple of codes that they developed to help  
3 implement their source terms.

4 CO-CHAIR PETTI: Michelle, let me --

5 MS. HART: Yes.

6 CO-CHAIR PETTI: -- the hard copy that the  
7 Members have is not this. This looks like maybe the  
8 one this afternoon.

9 MS. HART: For the afternoon, okay.

10 CO-CHAIR PETTI: Yes. So if someone could  
11 --

12 MS. HART: I will wait.

13 (Off microphone comment.)

14 CO-CHAIR PETTI: This is a thick one.

15 MS. HART: Oh yes, that's the wrong one.

16 CO-CHAIR PETTI: Okay.

17 (Off record comments.)

18 CO-CHAIR PETTI: Okay, here they come.  
19 So, Michelle, you can keep going.

20 MS. HART: Okay. So, as NuScale  
21 described, they did, in the topical report, they  
22 requested Staff approval of 15 specific positions in  
23 their scope of review.

24 Next slide please. So the revision that  
25 we're mostly talking about is topical report Revision

1 3. There are some aspects that are only described in  
2 a RAI response.

3 Equipment qualification, Appendix B is in  
4 an RAI response, it is not in the current version, the  
5 Revision 3, it will be in the next approved version of  
6 the topical report.

7 As you've probably heard before, we,  
8 NuScale proposed to make changes back in January 2018,  
9 and that was to do that, to make that difference with  
10 the environmental qualification of equipment using a  
11 design basis source term. That's when they added the  
12 iodine design basis source term.

13 And then stopped calling the core damage  
14 event the maximum hypothetical accident. They then  
15 started calling it the core damage event. And also  
16 added that iodine spike design basis source term.

17 In response to these proposed changes,  
18 when they proposed these back in January 2018. And  
19 subsequent to that, eventually it resulted in a SECY  
20 paper that we wrote to the Commission about this  
21 issue, about what their plans were for this. With  
22 specific discussion of the environmental qualification  
23 versus the equipment survivability for the core damage  
24 event.

25 The SECY paper does describe the Staff's

1 review approach. And it describes the basis for using  
2 the base term without core damage for environmental  
3 qualification.

4 MEMBER BROWN: Can I just a second? When  
5 you work on that, EQ versus equipment qualification.  
6 I'm trying to work a little bit back when I did  
7 qualification of equipment for the plants I use to  
8 operate within the Navy.

9 EQ included equipment survivability and  
10 they were not separated. In other words, my stuff had  
11 to work regardless.

12 Is there now a separate, are you talking  
13 about a separation or a different EQ or a mode  
14 compartmented environmental qualification?

15 MS. HART: We're talking about the  
16 environmental qualification of systems, structures and  
17 components that are required for the design basis  
18 events. And equipment survivability is used for  
19 severe accident response.

20 MEMBER BROWN: Okay. All right, thank  
21 you.

22 MS. HART: We will talk about it more.  
23 Were there any further questions about that right now?  
24 Okay.

25 So I will turn it over now to Jason to

1 describe the atmospheric dispersion.

2 MR. WHITE: Thank you, Michelle. Good  
3 morning. My name is Jason White and I'm a  
4 meteorologist in the Office of Nuclear Reactor  
5 Regulation.

6 I've been with the NRC for ten years and  
7 have served as the technical reviewer in the Office of  
8 New reactors, as well as the Office of Nuclear Reactor  
9 Regulation. Today I will be discussing a review  
10 portion of the topical report related to atmospheric  
11 dispersion.

12 PAVAN and ARCON96 are NRC computer codes  
13 approved for calculating atmospheric relative  
14 concentration values, which are also known as  
15 atmospheric dispersion factors or chi over q values.

16 PAVAN implements the guidance provided in  
17 Reg Guide 1.145 for determining offsite chi over q  
18 values at the exclusion area boundary, or EAB, and the  
19 outer boundary of the low population zone, or LPZ.

20 ARCON implements the guidance provided in  
21 Reg Guide 1.194 for determining onsite chi over q  
22 values for the control room.

23 Next slide. NuScale's topical report  
24 describes the Applicant's methods for determining --

25 MEMBER REMPE: Excuse me for just a

1 second. Folks out there on the line, please put your  
2 phones on mute. I hear a lot of wrestling papers.  
3 Thank you.

4 MR. WHITE: Okay. NuScale's topical  
5 report describes the Applicants methods for  
6 determining accident chi over q values for the EAB and  
7 LPZ using a methodology that differs from NRC's  
8 guidance.

9 The Staff reviewed the topical report in  
10 accordance with NUREG-0800, which is the standard  
11 review plan, and references section 2.3.4 of the SRP  
12 where it discusses the short-term atmospheric  
13 dispersion estimate for accident releases.

14 NuScale proposes using ARCON96 computer  
15 code methodology for calculating the offsite  
16 atmospheric dispersion values rather than using the  
17 providing computer code. This is listed as Position  
18 2 in Section 1.2 of the topical report.

19 ARCON96 is a general code for assessing  
20 atmospheric relative concentrations and building wakes  
21 under a wide range of situations. The ARCON96  
22 dispersion algorithms are based on field measurements  
23 taken out to distances of 1,200 meters.

24 Typical EAB and LPZ distances for light  
25 water reactors range from 800 to 6,000 meters.

1                   Next slide. NuScale postulates an EAB and  
2                   LPZ at the site boundary, which is estimated to be in  
3                   the range of 880 to 400 meters.

4                   NuScale states that ARCON96 computer code,  
5                   which is developed to model shorter distances, is more  
6                   appropriate for modeling NuScale EAB and LPZ chi over  
7                   q values. Thus, NuScale plans to use this version of  
8                   ARCON 96, NARCON, instead of PAVAN to calculate the  
9                   EAB and LPZ chi over q values.

10                  Next slide. The NRC Staff conducted an  
11                  audit of the topical report and of the NARCON computer  
12                  code. The Staff reviewed the documentation for the  
13                  computer code and executed several independent runs of  
14                  NARCON.

15                  Next slide. After reviewing the topical  
16                  report and conducting the audit, the Staff concluded  
17                  that subject to the conditions and limitations  
18                  specified in Section 6 of the SCR, NuScale methodology  
19                  described in the topical report is acceptable for  
20                  calculating offsite chi over q values for the EAB and  
21                  LPZ in a NuScale design or in a COL application  
22                  referencing NuScale design. Therefore, the Staff  
23                  finds topical report Revision 2 acceptable.

24                  CHAIR KIRCHNER: Jason, let me interrupt  
25                  here and just ask a question. Regardless of the 400

1 meter or 400 foot proposed EAB site boundary and such,  
2 did you put any limitation on use of ARCON96?

3 You mentioned in an earlier slide that it  
4 was developed for distances up to, what, 1,200 meters.

5 MR. WHITE: Yes. There was, based on  
6 measurements up to 1,200 meters.

7 CHAIR KIRCHNER: But --

8 MR. WHITE: Yes, so we -- I'm sorry.

9 CHAIR KIRCHNER: So your position though  
10 is basically that ARCON96 wouldn't be acceptable from  
11 the building to indeterminate distance and not require  
12 use of PAVAN at longer distances?

13 MR. WHITE: Yes. We feel that is  
14 acceptable for use --

15 CHAIR KIRCHNER: Okay.

16 MR. WHITE: -- at the shorter distances.

17 DR. SCHULTZ: I thought there was a  
18 limitation --

19 DR. CORRADINI: Yes.

20 MR. WHITE: Yes.

21 DR. SCHULTZ: -- that was proposed by  
22 NuScale for the NARCON application that was beyond 400  
23 meters?

24 MR. WHITE: Right. Yes, there is a  
25 limitation. I can find the exact number for you but



1       there is a limitation where we feel it would be more  
2       appropriate to use PAVAN for the longer distances.

3               CHAIR KIRCHNER: That was the essence of  
4       my question.

5               PARTICIPANT: Yes. See, I didn't --

6               CHAIR KIRCHNER: Where is that transition  
7       point in your mind? Because if I understand --

8               PARTICIPANT: Whoever is shuffling papers  
9       please stop. Or mute.

10              MR. WHITE: So, yes, to answer your  
11       question there is a distance, and I'm looking for the  
12       number if you'd give me one moment. I think in the  
13       documentation for the ARCON96, there is an  
14       approximation where the developer of the code  
15       approximates that it's possible to use ARCON96 out to  
16       distances.

17              DR. CORRADINI: Yes, I was going to say,  
18       is it in the limitation and conditions? That's what  
19       I was looking for.

20              MR. WHITE: One second.

21              DR. CORRADINI: And then the second  
22       question is, if it's proprietary you can just say it  
23       is and later on in closed session you can tell us  
24       where to look.

25              MR. WHITE: Well, NuScale actually does

1 have it marked as proprietary, so maybe we could  
2 discuss it in the close section.

3 DR. SCHULTZ: That's fine.

4 DR. CORRADINI: Okay, that's fine.

5 MR. WHITE: So I am aware of the number  
6 that you're referencing, but I don't want to  
7 specifically state it since NuScale has it marked  
8 propriety.

9 DR. CORRADINI: No, no.

10 DR. SCHULTZ: It is marked proprietary, we  
11 can discuss it later.

12 MR. WHITE: But there is a number --

13 DR. CORRADINI: Right, that's what I  
14 thought.

15 MR. WHITE: -- where we feel that there is  
16 a limit.

17 DR. CORRADINI: Okay, that's what I  
18 thought. Thank you.

19 MR. WHITE: Okay. Sorry, I was kind of  
20 dancing around it, I didn't want to put the number out  
21 there because it's marked as proprietary.

22 CHAIR KIRCHNER: And then just for the  
23 public record then beyond that distance then you would  
24 require them to use PAVAN?

25 MR. WHITE: Yes.

1 CHAIR KIRCHNER: If the analysis required  
2 it for some COL citing.

3 MR. WHITE: PAVAN is an acceptable method  
4 for the atmospheric dispersion calculations that  
5 belong to distance. If they decide to use an  
6 alternative method, they would have to provide a  
7 justification.

8 CHAIR KIRCHNER: Thank you.

9 MR. WHITE: That concludes my presentation  
10 on the atmospheric dispersion, Michelle will continue.

11 MS. HART: Okay, so next we will describe  
12 the, our review of the accident source terms that do  
13 not include core damage. And these are the following  
14 accidents that, as NuScale described, that are  
15 typically evaluated for PWRs, the fuel handling  
16 accident, the rod ejection accident, main steam line  
17 break outside containment and the steam generator tube  
18 failure and the failure of small lines carrying  
19 primary coolant outside containment.

20 They basically followed Reg Guide 1.183  
21 methodology for all of those events. They did add  
22 that additional iodine spike design basis source term,  
23 which in general terms follows similar guidance for  
24 similar type of accidents that do not have core  
25 damage, that are described in Reg Guide 1.183. And

1 I'll describe some more specifics in the following  
2 slides.

3 Next slide please. So now the iodine  
4 spike design basis source term, as they describe, is  
5 not based on any specific scenario. It's intended to  
6 be the bounding source term for any release of coolant  
7 into the containment without core damage.

8 It's an instantaneous release. It does  
9 have two cases for the iodine spiking, what is in Reg  
10 Guide 1.183.

11 It uses the Reg Guide 1.183 assumptions on  
12 the iodine chemical forms, which since there are no,  
13 there's no assumptions on removal of iodine within the  
14 containment for this event, those iodine chemical  
15 forms are kind of arbitrary. They don't really make  
16 a difference in the final dose result.

17 It is assumed to be homogeneously mixed  
18 throughout the free volume. And it assumes the design  
19 basis containment leak rate for the first 24 hours and  
20 half that rate for the remainder of the accident.  
21 Similar that you would find for large light water  
22 reactors, the LOCA dose analysis.

23 This iodine spike design basis source term  
24 is the basis for the equipment qualification radiation  
25 source inside containment and under the bioshield.

1 And it, the Staff did find it acceptable because it  
2 was consistent with similar accidents as described in  
3 Reg Guide 1.183.

4 DR. CORRADINI: Just so I am not confused,  
5 but it is also used as a bounding comparison for their  
6 design basis events also?

7 MS. HART: Correct, they do. It's used  
8 both for the equipment qualification radiation source  
9 and it's also evaluated for offsite dose, control room  
10 dose, and technical support center dose.

11 DR. CORRADINI: For design basis. That's  
12 where I am -- I want to make sure I am not  
13 misunderstanding.

14 MS. HART: So it's in that set of Chapter  
15 15 dose analyses. It is considered to be a design  
16 basis event for them for evaluation of equipment  
17 qualification and they say it's not a specific event  
18 so it's not categorized in Chapter 15 as an accident,  
19 but it is a design basis event.

20 DR. CORRADINI: Okay. Thank you.

21 MS. HART: So to go to the primary coolant  
22 iodine spiking, this is the way that they describe it,  
23 and there is a typo on this slide. I will tell you  
24 when we get to it.

25 The coincident iodine spike and a pre-

1 incident iodine spike they do both and as you had  
2 described earlier the coincident iodine spike assumes  
3 that you start at the design basis limit equilibrium  
4 activity concentration for the iodine and noble gases  
5 and that there is an eight hour coincident iodine  
6 spike at a certain amount of times the equilibrium  
7 appearance rate that would give you that equilibrium  
8 activity concentration and the pre-incident iodine  
9 spike is the design basis limit maximum allowed  
10 activity concentration.

11 And so the typo is under the table there.  
12 The main steam line break does have the coincident  
13 iodine spiking factor of 500. The iodine spike design  
14 basis source term also has that factor.

15 They both have coincident spike and pre-  
16 incident spiking. The steam generator tube failure  
17 has a coincident spike with spiking factor of 335,  
18 which is the same as in Reg Guide 1.183. That's where  
19 one of the typos is.

20 DR. CORRADINI: She is going too fast for  
21 me.

22 MS. HART: It also has a pre-incident  
23 spike at the maximum allowed activity concentration.  
24 The small line break has a coincident spike with a  
25 spiking factor of 500, that's also a typo, and there

1 is no pre-incident spike.

2 DR. CORRADINI: So you did a dutiful job  
3 of going through all this, but you lost me somewhere  
4 in there.

5 MS. HART: This is all the same as in Reg  
6 Guide 1.183.

7 DR. CORRADINI: Okay. So you followed Reg  
8 Guide 1.183?

9 MS. HART: Correct.

10 DR. CORRADINI: Okay.

11 CO-CHAIR PETTI: Mike, just the second 500  
12 and 335 are flipped.

13 MS. HART: Yes. And I only noticed it  
14 last night when I was practicing. I had looked at  
15 this like ten million times before and, of course, you  
16 elide over things when you see them.

17 So, yes, these are all consistent with  
18 guidance on these same type of events in Reg Guide  
19 1.183.

20 DR. CORRADINI: So these are considered  
21 bounding or are they based on experimental data with  
22 some sort of bounding value. I am trying to  
23 understand the history of this.

24 MS. HART: By "this" do you mean the  
25 spiking factor?

1 DR. CORRADINI: Yes.

2 MS. HART: So the spiking factor is kind  
3 of semi-empirically based. It was based off of some  
4 Westinghouse information and it was based off of --  
5 Actually, it was mostly based off licensee event  
6 reports that would say what the coolant activity  
7 concentration was after like a normal shutdown or a  
8 trip or, you know, that they saw elevated activity in  
9 the coolant that was --

10 DR. CORRADINI: Because there was water  
11 that essentially brought iodine out into the coolant?

12 MS. HART: So there is not a specific  
13 mechanism identified. Like I said it's semi-  
14 empirically based. They saw this increase in the  
15 activity concentration. They came up with a model  
16 that would seem to describe it.

17 This was something that was evaluated  
18 several times over the past. There was a proposed --  
19 A couple of times there was a proposed general safety  
20 issue and it wasn't followed up on because there  
21 wasn't a risk, you know, the cost wasn't, you know,  
22 justified for the, you know -- Cost benefit analysis,  
23 that's what I was trying to think of.

24 So it's not a mechanistic model and it's  
25 not a precise model. Is that a good way to talk about



1 it?

2 CHAIR KIRCHNER: We were -- Yes, looking  
3 at the precision of 335 for what's an empirical number  
4 --

5 MS. HART: Yes. This was --

6 CHAIR KIRCHNER: -- raises questions, but  
7 we'll let it go.

8 MS. HART: Right. It was back in 2000  
9 when we developed Reg Guide 1.183. We took the same  
10 information that came up with 500 and we said, well,  
11 for this where you don't have the large hole that you  
12 do with the main steam line break maybe you could  
13 interpret the results in a different way and allow a  
14 less conservative assumption on the spiking factor for  
15 the steam generator tube rupture or steam generator  
16 tube failure.

17 DR. CORRADINI: Thank you.

18 MS. HART: Okay. Next slide, please. So  
19 in the topical report they did neglect the small  
20 secondary side volume. This was one of their  
21 positions, Position 13, that could contain activity  
22 from the primary to secondary leakage.

23 They described that for the NuScale SMR  
24 design the ratio of the secondary coolant volume to  
25 the primary is approximately 1 percent of the primary

1 and the initial secondary activity concentration on  
2 top of that is already in order of magnitude less than  
3 the primary.

4 And as NuScale had described there was  
5 some sensitivity analysis done and it does not impact  
6 the final dose results. It doesn't add significant  
7 dose, so, therefore, the staff found that neglecting  
8 that small secondary side volume was acceptable.

9 Next slide, please. One of their  
10 positions, also Position 12, was that the fuel  
11 handling accident would assume the same iodine  
12 decontamination factor of 200 for release from the  
13 reactor pool.

14 The NuScale fuel is similar to the fuel  
15 that was covered by Reg Guide 1.183 and the water  
16 depth above the damaged fuel is greater than 23 feet,  
17 which was used as the basis for the decontamination  
18 factor in Reg Guide 1.183. Therefore, we found their  
19 assumption to be acceptable and the position  
20 acceptable.

21 Next slide, please. Position 15 in their  
22 topical report described that the dose analysis assume  
23 that containment shine is negligible, and this is for  
24 the core damage event, and for any event the core  
25 damage event would be bounding for any event.

1           The staff audited some proprietary  
2           calculations that NuScale had done to show that the  
3           shine contribution from the core damage source term in  
4           containment was, indeed, negligible including at the  
5           control room and offsite.

6           And so, therefore, we found the topical  
7           report position to be acceptable that containment  
8           shine is negligible.

9           DR. SCHULTZ: Michelle, was there any  
10          reason for the staff to do a confirmatory calculation  
11          of that?

12          MS. HART: We did not do a confirmatory  
13          calculation of the shine analysis. It was pretty  
14          clear from evaluating their dose calculation it was  
15          very well documented.

16          DR. SCHULTZ: Good. Thank you.

17          MS. HART: So for the environmental  
18          qualification source the Reg Guides 1.89 and 1.183  
19          indicate a core damage source term should be  
20          considered for environmental qualification and in the  
21          past we have used that LOCA, the design basis LOCA  
22          released to the containment as a source term and it  
23          was considered to be bounding for the equipment  
24          qualification, or environmental qualification of  
25          equipment.

1           However, for the NuScale design since  
2           their design basis events do not result in core damage  
3           the staff determined that the core damage need not be  
4           considered or assumed for environmental qualification,  
5           which was also described in some detail in SECY 19-  
6           0079.

7           And this was mostly based, or it was  
8           partially based on the fact that the regulation  
9           describing the environmental qualification requirement  
10          states that the post-accident radiation environment  
11          must be based on the most severe design basis accident  
12          whereas the description of that fission product source  
13          term is not included or is not a footnote to that  
14          particular regulation like it is to the offsite dose  
15          analysis or to some of the TMI-related requirements  
16          which specifies the accident source term that you  
17          should use, that substantial release to the  
18          containment footnote.

19                 DR. CORRADINI: Can you say that again  
20                 slower?

21                 MS. HART:         So the environmental  
22                 qualification regulation, 10 CFR 50.49(e)(4) --

23                 DR. CORRADINI: Okay.

24                 MS. HART: -- says that the post-accident  
25                 radiation environment should be based on the most

1 severe design basis accident but it does not tell you  
2 what that source term would be.

3 DR. CORRADINI: Indifference to offsite?

4 MS. HART: Correct.

5 DR. CORRADINI: Okay.

6 MS. HART: The offsite criterion does have  
7 that footnote that tells you the substantial release  
8 into the containment with substantial fission  
9 products, yes.

10 MEMBER BLEY: Yes, earlier I had said this  
11 argument between the, that the sort of legal argument  
12 that was going on ought not concern us.

13 What you just said though could equally  
14 apply to any LWR because none of the Chapter 15 design  
15 basis accidents lead to core damage, contrary to what  
16 was mentioned, for any of them.

17 MS. HART: So this has been an interesting  
18 discussion that we have been having about what is  
19 design basis, what is a design basis accident, is the  
20 LOCA a design basis accident.

21 I can tell you as an dose analyst and if  
22 you look at Reg Guide 1.183 there has certainly been  
23 an assumption in the past that, yes, that's a design  
24 basis accident for the purposes of citing and safety  
25 analysis to look at that.

1           It's not in the regulation that it is  
2           though, but we have been considering that -- If you  
3           look at the LOCA there may be several different design  
4           basis events that you look at design basis accidents  
5           the way you evaluate them.

6           So you have the one where you evaluate the  
7           operability of the ECCS to ensure that the core does  
8           not get damaged. Then you have an evaluation of the  
9           containment capability, you know, the containment  
10          thermal hydraulics basically, and then there is an  
11          evaluation of assuming a large release into the  
12          containment, the capability of the containment, and  
13          the mitigating systems to maintain the doses offsite  
14          and in the control room at appropriate values as far  
15          as design.

16          So it's an evaluation to evaluate the  
17          design, is it a design basis accident and what is the  
18          design basis accident. I think most of the dose  
19          analysts in the past have thought that the LOCA was a  
20          design basis accident, that it's the maximum  
21          hypothetical accident as a proxy for a design basis.

22          But we did have a lot of discussions and  
23          the SECY goes into some detail about that and a lot of  
24          it was keying off of the fact that for the offsite  
25          dose analysis this fission product release was kind

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1 of, although that footnote it has some mushy words in  
2 it, was defined whereas for the environmental  
3 qualification of equipment it was really only  
4 describing that you need to look at the most severe  
5 design basis accident.

6 I know that that was a lot of words that  
7 maybe -- This has been a very confusing time for me I  
8 can tell you that. They are all in Chapter 15.

9 DR. CORRADINI: Go ahead.

10 MS. HART: So if you say a design basis is  
11 Chapter 15 then it's, you know, it's very confusing.  
12 But that was what they determined that they were going  
13 to do for the NuScale environmental qualification  
14 sources that they were going to use, the reactor  
15 coolant design basis iodine spiking --

16 (Simultaneous speaking.)

17 DR. CORRADINI: I guess I had a similar  
18 impression that Dennis did, but I want to make sure.  
19 To be honest, I didn't understand your answer to him.  
20 I was a bit confused.

21 MS. HART: I think that in the past most  
22 of your design basis accident did use that LOCA source  
23 term so we thought that the LOCA source term is a  
24 source term for a design basis accident.

25 It may not be the same LOCA design basis

1 accident that you think of classically for evaluating  
2 ECCS operation.

3 DR. CORRADINI: Okay.

4 MEMBER BROWN: Were previous applicants --  
5 Is NuScale being asked to operate to a lower standard  
6 than the other ones, the previous applicants? Because  
7 you're saying your understanding is in the past that's  
8 been a circumstance, now they want something different  
9 that's an easier standard.

10 (Off microphone comment.)

11 MEMBER BROWN: That's what I got out of  
12 your large volume of words. I didn't have a problem  
13 with them, it just --

14 MS. HART: They have proposed something  
15 different, I can say that.

16 MEMBER BROWN: It sounds like it's easier  
17 though. I mean it's a reduction in the standard that  
18 you have to meet as opposed to prior applicants. I am  
19 just -- It seems to me it's a dividing line.

20 They are asking to not have to meet the  
21 same standard --

22 PARTICIPANT: Right.

23 MEMBER BROWN: -- that people have a more  
24 rigorous standard, they didn't have to meet that.  
25 That's what they are asking for. That's the way I'm



1       --

2                   (Simultaneous speaking.)

3               MS. HART: I don't know that it was a more  
4 rigorous standard in the past. I think it was just  
5 they used that source term that already existed. They  
6 didn't try to determine what the bounding design basis  
7 source term would be outside of the thing that were  
8 already doing for the offsite dose analysis.

9                   (Simultaneous speaking.)

10              MEMBER REMPE: If I literally take your  
11 words, you said we looked at the release from the  
12 vessel to evaluate the ability of the equipment in the  
13 containment to function appropriately.

14              Now what I think I heard you say is we  
15 don't have to do such a big source term to evaluate  
16 the functionality of the equipment in the containment,  
17 and so I am kind of quoting you back a little more  
18 detail of why Charlie is asking that question.

19              So are we not -- Even though maybe the law  
20 didn't require it are we, because they caught onto the  
21 details of the law they don't have to do as much is  
22 kind of what Charlie I think is saying and I got that  
23 intent from your question and is that true?

24              MS. HART: I will let Ed Stutzcage answer  
25 some of that.

1 (Off microphone comment.)

2 MS. HART: It's on the thing.

3 MR. STUTZCAGE: There we go. No? I think  
4 I got it.

5 MS. HART: Yes.

6 MR. STUTZCAGE: Okay. Yes, this is Ed  
7 Stutzcage. I was involved in the EQ review team with  
8 the --

9 CHAIR KIRCHNER: You need -- Yes, get  
10 closer and also --

11 MS. HART: It's on the stem.

12 MR. STUTZCAGE: Hello?

13 PARTICIPANT: It's on.

14 MR. STUTZCAGE: It says it's -- I just got  
15 to get real close.

16 (Off microphone comment.)

17 MR. STUTZCAGE: Okay, got you.

18 MS. HART: Okay.

19 MR. STUTZCAGE: Yes, so now I lost my  
20 train of thought.

21 CHAIR KIRCHNER: Well you can start with  
22 your name so that we get that on the record.

23 MR. STUTZCAGE: Yes, Ed Stutzcage. Ed  
24 Stutzcage, I was part of the environmental  
25 qualification review team for the radiation dose.

1 I think based on some of the argument  
2 NuScale made in their, since, you know, it is true  
3 that you could say that for large LWRs that they, too,  
4 you know, their design basis events wouldn't result in  
5 core damage.

6 We took a look at the way the regulations  
7 are worded and that we thought based on the way the  
8 regulations are worded and NuScale's position,  
9 NuScale's, you know, description of their accidents  
10 and how they were treating them we thought that it was  
11 in accordance with the requirements to use the, you  
12 know, the non-core damage source term for EQ and the  
13 core damage source term for the survivability. That's  
14 about --

15 MEMBER REMPE: So if I am a smart owner of  
16 an operating LWR and I say, well, gee, I didn't  
17 realize the regulation didn't require this can I  
18 backtrack?

19 DR. SCHULTZ: No, this is different.

20 DR. CORRADINI: No, because the LOCA would  
21 have failed fuel. This does not have failed fuel.

22 DR. SCHULTZ: There is no --

23 (Off microphone comment.)

24 DR. SCHULTZ: There is no large break LOCA  
25 for the NuScale design.

1 MEMBER REMPE: Yes, I get that difference,  
2 but --

3 DR. SCHULTZ: There isn't large scale --  
4 There is a large break LOCA for the light-water  
5 reactor designs.

6 MEMBER REMPE: If I can show because of a  
7 risk-based argument like someone has that it is very,  
8 very unlikely to have a large break LOCA might I come  
9 back and say, hey, I'd like an exemption, too?

10 DR. SCHULTZ: Well, NuScale has not done  
11 a risk evaluation here to demonstrate they don't have  
12 a large break LOCA. They don't have large pipes.

13 MEMBER REMPE: Yes, but some maybe large  
14 PWRs have -- Large PWRs might try and do that is what  
15 I am going.

16 DR. CORRADINI: They might, but I don't  
17 think they'd survive because this is deterministically  
18 the design doesn't have a large break.

19 DR. SCHULTZ: Exactly.

20 MEMBER REMPE: I get it with NuScale, but  
21 I what I am worrying about is because we now have a  
22 different interpretation of the details of the law  
23 that somebody else could come in and say, well, it's  
24 a very low frequency, it's not within the design  
25 basis. I'm just wondering if we are opening up

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1 something here.

2 DR. SCHULTZ: Well the large break --

3 DR. CORRADINI: Well the report -- I'm  
4 sorry.

5 DR. SCHULTZ: Go ahead.

6 DR. CORRADINI: No, you go ahead.

7 DR. SCHULTZ: The large break LOCA is part  
8 of the design basis accident for large light-water  
9 reactors. So a large light-water reactor has to abide  
10 by 10 CFR 50.49(e)(4) and evaluate the dose for  
11 equipment qualification for a large break LOCA.

12 MEMBER REMPE: And what if they show it's  
13 not within the design basis because the frequency of  
14 a large break --

15 DR. CORRADINI: Well --

16 DR. SCHULTZ: But that would be 10 --

17 MEMBER REMPE: -- is less than --

18 (Simultaneous speaking.)

19 DR. CORRADINI: That would be 10 50.46(a),  
20 which was transition break size in 2006 and it went  
21 nowhere.

22 MEMBER REMPE: Okay. Just curious because  
23 there has been -- Some other folks have tried it.

24 DR. CORRADINI: But if it went somewhere,  
25 and I would say there is no such thing, a large break

1 LOCA, because those pipes just can't break and I get  
2 down to a smaller size then I might go further and do  
3 this, but as far as I can --

4 MEMBER REMPE: That's where I am thinking.

5 DR. CORRADINI: -- understood the  
6 transition break size argument that was put forward in  
7 2006 never went anywhere.

8 MEMBER REMPE: But I know of another case  
9 where someone has tried to do that with a large  
10 operating PWR. I just am curious, but it's beyond  
11 what we're talking about today. Maybe.

12 MS. HART: Okay. So next slide, please.  
13 So the iodine spike source for the equipment  
14 qualification is provided for the equipment  
15 qualification inside the containment and under the  
16 bioshield.

17 The source term uses the Reg Guide 1.183  
18 assumptions on iodine spiking. However, it does not  
19 consider spiking of radionuclides other than iodine.

20 The staff does have some information that  
21 indicates that other radionuclides may spike after  
22 reactor accidents or during reactor accidents, for  
23 example the cesiums and noble gases may spike, may  
24 increase, and these other radionuclides could be  
25 significant to the environmental qualification dose of

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1 a equipment whereas it may not be as significant for  
2 a human dose.

3 The applicant included numerous  
4 conservatisms in the environmental qualification dose  
5 methodology for inside the containment, both the  
6 liquid and the vapor regions were considered and under  
7 the bioshield, but it was unclear that these  
8 conservatisms bound the potential spiking of other  
9 radionuclides.

10 Next slide, please. So the staff's  
11 resolution of our concerns about only considering  
12 spiking of iodine and not spiking these other  
13 radionuclides due to the small containment size in the  
14 NuScale design the dose rates inside the containment  
15 and under the bioshield are high during normal  
16 operation.

17 Some of that is due to the high neutron  
18 field, some of it is due to just the fact that you are  
19 close to the core during that, during operation, you  
20 know, the equipment is close to the core.

21 The staff performed conservative  
22 independent analyses with spiked source terms,  
23 including some of the other radionuclides, and based  
24 on the staff's independent review and analysis staff  
25 found that normal operation dose provided sufficient

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1 margin over the accident doses, such the potential  
2 spiking of other radionuclides would not be a  
3 significant contributor to the total integrated dose.

4 And the staff, therefore, approves the use  
5 of the EQ dose methodology for calculating doses  
6 inside the containment and under the bioshield and  
7 direct radiation sign from associated sources for the  
8 NuScale design.

9 Next slide, please. So in the FSAR the EQ  
10 source term and total integrated dose values are more  
11 conservative than the values that would be obtained  
12 using the topical report methodology as written.

13 Since the use of these values would result  
14 in equipment being qualified to a higher value than  
15 indicated in the proved methodology the staff also  
16 found the FSAR source term and the total integrated  
17 doses to be acceptable, and we'll describe a little  
18 bit more of that this afternoon when we talk about the  
19 related areas.

20 In contrast, we'll talk a little bit here  
21 about equipment survivability, just a little bit of an  
22 overview. While core damage was not assessed for the  
23 environmental qualification equipment, certain  
24 equipment associated with containment integrity and  
25 combustible gas monitoring is designed to function to

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1 withstand core damage events in accordance with these  
2 regulations and SECY 93-087 and SECY 90-016.

3 The equipment survivability will be  
4 discussed in more detail in the afternoon session.  
5 Next slide, please. So now we get to the core damage  
6 event.

7 So this is the footnote that I had  
8 referred to earlier about the fission product release  
9 and this is in the, for the regulation for the design  
10 certification safety analysis.

11 The TMI requirements for shielding for  
12 vital area access post-accident sampling we could  
13 control outside containment and control room  
14 habitability problem pathways, also referred to a  
15 similar source term footnote with the major accident  
16 hypothesized for purposes of site analysis or  
17 postulated for considerations of possible accidental  
18 events.

19 Next slide, please. So Topical Position  
20 1 NuScale had proposed that the treatment of the core  
21 damage event be designated as an appropriate beyond  
22 design basis event for the NuScale design.

23 The reasoning they were saying was this  
24 core damage event it's not a specific accident  
25 scenario. The source term is derived from a range of

1 accident scenarios that result in significant damage  
2 to the core to meet the footnote, however, the core  
3 damage is beyond the design basis for their facility.

4 As noted in our SER the staff makes no  
5 finding on Position 1. It is not really germane to  
6 our decision as to whether they have met the specific  
7 regulations.

8 The regulations don't say if it needs to  
9 be design basis or beyond design basis. Therefore, we  
10 did not make a finding on that position. We didn't  
11 accept it or not accept it. We just didn't think it  
12 was material to the decision. Next.

13 DR. CORRADINI: Is that unique to NuScale  
14 or would that be a position held by other folks who  
15 would come in?

16 MS. HART: So nobody else has proposed to  
17 take that position so we have not -- We're not taking  
18 a position on their position in the first place, so  
19 not necessarily.

20 DR. CORRADINI: But let me make sure --  
21 This confused me. I have to admit I read what they  
22 asked for and I read your response and I don't get it.  
23 So it seems like we are two ships passing in the  
24 night.

25 So I don't understand their logic for

1 asking about it and what I am hearing you say on your  
2 side is is that staff says this is their identified  
3 MHA, we're okay with the MHA per their -- Well, you're  
4 going to talk about your analysis --

5 MS. HART: Yes.

6 DR. CORRADINI: -- but upon review and  
7 reflection you've looked upon it and it's acceptable.  
8 Whether it's severe or not is immaterial?

9 MS. HART: That's correct.

10 DR. CORRADINI: Okay.

11 DR. SCHULTZ: But let me ask it a  
12 different way, and that is the fact that you haven't  
13 made a determination with regard to Position 1 does  
14 that affect any of the open item issues that we are  
15 going to be discussing this afternoon?

16 MS. HART: No, it does not.

17 DR. SCHULTZ: Okay. Thank you.

18 MS. HART: Okay. So to go to the core  
19 damage event, they developed a core damage source term  
20 which is representative of core damage events. It's  
21 also not a specific scenario.

22 This is similar to what we had done when  
23 we developed the alternative source term in NUREG-1465  
24 and then was adopted into Reg Guide 1.183. It's  
25 derived from a spectrum of surrogate accident

1 scenarios that involved substantial damage to the core  
2 and it is informed by the NuScale PRA.

3 It is single module events at full power  
4 and assumes intact containment and it's intended to be  
5 representative or bounding of a dominant majority of  
6 intact containment core damage events for their  
7 NuScale nuclear power module.

8 The consideration of a range of core  
9 damage scenarios is consistent with the source terms  
10 that we developed in NUREG-1465 and Reg Guide 1.183  
11 for large light-water reactors and we did find it  
12 acceptable because this is consistent with Footnote 3  
13 about the fission product release.

14 Next slide, please. So the core damage  
15 released to the containment was based on NuScale's  
16 specific analyses that used the MELCOR code, or that  
17 will be performed by the applicant that references the  
18 topical report. Of course, NuScale did also do this  
19 for the design certification application.

20 The core damage source term is  
21 characterized by the magnitude, timing, chemical  
22 grouping, and iodine chemical forms of the release to  
23 the containment.

24 Topical Report Position 5 is about the  
25 release magnitude and this is the fraction of core

1 inventory released to the containment. It is grouped  
2 by a chemical group so each of the nuclides is grouped  
3 together in similar chemical groupings.

4 They use the median release values over  
5 all scenarios for each grouping and it is not  
6 consistent with NUREG-1465 which use mean values with  
7 some adjustments.

8 And the staff evaluated the sensitivity of  
9 the release magnitude in an integrated fashion with  
10 some other questions that we had with about their  
11 source term using an independent confirmatory analysis  
12 which I will describe more later.

13 CO-CHAIR PETTI: So just a question on  
14 mean versus median. As I recall in the Sandia report  
15 that the applicant cites they go into some discussion  
16 about mean being, median being more conservative,  
17 although they prefer that --

18 (Simultaneous speaking.)

19 MS. HART: So, yes, I did a little  
20 Googling during the discussion that you had earlier  
21 and, yes, there is a discussion about median and, yes,  
22 that Sandia report that they refer to that is a more  
23 recent version of accident source term for large  
24 light-water reactors for high burn-up and for MOX did  
25 use median values and they describe why they chose

1       that over the mean.

2                   CO-CHAIR PETTI:   Okay.

3                   MEMBER BLEY:   Are you going to talk about  
4       that in any more detail later?

5                   MS. HART:   No.

6                   MEMBER BLEY:   What Dave just said makes no  
7       sense to me.   There is no distribution for which the  
8       median is higher than the mean.

9                   CO-CHAIR PETTI:   Yes, I got it backwards.  
10       Okay.

11                   MEMBER BLEY:   Okay.  As long as you got it  
12       backwards.

13                   CO-CHAIR PETTI:   Yes.

14                   MEMBER BLEY:   Are they fairly close is  
15       that why you are comfortable with them using the  
16       median?

17                   MS. HART:   So we --

18                   MEMBER BLEY:   Because they can be very far  
19       apart.

20                   MS. HART:   They could be very far apart.  
21       We did look at the specific scenarios and the specific  
22       results they gave us.  You know, they gave the example  
23       calculations for the topical report.

24                   We did look at the actual values that they  
25       used in their FSAR and we also, like I said, we did

1 some confirmatory analyses to determine, you know,  
2 based on our own assessment whether that would make a  
3 big difference in the final dose result and we did not  
4 see that it would.

5 MEMBER BLEY: Interesting, okay.

6 MS. HART: Yes.

7 MEMBER BLEY: But you don't intend to talk  
8 about that anymore?

9 MS. HART: We were not -- We do not have  
10 any presentation materials on that.

11 MEMBER REMPE: But you'll be open to  
12 questions on that topic in the closed section?

13 MS. HART: Correct.

14 MEMBER BLEY: I forget, you didn't raise  
15 this as an open item in the SER that I recall.

16 MS. HART: So in the SER all of this was  
17 an open item because it is --

18 MEMBER BLEY: The whole conglomeration.

19 MS. HART: We were waiting on the next  
20 revision of their topical report to see what they were  
21 going to change.

22 That particular aspect did not change.  
23 However, because of the timing of the SER I was not  
24 sure what they were going to change from Revision 2 to  
25 Revision 3 of their topical report, so the SER in

1 Phase 2 was just an open item for all of these topics.

2 MEMBER BLEY: Because of the potential for  
3 the mean to be very different from the median and if  
4 it's a broad uncertainty distribution it will be, it  
5 seems to me in resolving the open item in writing you  
6 ought to give some real consideration to explaining  
7 why using the median is acceptable to you, to the  
8 staff.

9 MS. HART: Well we did describe our  
10 evaluation of that and it was included in the  
11 independent analyses that that did not create a big  
12 problem. There is only five scenarios that they are  
13 talking about here.

14 MEMBER BLEY: Mm-hmm.

15 DR. CORRADINI: So we're talking about  
16 your guy's audit calculations which is a separate  
17 document?

18 MS. HART: Correct. There is a reference  
19 document.

20 (Simultaneous speaking.)

21 DR. CORRADINI: And the comparison --  
22 Okay. Okay.

23 DR. SCHULTZ: I guess it's probably  
24 included in that document, but it's -- So it's not,  
25 it's kind of apples and oranges because the audit

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1 calculation was two scenarios.

2 MS. HART: Correct.

3 DR. SCHULTZ: The median evaluation that  
4 NuScale is doing is to take five scenarios and take  
5 the median values from those five scenarios.

6 MS. HART: Correct.

7 DR. SCHULTZ: So I'm trying to --

8 MS. HART: And so some of the differences  
9 that we had were that we had a different amount of  
10 core damage basically. We had a much higher amount of  
11 noble gases especially and iodines that were released  
12 in our assumptions and those findings were different.

13 DR. SCHULTZ: In those scenarios that you  
14 examined?

15 MS. HART: Right. And so we looked at  
16 those specific scenarios and we compared them to what  
17 they chose as their representative scenario and found  
18 that it would not make a major difference to the dose  
19 results.

20 DR. SCHULTZ: Right. But what I am  
21 getting at is that the audit calculations didn't  
22 address this median versus mean issue.

23 MS. HART: Not directly but they were --  
24 We used the results of them to inform whether we  
25 thought it was important to make that differentiation

1 between the median and the mean and we did not.

2 DR. SCHULTZ: Understood in terms of  
3 looking at the magnitude of the differences --

4 MS. HART: Correct.

5 DR. SCHULTZ: -- of one approach versus  
6 the other plus the magnitude of the difference if you  
7 did mean evaluation versus median evaluation.

8 MS. HART: Correct. We did not do that.  
9 We did not take the mean versus the median of their  
10 results and do that assessment, we did not.

11 DR. SCHULTZ: Understood. Thank you.

12 MS. HART: Okay. So for the release  
13 timing to the containment, which is their topical  
14 report Position 4, they used the minimum release onset  
15 of any scenario evaluated and use the associated  
16 release duration from the same scenario, which was  
17 also the shortest duration it turned out to be.

18 And we found this to be consistent with  
19 the development of the source term in NUREG-1465 and  
20 Reg Guide 1.183.

21 CO-CHAIR PETTI: So, Michelle, the  
22 question I had was so does that mean that they just  
23 used Case 2 thermal hydraulics to drive it?

24 MS. HART: For the timing. For the timing  
25 they used Case 2.

1 CO-CHAIR PETTI: Well, right, and the  
2 rates and the amount of steam going in, I mean, all of  
3 the thermohydraulic stuff?

4 MS. HART: So the thermohydraulics for  
5 STARNAUA was evaluated. It was looked at differently,  
6 but it is the same scenario. Yes.

7 Topical report, Position 6, they used the  
8 chemical grouping from that 2011 Sandia report. It's  
9 generally consistent with the state of the art in  
10 severe accident modeling. It just regroups some of  
11 the isotopes. It did not leave any isotopes out or  
12 add any new isotopes. And so therefore, we found the  
13 topical report, Position 6, about the chemical  
14 grouping, to be acceptable.

15 The iodine chemical form fractions, they  
16 used the same fractions as in NUREG-1465 and Reg Guide  
17 1.183, 95 percent elemental and the rest gaseous.  
18 PHEBUS tests do demonstrate that this assumption is  
19 conservative for light-water reactors, so we found no  
20 problem with that.

21 And then the staff found the methodology  
22 to characterize the core damage source term release to  
23 the containment, you know, the magnitude mix,  
24 chemical grouping, and IDA and form fractions to be  
25 acceptable. They're similar to the development of

1 water reactor source terms that we did in NUREG-1465  
2 and put in the accident source term or the alternative  
3 source term methodology regulatory guide, Reg Guide  
4 1.183.

5 And the staff, like I said, we did do an  
6 integrated evaluation of the core damage evaluation,  
7 or core damage event methodology in a staff  
8 confirmatory analysis to look at some of the  
9 assumptions.

10 Next slide, please. So to go to IDA and  
11 retention and containment, they did provide a  
12 temperature dependent pH calculation method in their  
13 methodology. These methods are consistent with the  
14 information in NUREG CR-5950 and guidance in Reg Guide  
15 1.183, and the Standard Review Plan 6.5.2. The  
16 NuSCALE code that they developed implements that  
17 topical report section which is consistent with the  
18 information in the NUREG.

19 Their Position 14 in their topical report  
20 states that IDA and re-evolution is negligible for the  
21 pH values between six and seven. Reg Guide 1.183,  
22 based on information in NUREG-1465, says that IDA and  
23 re-evolution need not be considered for in-containment  
24 water pools with a pH of seven or greater. And that  
25 was also based on NUREG CR-5950.

1           NuSCALE estimated less than one percent of  
2           the aqueous iodine is converted to elemental iodine  
3           for a pH temperature, dependent pH value of 6.0 using  
4           Figure 3.1 from NUREG CR-5950.

5           The staff also looked at that graph and  
6           determined that that was a reasonable determination  
7           using that figure and found that the methods were  
8           consistent with the guidance and therefore acceptable  
9           to the staff. Therefore, we find the position to be  
10          acceptable. The iodine re-evolution is negligible for  
11          a pH above six.

12          Next slide please.

13          CHAIR KIRCHNER: There, Michelle, you're  
14          not reaffirming the topical report, you're just  
15          reaffirming what's an NRC position about re-evolution.

16          MS. HART: Correct.

17          CHAIR KIRCHNER: Because the test will  
18          come later when you actually do a design specific  
19          analysis and then look at what mix is --

20          MS. HART: What the actual pH is.

21          CHAIR KIRCHNER: -- the actual mix of,  
22          yes, and pH is.

23          MS. HART: Yes.

24          CHAIR KIRCHNER: So really, in a sense,  
25          you're just saying that, yes, they're consistent with

1 previous guidance. And so methodology-wise, it's  
2 acceptable. But the proof comes in Chapter 15 or  
3 elsewhere when you apply it.

4 MS. HART: That's fair to say, yes.

5 CHAIR KIRCHNER: Yes.

6 MS. HART: So the air flow removal and  
7 containment, they did model natural deposition  
8 phenomena. They used the STARNAUA code which includes  
9 the model for natural deposition and containment,  
10 gravitational settling, and some thermophoresis,  
11 diffusiophoresis, things like that.

12 Staff has previously approved new light-  
13 water reactor applications that have used STARNAUA and  
14 the models in STARNAUA to calculate aerosol deposition  
15 factors. However, we have not necessarily reviewed  
16 the code for approval or endorsement itself. So we've  
17 not approved STARNAUA itself.

18 The staff has found the topical report  
19 methodology describes modeling of a possible aerosol  
20 natural deposition phenomena in containment and  
21 therefore find that it's acceptable.

22 CO-CHAIR PETTI: So the independent  
23 analysis done by staff --

24 MS. HART: Yes.

25 CO-CHAIR PETTI: -- it just stops at

1 source term in the containment? It doesn't use MELLOR  
2 to do the modeling in containment, because you can do  
3 that.

4  
5 MS. HART: The independent analysis that  
6 we did actually took the release from containment and  
7 disbursed it through the environment. So we did look  
8 at modeling within the -

9 CO-CHAIR PETTI: You did everything?

10 MS. HART: Yes.

11 CO-CHAIR PETTI: Good, okay.

12 CHAIR KIRCHNER: A little puzzled here,  
13 Michelle. Typically, elsewhere what you, the staff,  
14 present to us is that if an Applicant comes with a  
15 methodology like STARNAUA, you do a review of that  
16 methodology, the physical basis, the validation of the  
17 code, et cetera. And then you say, okay, STARNAUA is  
18 now an acceptable code for use in licensing  
19 applications.

20 But here you're saying in the view graphs,  
21 if I read this literally, that STARNAUA was not  
22 reviewed by the staff for approval or endorsement.

23 Did you, when you did your independent  
24 evaluation, then just bound that particular set of  
25 phenomena that STARNAUA is supposed to model and put

1 some conservative limits, so to speak, on your own  
2 analysis to see if all this detail in STARNAUA is even  
3 important or material to the final dose consequence  
4 analysis?

5 MS. HART: So we did, and some of this was  
6 done in the past with some of these analyses too, we  
7 have looked at the models that STARNAUA intends to  
8 implement. Like they were saying, it's a version of  
9 the NAUAHYGROS code which is a pretty well broadly  
10 used code.

11 However, when we evaluate what is  
12 important to us, as far as the dose analysis, is the  
13 removal coefficients that are used in the containment.

14 CHAIR KIRCHNER: Right.

15 MS. HART: And so we do an independent  
16 look with MELCOR's models to determine what the  
17 releases would be and what the removal would be in the  
18 containment. So that's how we evaluate whether we  
19 think it's a reasonable model or not. But we are not  
20 saying that we agree that the code, you know, is  
21 endorsed. We did not review the code itself.

22 CHAIR KIRCHNER: So I'm just, for  
23 completeness, curious how you -- because they make use  
24 of the STARNAUA code in their topical report, and  
25 henceforth in their actual analyses for the license



1 application. It just, I don't know, it strikes me as  
2 like a hole in the story.

3 MS. HART: But it's really, in the design  
4 certification, it's really the results of the STARNAUA  
5 code that we are finding acceptable. It's the  
6 methodology develops those. And so we have reviewed  
7 the final values as well.

8 CO-CHAIR PETTI: And they've compared, in  
9 their work they compared MELCOR's estimates of hold up  
10 in the containment to --

11 MS. HART: Correct.

12 CO-CHAIR PETTI: -- the Applicant's  
13 STARNAUA.

14 CHAIR KIRCHNER: Yes.

15 CO-CHAIR PETTI: So it's ironic that it's  
16 those removal coefficients is where the error was.  
17 And that's the important stuff with those processing  
18 --

19 CHAIR KIRCHNER: Right.

20 MS. HART: Yes, and we did have a lot of  
21 questions about, you know, we did not follow what  
22 they were doing for the follow-up. We also  
23 identified, in the RAI, we said, you know, this graph  
24 in your topical report does not look like we would  
25 expect to see as far as the removal coefficients and

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1 the amount in the containment. So that was a  
2 discussion we did have with them.

3 DR. SCHULTZ: So in either case, you have  
4 not reviewed or approved the STARNAUA code, but --

5 MS. HART: In and of itself, correct.

6 DR. SCHULTZ: In and of itself. But in  
7 the previous applications and in this application  
8 you've --

9 MS. HART: We've recognized it --

10 DR. SCHULTZ: -- approved the results.

11 MS. HART: -- as a reasonable code to use,  
12 yes.

13 DR. SCHULTZ: Yes. All right, thank you.

14 MEMBER BLEY: And it gets similar results  
15 to your own code, is really --

16 MS. HART: Correct.

17 MEMBER BLEY: -- what you're saying.

18 MS. HART: Correct.

19 MEMBER BLEY: Yes.

20 MS. HART: Okay, next slide, please. So  
21 now we'll describe a little bit about the staff's  
22 integrated evaluation of the core damage event  
23 analysis methodology. And our objective was to  
24 confirm their methodology for the radiological  
25 consequence analysis.

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1           NuSCALE did adopt two assumptions for Reg  
2       Guide 1.183, the containment leak rate, which is a  
3       design leakage rate for the first 24 hours of release,  
4       then half that rate for the remainder of 30 days. And  
5       that's a weight percent of the containment.

6           And then the iodine gaseous fraction, five  
7       percent of the iodine was assumed to be gaseous and  
8       not deposited in the containment.

9           For other aspects, NuSCALE developed their  
10      own methodology. In other words, their release  
11      timing was delayed compared to what we do for large  
12      light-water reactors. And their magnitude was lower  
13      than we assumed for large light-water reactors.

14           Next slide please.

15           CO-CHAIR PETTI: So just, they did not, in  
16      terms of the leak rate, those were deterministically  
17      set. They didn't use their pressure calculation to  
18      modulate the leakage -

19           MS. HART: Correct. And that was very  
20      similar to what was, I mean, that's what we've been  
21      doing for large light-water reactors. And it's based  
22      on, the regulation talks about using a demonstrable  
23      leak rate. And so they have always taken the  
24      technical specification leak rate.

25           However, the containment for NuSCALE is

1 much smaller. It has a lot of hydrogen in it as  
2 compared to those larger containments. So we did have  
3 some concerns about that, especially based on our  
4 severe accident analyses that we had done with MELLOR.

5 The next slide, please. So to go into  
6 that, the initial staff concerns were that the  
7 containment leak rate depends on containment  
8 atmosphere composition. Like I was saying, there's  
9 mostly hydrogen for NuSCALE versus mostly air for  
10 larger containments.

11 There was an uncertainty in the extent of  
12 severe core damage. So when we compared our severe  
13 accident analyses, we were getting a higher amount of  
14 core damage than NuSCALE was showing in their design  
15 certification analysis.

16 We had some questions about some of the  
17 assumptions that they were using in their aerosol  
18 deposition calculation, the amount of non-radioactive  
19 aerosol and the thermohydraulic conditions, the use of  
20 the conditions that went with Scenario 2 instead of  
21 with the other scenarios.

22 And the magnitude of release, the core  
23 release fraction in the aerosol deposition rate were  
24 taken from different scenarios. So you had the  
25 magnitude of release was from one that had a later

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1 release as compared to the aerosol deposition rate  
2 which was based on an earlier release.

3 MEMBER REMPE: Michelle, I don't know if  
4 you want to talk about the magnitude of the non-  
5 radioactive aerosols, but I looked at the report, and  
6 you, or someone, whoever wrote it, referred to this  
7 RAI. And I actually pulled up the RAI and reviewed  
8 it. But what I didn't know was how you felt about  
9 their response.

10 And if you want to wait and have someone  
11 talk about that in a closed session, that's fine. But  
12 I just was a little puzzled by what I saw in the  
13 report.

14 MS. HART: I think in general terms, the  
15 one of the reasons that we did this independent  
16 analysis was we didn't necessarily agree with all of  
17 their responses in the RAIs to all of the aspects.  
18 And so we wanted to determine if it was something  
19 further that we needed to follow-up on.

20 MEMBER REMPE: So maybe let's talk about  
21 it a little bit more. But I just was, it was kind of,  
22 like, well, we asked for an RAI, and then I didn't  
23 know, well, what happened. Did you like what they  
24 said or not? So anyway, let's talk about it more.

25 MS. HART: Okay. This analysis also

1 helped us evaluate positions, several of the positions  
2 in the topical report. And we evaluated these  
3 positions that they wanted approval on in an  
4 integrative fashion by this independent confirmatory  
5 analysis that we did using MELLOR and RADTRAD.

6 Next slide, please. So for the MELLOR  
7 modeling of the core damage releases, we did take best  
8 estimate prediction as a release to the environment  
9 for two core damage scenarios. And they're described  
10 there.

11 The staff independently developed MELLOR  
12 input models using design certification application  
13 information. This is not related to the example  
14 calculations in the topical report.

15 So we did, because we were reviewing the  
16 design certification at the same time as reviewing the  
17 topical report, at this time we did use specific  
18 information on the plant design to help us with this  
19 evaluation.

20 DR. CORRADINI: This was two of the five,  
21 though, was it?

22 PARTICIPANT: Yes.

23 DR. CORRADINI: That's my memory.

24 MS. HART: These are two of the five,  
25 that's correct.

1                   Next slide, please? The staff then took  
2                   those releases to the environment. So this did  
3                   include the modeling within containment, the aerosol  
4                   deposition, and developed a RADTRAD model using those  
5                   predicted releases to the environment for each of the  
6                   two scenarios.

7                   The other dose analysis assumptions, like  
8                   the atmosphere dispersion factors, the breathing  
9                   rates, the dose conversion factors, were taken from  
10                  the topical report methodology as informed by the  
11                  design certification application values for the  
12                  control room modeling.

13                  As we had discussed in the subcommittee  
14                  for Chapter 6, there were two ventilation system  
15                  cases, one for the containment, control room  
16                  habitability system, which is the bottled air system  
17                  operating for 72 hours. And then the normal control  
18                  room HVAC systems comes back in and filtered operation  
19                  for the remainder of the accident duration.

20                  And the other case is that the normal  
21                  control room ventilation system is in filtration mode  
22                  for the duration of the accident.

23                  When we do these analyses, we predict  
24                  doses comparable to the Applicant's results and below  
25                  the regulatory dose criteria. Our results were

1 higher, but they were well within the uncertainty that  
2 you would expect for these types of analyses.

3 Next slide please. So to resolve these  
4 issues, we determined that the issues did not require  
5 additional follow-up based on our confirmatory  
6 analysis, or independent analysis, excuse me.

7 Considering the technical basis as  
8 described in the topical report, along with our  
9 analysis of the sensitivity of the overall dose  
10 results, to the uncertainty in the dose analysis  
11 modeling of these phenomena that we had questions on,  
12 the staff found the methodology to develop the core  
13 damage source is acceptable and found that those  
14 positions that we are evaluating are acceptable.

15 DR. SCHULTZ: Michelle, you mentioned that  
16 the evaluations that you performed were then expected  
17 within a typical uncertainty range of similar results,  
18 and that they were within limits. How much margin to  
19 limits did your results demonstrate?

20 MS. HART: So for the offsite, we were,  
21 you know, within an order of magnitude of the  
22 offsite.

23 DR. SCHULTZ: And for the control room?

24 MS. HART: For the control room, actually,  
25 the control room, we were about halfway to the limit.



1 We were around three to four, I mean, two to three rem  
2 --

3 DR. SCHULTZ: Okay.

4 MS. HART: And we did both cases.

5 DR. SCHULTZ: Right. That was the highest  
6 -

7 (Simultaneous speaking.)

8 DR. SCHULTZ: -- the higher of two, the  
9 higher of the cases was in the range of two to three  
10 rem?

11 MS. HART: If I remember correctly.

12 DR. SCHULTZ: Okay.

13 MS. HART: I don't have that note in front  
14 of me.

15 DR. SCHULTZ: Thank you.

16 CHAIR KIRCHNER: Perhaps we can go into  
17 this in more detail in the closed session later. But  
18 I lost the thread there for a moment. I'm just  
19 curious. The difference in your results versus those  
20 of NuSCALE, is the main factor what comes out of the  
21 MELLOR analysis -

22 MS. HART: So because we varied -

23 CHAIR KIRCHNER: -- in determining the  
24 dose?

25 MS. HART: -- several things, it's hard to

1 tell what may be driving that. I think one of the  
2 major things that we did have in the MELLOR analysis  
3 is a larger amount of core damage, a larger amount of  
4 release --

5 CHAIR KIRCHNER: You mentioned that.

6 MS. HART: -- to the containment. Yes.

7 CHAIR KIRCHNER: So perhaps we can explore  
8 that. But that comment you made earlier, that led me  
9 to think that maybe the biggest difference is  
10 attributable to the MELCOR results. I mean, actually,  
11 well, that's where the release comes from.

12 MS. HART: Yes. Well, I mean, the source  
13 term is based on MELLOR in a lot of the aspects. So  
14 yes, I don't believe that the difference that you may  
15 potentially get from a variable containment leak rate  
16 using more specific scenario thermohydraulics, which  
17 we did use in our confirmatory analyses, was driving  
18 that difference. It did --

19 CHAIR KIRCHNER: I wouldn't expect it to  
20 be in things like atmospheric dispersion factors,  
21 those are --

22 MS. HART: No, those were exactly the  
23 same.

24 CHAIR KIRCHNER: Those are pretty  
25 standard.

1 MS. HART: Yes.

2 CHAIR KIRCHNER: Yes.

3 MS. HART: We did not use anything  
4 different than they did.

5 CHAIR KIRCHNER: Thank you.

6 MEMBER REMPE: Actually, the site  
7 assumptions that they made, and I don't know, maybe  
8 this belongs in the closed session too, but they  
9 differed every time we've seen the Chapter 9 analysis,  
10 the PRA analysis, and this one.

11 And this one actually, I thought, was what  
12 they should have used all along. But I assume it  
13 doesn't matter when a COL applicant comes in. They'll  
14 have theirs, and they'll know they've got a bunch of  
15 different site parameters to compare against, right?

16 MS. HART: I'm not sure I understand what  
17 you're -

18 MEMBER REMPE: Sometimes they use Peach  
19 Bottom, sometimes they use Surry if you go back.

20 MS. HART: Oh.

21 MEMBER REMPE: And that part I'm pretty  
22 sure is in the open version of the DCA that they  
23 submitted. But it seems like that every time we see  
24 a different analysis, they use a different site. But  
25 anyway, that's --

1 MS. HART: Right.

2 MEMBER REMPE: -- something to think  
3 about.

4 MS. HART: Right. This specific analysis  
5 that we did, we used the design certification  
6 hypothetical chi over Qs -

7 MEMBER REMPE: Yes.

8 MS. HART: -- that they had developed  
9 through whatever means.

10 MEMBER REMPE: Yes.

11 MS. HART: Okay. So the next slide.  
12 Conditions and limitations, the approval applies only  
13 to the NuSCALE SMR design. They must maintain the  
14 same fundamental size, geometry, and safety features  
15 of the design docketed in 52-048.

16 There's no finding on the treatment of the  
17 core damage event as a beyond design basis event for  
18 the NuSCALE design. It's not germane for this  
19 particular purpose. And it's limited to specific  
20 assessments.

21 But by that I mean it's mostly with regard  
22 to the equipment. The environmental qualification of  
23 equipment is only, the description of the methodology  
24 is really only applied to under the bioshield and in  
25 containment.

1                   Evaluation is not for other areas  
2 necessarily, and there are several conditions that we  
3 placed on the use of the topical report for the  
4 atmospheric dispersion calculations by COL applicant.

5                   Next slide please. In conclusion, the  
6 staff found 14 of the 15 positions specified by  
7 NuSCALE Inspection 1.2 of the topical report to be  
8 acceptable. We found acceptable methods for  
9 developing the accident source term to performing  
10 accident radiological consequence analyses to be  
11 referenced by the NuSCALE SMR design or somebody else  
12 referencing that design.

13                  And the staff approves the use of the  
14 NuSCALE topical report, Revision 3, with the changes,  
15 that were proposed for the Appendix B for the  
16 environmental qualification of equipment, subject to  
17 the conditions and limitations specified in the safety  
18 evaluation report.

19                  So this is kind of a confirmatory item  
20 they need to make sure that that Appendix makes it  
21 into the final approved version of the topical report.

22                  MEMBER MARCH-LEUBA: Michelle, can you go  
23 back to Slide 35 --

24                  MS. HART: Yes.

25                  MEMBER MARCH-LEUBA: -- the conditions?

1 Yes, how about power. If they decide to uprate by two  
2 percent, is this topical report still applicable, or  
3 do they need to start all over --

4 MS. HART: So if it maintains the same  
5 fundamental size, geometry, and safety features of the  
6 design docketed in 52-048, we would have to have that  
7 discussion. But it's not automatically assumed to be  
8 applicable to an uprated version of the design.

9 MEMBER MARCH-LEUBA: So it is not  
10 automatically --

11 MS. HART: Not automatically.

12 (Simultaneous speaking.)

13 MEMBER MARCH-LEUBA: -- some discussion?

14 MS. HART: We would have to evaluate that.  
15 I think, you know, some of the decisions that we made  
16 were based on, as we were saying, the geometry. This  
17 is about the direct dose. This is about the  
18 environmental qualification aspects.

19 MEMBER MARCH-LEUBA: Yes. But, I mean,  
20 it has to be major geometry. If they decide to make  
21 a 12-inch pipe a 13-inch pipe, they need to come back  
22 and do it again?

23 MS. HART: I don't know.

24 DR. SCHULTZ: But even if there were a  
25 power uprate, that would have to be evaluated with

1 regard to radiological consequences, no matter how  
2 much that power uprate might be.

3 MS. HART: Correct. And you would have to  
4 still do your environmental qualification equipment  
5 and so forth.

6 DR. SCHULTZ: Right.

7 MEMBER MARCH-LEUBA: But typically, you  
8 will have a topical report that is approved with an  
9 associated SER they can use to do the -- the way I'm  
10 reading that first condition, it may not allow you to  
11 do it.

12 MS. HART: That is correct. It's  
13 something that there is a limitation there that's  
14 talking about that particular aspect that you need to  
15 justify whether this topical report really applies to  
16 your --

17 MEMBER MARCH-LEUBA: Yes. Just thinking  
18 ahead that is going to cause problems in the future.  
19 I mean, I think it's a little too restrictive.

20 MS. HART: That completes my presentation.  
21 Are there any further questions?

22 CO-CHAIR PETTI: Thank you, Michelle.  
23 Members, further questions at this point?

24 I think before we -- I know we need a  
25 break. But before we go to the break, this would be

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1 a good opportunity for taking any public comments. So  
2 why don't we get set up on the phone?

3 MR. SNODDERLY: This is Mike Snodderly,  
4 from the ACRS staff. Before we open up the public  
5 phone line to request any public comment, we're about  
6 a half hour ahead of schedule. And the next public  
7 session would be scheduled to start at 1 o'clock.

8 Do you want to tell the public that you  
9 plan to start earlier, or do you want to stay with 1  
10 o'clock to start the second session, or did you want  
11 to move it up to 12:30? Otherwise we need to let the  
12 public know that we're going to start at 12:30 instead  
13 of 1:00.

14 CHAIR KIRCHNER: Now, hold on. Let's just  
15 test the members first. Do we have any need to have  
16 another open session?

17 MR. SNODDERLY: The agenda calls for an  
18 open session, yes, in the afternoon to discuss the  
19 application of the, where we're going to go over the  
20 open items. If you look at the agenda, 1:00 p.m. is  
21 an open session presentation by NuSCALE and then the  
22 staff --

23 CHAIR KIRCHNER: Oh, I see.

24 MR. SNODDERLY: -- to do the open items,  
25 so --



1 CHAIR KIRCHNER: So this will be the  
2 broader conversation of the source term as it impacts  
3 the other --

4 MR. SNODDERLY: Exactly. It's the open  
5 item discussion for the area of focus.

6 CHAIR KIRCHNER: What do you think?

7 CO-CHAIR PETTI: I have a feeling we  
8 should stick with the schedule, and perhaps the closed  
9 session could go a little longer.

10 MR. SNODDERLY: Very good. Yes, I just  
11 wanted to make sure that if you did want to move it  
12 up, we need to tell people now. So we're going to  
13 stay --

14 CO-CHAIR PETTI: No.

15 MR. SNODDERLY: -- to the original  
16 schedule?

17 CO-CHAIR PETTI: Yes. We'll stick to the  
18 schedule as you have it then.

19 MR. SNODDERLY: Very good. And we'll give  
20 us --

21 MEMBER BROWN: What does that mean,  
22 original schedule? You're saying the next session  
23 start at 11:25, an hour from now.

24 MR. SNODDERLY: No, no, no, no.

25 CO-CHAIR PETTI: No, no, no, no. We take

1 a break.

2 MR. SNODDERLY: Well, we're going to open  
3 up the phone line and take public comments. We're  
4 going to take a break, and then we're going to go into  
5 closed session.

6 MEMBER BROWN: And then we'll do the open  
7 session starting at 1:00.

8 MR. SNODDERLY: 1:00, yes.

9 CO-CHAIR PETTI: That's good.

10 MR. SNODDERLY: That's what we're going to  
11 do.

12 CHAIR KIRCHNER: So with that are there,  
13 while we're waiting for the phone line to be opened,  
14 is there anyone with us who would like to make a  
15 comment? If so, please come forward, use the  
16 microphone, state your name, and make your comment.

17 Seeing no one, we'll just wait until we  
18 know the public line is open.

19 MR. SNODDERLY: So right now the phone is  
20 not muted. So I think if there's someone from the  
21 public that would like to make a comment, I think just  
22 take your phone off mute and begin to talk. And  
23 hopefully, we'll hear you. Is there anybody on the  
24 phone?

25 Sarah Fields, someone?

1 MS. FIELDS: Yes, excuse me. Yes, I am on  
2 the phone. My comment is that, except for one  
3 evaluation, it appears that everything that has been  
4 discussed is based on an accident only involving one  
5 module, not an accident involving more than one module  
6 or a number of them. Because there are going to be 12  
7 total modules inside the containment.

8 And I think that the NRC needs to do a  
9 more in depth evaluation of what would happen during  
10 accident scenarios involving releases from multiple  
11 modules and damages as to multiple modules. That's my  
12 comment. Thank you. Thank you for all the hard work  
13 on this, to the NRC staff.

14 CHAIR KIRCHNER: Thank you. Anyone else  
15 from the public?

16 MR. LEWIS: Hello? Is this line open?

17 CHAIR KIRCHNER: We hear you.

18 MR. LEWIS: Okay, this is Marvin Lewis.  
19 I'm sitting here in Pennsylvania, and I'd very much  
20 like to give a comment as a member of the public.

21 CHAIR KIRCHNER: Please go forward.

22 MR. LEWIS: Thank you. I've listened to  
23 many, many of these hearings, and I've participated in  
24 ASLB and whatever. And there's something that's  
25 really worrying me. It goes all the way to Three Mile

1 Island.

2 Three Mile Island was operating when it  
3 should have been shut down. And nobody bothers about  
4 that. Who cares? So it's operating when it was  
5 supposed to be off. And that is what I'm worrying  
6 about. You're doing a wonderful job following the  
7 letter of the law, following the rules and  
8 regulations, and I don't think it's for much  
9 (phonetic). Nothing's to it.

10 A man in California was able to kill  
11 himself while two correctional officers were within a  
12 few feet of him. He was under suicide watch, of  
13 course.

14 I think of all the times, all the numbers,  
15 all the letters, all the rules, all the regulations,  
16 all the goddamn thing he could wander about (phonetic)  
17 forever. I followed, I counted up (phonetic), I obey  
18 the rules, I obey the regulations. And it doesn't  
19 work.

20 A canister is dropped in California. No,  
21 it didn't get dropped. It hung up, luckily. Spent  
22 nuclear fuel, tons of it, it was hanging there, and  
23 nobody knew what to do for a while. Thankfully, these  
24 people got their heads together and did something  
25 worthwhile.

1           There's a nuclear reactor, ANO, Arkansas  
2       Nuclear One, something, all the rules, all the  
3       regulations, wonderful, wonderful, wonderful, until,  
4       thankfully, the electricians got there, lined up the  
5       resolve and went in. And finally, got the switch gear  
6       turned properly --

7           CHAIR KIRCHNER: Marvin, may I interrupt  
8       you?

9           MR. LEWIS: -- while they were in a room  
10      with water up their waist.

11          CHAIR KIRCHNER: Marvin? This is Walt  
12      Kirchner.

13          MR. LEWIS: I'm just trying to point out,  
14      there's a difference between sitting around and saying  
15      what's right and doing what's right. Thank you.

16          CHAIR KIRCHNER: Thank you, Marvin. Are  
17      there any other members of the public who wish to make  
18      a comment?

19          Hearing none, we will close the public  
20      line or mute the public line at this point.

21          MR. SNODDERLY: Until 1 o'clock, and then  
22      at 1 o'clock will restart the public line. But yes,  
23      now we will, given no other public comments, we'll now  
24      close the public line.

25          CHAIR KIRCHNER: Okay, thank you, Michael.

1 We're going to take a recess here for a break and come  
2 back at five of 11:00 on the clock on the wall here.  
3 And so we are now recessed. And we'll come back for  
4 a closed session.

5 (Whereupon, the above-entitled matter went  
6 off the record at 10:38 a.m. and resumed at 1:03 p.m.)

7 CHAIR KIRCHNER: Okay, we'll reconvene the  
8 meeting of the NuScale Subcommittee. And we'll turn  
9 to NuScale again to talk about applications of their  
10 source term methodology.

11 Carrie?

12 MS. FOSAAEN: All right, so I'm actually  
13 going to introduce Mark Shaver, who is in Corvallis to  
14 lead off this presentation, but again, if we have to  
15 have technical difficulties, we do have a plan here to  
16 take over. So Mark, to you.

17 MR. SHAVER: Yes, this is Mark Shaver,  
18 Radiological Engineering at NuScale Power. I  
19 apologize for not being there in person.

20 The agenda for this presentation is we'll  
21 go over some source term related open items and then  
22 applications of the accident source term NRSR. And  
23 really what I want to get out of this presentation is  
24 convey how our different source terms fit together  
25 with each other and how and where they're implemented

1 in our FSAR.

2 The next slide is acronyms for your  
3 referral for the presentation. Slide 5 and 6 are a  
4 list of previous source term-related open items. I'm  
5 not going to go over them all in detail, but it's just  
6 to kind of get a feel for where the open items were in  
7 the FSAR related to source terms. Most of them we're  
8 planning today as the topical report saying that they  
9 would be closed when the topical report methodology  
10 was approved. And I believe most, if not all of them  
11 are closed with little changes in the methodology. A  
12 lot of them were open just because at the time the  
13 draft SERs were made the final submittal had not been  
14 reviewed and subsequent to the review, I believe most  
15 of them have been closed.

16 Moving on to Slide 7, the source term  
17 overview. This is the same slide as our previous  
18 presentation that Paul mentioned we've come back to.

19 I'd like to go through some of the boxes  
20 that aren't in the topical content that are outside of  
21 that hashed line. At the very top is our bounding  
22 fuel isotopics and this was done with a lot of  
23 conservatism and it truly is bounding. There's an  
24 overpower margin of 102 percent power to maximum  
25 burnup and maximum burnup rate with zero credit for

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1 any decay time.

2 From those bounding fuel isotopics and the  
3 standard escape rate coefficients, we determined the  
4 normal primary coolant activity assuming 66 ppm fuel  
5 failure rates and that number was taken from EPRI data  
6 of the entire U.S. fleet and it was the maximum value  
7 in the latest ten years of data. So there's  
8 conservatism there and that's used for the normal  
9 effluent (phonetic) in Chapter 11 which we discussed  
10 at a previous ACRS meeting.

11 That normal primary coolant activity for  
12 conservatism has been multiplied by 10 for a fuel  
13 failure fraction of 650 ppm. So that's an order of  
14 magnitude above the highest industry volume.

15 DR. SCHULTZ: Mark, this is Steve Schultz.  
16 We talked a little bit about it this morning, but just  
17 to be sure we're on the same page with regard to the  
18 failed fuel fraction, the 66 ppm corresponds to how  
19 many fuel rods or what fraction of fuel rods?

20 MR. SHAVER: The 66 ppm for the normal  
21 primary coolant activity corresponds to between 6 and  
22 7, or between .6 to .7 rods per module.

23 DR. SCHULTZ: Okay.

24 MR. SHAVER: So it's slightly less than  
25 one rod per module.



1 DR. SCHULTZ: And so then if you multiply  
2 that by 10 to get the larger design basis PCA, and  
3 that corresponds to the tech spec activity?

4 MR. SHAVER: That is correct.

5 DR. SCHULTZ: And that would be six or  
6 seven rods.

7 MR. SHAVER: The design basis for a  
8 primary coolant activity which corresponds to the tech  
9 spec coolant levels is based upon 6 or 7 failed rods  
10 continuously in each of the 12 modules.

11 So that's been as a tech spec primary  
12 coolant activity that corresponds to that is what is  
13 input into the AST methodology. So from that  
14 standpoint our methodology isn't really different from  
15 previous applicants and that we take the tech spec  
16 value and then add the iodine spike on top of it.

17 The core damage dose, the core damage  
18 source term and associated doses also use those same  
19 bounding fuel isotopics as does the fuel handling  
20 accident and the rod eject. So the cases where we  
21 have fuel failure use those same bounding fuel  
22 isotopics and any other DBAs use the tech spec plus  
23 the iodine spike.

24 So in that regard, the input that go into  
25 the topical reports are very similar in terms of it

1 being the tech spec rather than the bounding  
2 isotopics.

3 And I'll note just for completion or  
4 completeness of source terms, we also do the gaseous  
5 tank failure analysis in Chapter 12 and that has a one  
6 percent fuel fail fraction just like previous  
7 applicants have done.

8 CO-CHAIR PETTI: So just to be clear, can  
9 we -- you're not using 660 ppm, you're using 1 percent  
10 to do the gaseous tank failure if that's what that  
11 means?

12 MR. SHAVER: Correct.

13 CO-CHAIR PETTI: Okay. Great. Thank you.

14 DR. SCHULTZ: Mark, we were discussing  
15 this earlier offline. The one percent failed fuel  
16 value comes from where? You said other applicants  
17 have used that in the past, but is that based on  
18 particular regulation or is that historical?

19 MR. OSBORN: Yes, that comes from the  
20 regulatory guidance and from a branch technical  
21 position on how to do the gas tank failure analysis.

22 DR. SCHULTZ: Thank you.

23 MR. OSBORN: Sorry, Mark.

24 MR. SHAVER: No problem, that's correct.  
25 And we didn't deviate from it. And how we got there

1 was by ratioing up our design basis values to one  
2 percent which is consistent with how previous  
3 applicants have done it as well.

4 So hopefully that gives a feel for how all  
5 the source terms fit together and also the level of  
6 conservatisms in each level as we go down to  
7 demonstrate the conservative nature of the source  
8 terms.

9 Moving on to Slide 8, this was discussed  
10 a little bit earlier this morning as well, but the  
11 core damage events and the iodine spike design basis  
12 source terms are used in conjunction to evaluate  
13 consequences or like in MHA-type applications for  
14 offsite and control room basis.

15 And the next Slide 9, to assure equipment  
16 functionality and to make sure that all of our  
17 equipment is designed to operate in the environment  
18 that it needs to operate in, we do a combination of a  
19 core event survivability and environment qualification  
20 in Chapters 19 and 3.

21 As was mentioned, the core damage is in 19  
22 and survivability and the environmental qualification  
23 uses the iodine spike system in Chapter 3.

24 CO-CHAIR PETTI: So again a question, if  
25 we can go back one slide to 8. I just wondered why

1       you didn't extend the blue dotted line to encompass  
2       the two boxes under single assembly activity content  
3       because those are done to evaluate radiological  
4       consequences, right?

5               MR. SHAVER: Well, yes. And in fact, the  
6       box could be the entire accident source term LTR box.  
7       If we go back to the previous slide it could have been  
8       that entire box because all of those cases were done  
9       for -- to demonstrate radiological consequences. So  
10      I struggled with what words to put on Slide 8, but  
11      kind of the point was we took this kind of idea of a  
12      maximum hypothetical accident, and as Paul said, we  
13      bifurcated it into a design basis, no core damage one,  
14      and a -- what we call beyond design basis core damage  
15      event. So they were using these two scenarios, you  
16      know, the extreme core damage and the LOCA surrogate  
17      we'll say, and we separated those. They had always  
18      been one analysis before. So that's what I was  
19      intending to point out here is that what had been done  
20      as a single analysis was split into two for us.

21              CO-CHAIR PETTI: Okay, but the FHA dose is  
22      actually the largest dose as I recall in reading the  
23      document.

24              PARTICIPANT: I believe the core damage  
25      event dose would be the largest --

1 CO-CHAIR PETTI: Oh, yeah, yeah, yeah,  
2 yeah. But, I mean, compared to the iodine spike one,  
3 it's the largest of the four.

4 MR. SHAVER: Yes, so for offsetting the  
5 control room, the core damage dose is the highest.  
6 FHA is the second. For EQ in containment the iodine  
7 spikes is the highest. Some of the doses out in the  
8 gallery spaces with some of the other design basis  
9 accidents, so it depends on what application and  
10 location as to which accident bounds.

11 DR. SCHULTZ: Mark this is Steve Schultz  
12 again. What's the significance -- I missed it this  
13 morning if it was stated, of the REA dose boxing a  
14 dotted line?

15 MR. SHAVER: Oh, yes, so we didn't in our  
16 topical report, we didn't and that's just assuming an  
17 assembly just for example calculations, but per  
18 Guideline 1.83 we didn't analyze it in Chapter 15 in  
19 the FSAR because there was no fuel failure from the  
20 rod eject accident. So we had it in our methodology  
21 how to do it, but we don't actually have doses from  
22 rod eject in Chapter 15.

23 DR. SCHULTZ: I understand. Thank you.

24 CHAIR KIRCHNER: Mark, this is Walt  
25 Kirchner. Just to follow up on Steve's question, when

1       you do your -- I don't recall off the top of my head,  
2       when you do your Chapter 15 rod ejection accident, are  
3       you using the new draft reg guide for that analysis as  
4       a guideline?

5               MR. GUINN:  Mark, this is Paul Guinn.  I  
6       can take that question and the answer is no.

7               CHAIR KIRCHNER:  So you're aware that the  
8       significant change in that draft reg guide is the much  
9       lower threshold for damage in terms of well, the  
10      metric I think they use is joules per gram of fuel.

11              The old reg guide which is quite out of  
12      date and hence the reason for the revision has a  
13      significant step change down in the threshold for fuel  
14      damage for reactivity insertion systems.  So it's  
15      guidance at this point, draft, so I just point that  
16      out.

17              I don't recall the details of your Chapter  
18      15 analysis as to how it would compare with the new  
19      draft guide.  Just a note.

20              DR. SCHULTZ:  Paul, have you looked at it,  
21      the new guidance, the draft guidance I should say?

22              MR. GUINN:  DG 11-99?

23              DR. SCHULTZ:  Is it 13-27?  It's on my  
24      desk.

25              MR. GUINN:  Well, again, I'm not

1 intimately familiar with it. I might have looked at  
2 it. I don't recall.

3 DR. CORRADINI: I think the methodology  
4 though is different, too. It's not just the -- the  
5 methodology, the old methodology was much more  
6 conservative with a higher energy input.

7 MEMBER MARCH-LEUBA: The concept of time  
8 and temperature.

9 DR. CORRADINI: DG 13-27, we were told.

10 MEMBER MARCH-LEUBA: Yes, 13-27.

11 CHAIR KIRCHNER: Its current status is  
12 it's gone out for a second round of public comment.  
13 I think that public comment period is over. Probably  
14 offline it's something to take a look at because in  
15 general my summary of it is that the thresholds for  
16 fuel damage had come down substantially since the  
17 original RIA reg guide was written in like 1977. So  
18 they have much more data from mainly international  
19 experiments, but the thresholds I think you'll see are  
20 much lower for certain classes of events like where  
21 you're concerned about joules per gram. And then  
22 there's the revision for the cladding as well which is  
23 the time and temperature as Jose mentioned.

24 MR. SHAVER: All right, moving on then to  
25 specific applications. We'll go to Slide 10. So

1 implementation and methodology in Chapter 2 is the chi  
2 over Q values. A lot of this is repeated from earlier  
3 discussions so I won't go into detail. We used NARCON  
4 which is based on ARCON to develop chi over Q in  
5 Chapter 2.

6 DR. SCHULTZ: So Mark, in your second  
7 sub-bullet on the atmospheric dispersion methodology,  
8 it says ARCON 96 used in controlling chi over Q is  
9 closer to, but you used NARCON, correct?

10 MR. SHAVER: Correct.

11 DR. SCHULTZ: For all your analyses.

12 MR. SHAVER: Correct.

13 DR. SCHULTZ: Thank you.

14 MEMBER REMPE: And I guess this is a good  
15 place to mention my comment earlier about that over  
16 the time that we've been reviewing this design, now  
17 you're using the EPRI user requirements document for  
18 your site parameters, but when we were reviewing the  
19 PRA in Chapter 19, we heard that you were using Surry  
20 or Browns Ferry and somehow or other you got some  
21 collection of site parameters when a COL applicant  
22 comes in, it will be trapped and taken care of  
23 appropriately?

24 MR. SHAVER: Yes, I believe so. We do use  
25 different chi over Q for different amounts in



1 different chapters, so yes, the severe accidents in 19  
2 and you know, the normal doses often goes to 11 and  
3 accident doses in 15 do all use different chi over Qs.

4 But we have explained in methodology in  
5 the FSAR how to determine those. And they do all have  
6 to be evaluated at the COLA phase to make sure they  
7 are appropriate.

8 DR. SCHULTZ: Is there any expectation, I  
9 mean, you've done it for many different applications,  
10 so -- and so you have an understanding or an  
11 appreciation for the differences between the various  
12 sets that you have used, determined to use, but is  
13 there any expectation that a licensee would have a  
14 difficulty in an application to a particular site in  
15 any of the evaluations that are provided?

16 In other words, if you used different ones  
17 in different applications throughout the overall  
18 process here, you certainly want to be sure that  
19 you've got the basic groundwork covered so that a  
20 licensee, an applicant isn't surprised in one of the  
21 analyses or another. You don't expect that to happen,  
22 do you?

23 MR. SHAVER: No, we don't. And for the  
24 accident curve, for example, we use between 80th and  
25 90th percentile, I believe, to bound most sites in the

1 U.S. So the approach was to take a national survey of  
2 data and pick something appropriate, you know, could  
3 there be a site where one of the chi over Qs is  
4 exceeded? There could, but I will also mention that  
5 we have margin to acceptance criteria for all of these  
6 as well. So even if site specific chi over Q would  
7 exceed it or exceeded with -- we don't necessarily  
8 anticipate, but it's possible. There's margin to  
9 acceptance criteria so we really don't think that that  
10 would be exceeded, that those would be exceeded.

11 DR. SCHULTZ: Thank you.

12 MR. SHAVER: Slide 11, the next  
13 implementation of the source term topical in Chapter  
14 3 -- or not source terms, not the topical, sorry,  
15 Chapter 3 EQ doses, and we have normal EQ doses that  
16 are based on our design basis fuel failure fraction  
17 which is also the tech spec DEIND xenon concentrations  
18 at a steady state. And so remember this is between  
19 six and seven failed rods per core, and we assume that  
20 it's in all 12 closed. So this is a lot of failed  
21 fuel, or there's a lot of fuel failed in the plants  
22 continuously for a 60-year lifetime to give a  
23 conservative lifetime equipment dose.

24 In Chapter 12, the accident environmental  
25 qualification doses are added to the 60 year normal

1 doses for the total dose that the equipment needs to  
2 survive and these EQ doses per accident are also based  
3 on the design basis fuel failure fraction because it's  
4 the tech spec value plus the iodine spike on top of  
5 it.

6 And I will note that we have a later  
7 revision now that I believe you've all seen of AST LTR  
8 with revision 4 and we expanded the scope of that to  
9 provide accident into both methodologies rather than  
10 just the first term. And that's because we had some  
11 conservatisms in not only our source term but also our  
12 dose methodology and we wanted to document them, to  
13 take credit for them, and also make them part of the  
14 overall method.

15 CO-CHAIR PETTI: So question in terms of  
16 this accident and Q dose, the iodine spike. You just  
17 assume it's one of the modules. Does the methodology  
18 require you to from an EQ perspective to look at more  
19 than one module at a time so that the actions can be  
20 cumulative in terms of the EQ? How does that work?

21 MR. SHAVER: It's a single module. All of  
22 our accident analyses is a single module and there's  
23 no regulation or guidance to assume multi-module  
24 phase.

25 CO-CHAIR PETTI: Thank you.

1 CHAIR KIRCHNER: It would seem to me just  
2 a thought, I would answer that question a little  
3 differently. That's the equipment that is in this  
4 environment is module by module. And not answer the  
5 question there's no regulation about multiple modules.  
6 Just an observation, just one person's opinion. I  
7 think of the equipment that you're qualifying is  
8 module by module in general. I'm trying to think  
9 through your system. Your other comment equipment is  
10 outside of the boundaries for this EQ process.

11 DR. SCHULTZ: Is that --

12 MR. SHAVER: That is true for the appendix  
13 methodology in Revision 4, but the accident dose is in  
14 containment and under the bioshield. You would only  
15 see doses from a single module anyway because that  
16 would be module-specific equipment.

17 Slide 13, Chapter 11 source terms. This  
18 is largely a repeat of the Chapter 11 and 12 HRS  
19 meeting that we had previously. These are source  
20 terms that I have discussed. We also have water  
21 activation products and corrosion activation products.  
22 It's all through the zones, together in the Chapter 11  
23 source terms. And our methodology hasn't changed  
24 since the Chapter 11 and 12 HRS meeting. I believe  
25 the only difference is that we set the tech spec equal

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1 to the design basis source term

2 Slide 14 is more information on the 11  
3 source terms, what we've gone over. It was again  
4 using industry data and between 90 and 95 percent of  
5 U.S. light water reactors have zero detects in 2010  
6 and most of them are from grid-to-rod fretting. We  
7 use natural circulation which should mitigate most of  
8 the mechanisms especially the largest. So we actually  
9 think that we'll have fewer rod fails in enlarged TWR.  
10 We didn't credit it, but we think it's just highly  
11 conservative to use industry data for our modules.

12 Slide 15, application (telephonic  
13 interference). That's the design basis short term,  
14 this is equal to the tech spec that's used for design  
15 and shielding and operations. And I will say the open  
16 items, the previous open item in Chapter 11 on the  
17 fuel failure question was arguably related to this  
18 source term on its implementation of shielding in both  
19 in Chapter 12. So that has been closed and we  
20 received the Chapter 11 SER, the draft. The SER was  
21 not open item.

22 So all of the shielding and radiation  
23 protection and dose analyses are all based on the  
24 design basis fuel failure fraction which is also the  
25 tech spec limit to ensure that it won't be exceeded

1 during normal operations.

2 Slide 16 goes into the Chapter 15 design  
3 basis events. This is the same list of events of the  
4 previous presentation that Paul gave. The difference  
5 here is rather than saying which reg. guide we took  
6 them from, the reg guidance, this is where they're  
7 implemented in the FSAR, so the design basis accidents  
8 are all in FSAR 15.0.3 in subsections.

9 DR. SCHULTZ: Mark, excuse me. I'm sorry  
10 to have to back you up, but on the previous slide on  
11 Chapter 12, I thought the figure showed Chapter 12  
12 evaluation connected to the 1 percent fuel failure  
13 assumption.

14 MR. SHAVER: The 1 percent fuel failure is  
15 only for the gaseous radwaste tank failure analysis.

16 DR. SCHULTZ: Right.

17 MR. SHAVER: Which is actually in Chapter  
18 11 of the --

19 DR. SCHULTZ: Okay. Okay. I --

20 MR. SHAVER: -- system failure analysis.

21 DR. SCHULTZ: Well, the figure shows it's  
22 running to Chapter 12. That's where I got confused.  
23 Okay. That clears my question. Thank you.

24 MR. SHAVER: Thank you. Yes, I think that  
25 might be a typo, I believe. That's actually Chapter

1 11.

2 Slide 17. Slide 17 shows the results from  
3 the design basis accident and as I mentioned, there's  
4 a lot of margin to the acceptance criteria. The  
5 highest of the offsite doses or controlled doses from  
6 the DBAs is a fuel handling accident. That again has  
7 plenty of margin. So while we think the chi over Qs  
8 are conservative again if they go up on a particular  
9 site, we're not anticipating a large impact.

10 CHAIR KIRCHNER: Mark, this is Walt  
11 Kirchner. Could you just for the record and for my  
12 memory tell us what the failed fuel amount is for the  
13 fuel handling accident?

14 MR. SHAVER: We assume the bounding fuel  
15 isotopics and we assume that one entire assembly fails  
16 and that is the fuel assembly that is being handled  
17 that has the accident, we assume, and entire failure  
18 as a whole -- assembly.

19 CHAIR KIRCHNER: So the 17 by 17 bundle,  
20 each of the fuel rods is ruptured essentially.

21 MR. SHAVER: Correct.

22 CHAIR KIRCHNER: Thank you.

23 DR. SCHULTZ: Plenty of margin here, Mark.  
24 I'm just picking at the results. Most of them are  
25 showing that the EAB and LPZ doses are the same. For

1 the primary coolant line break, you've got the EAB  
2 dose lower than the LPZ dose.

3 Have you looked into the reasons why that  
4 could be the case? I understand it's very, very low.

5 MR. SHAVER: I'll defer to Paul Guinn on  
6 that question.

7 MR. GUINN: Yes, this is Paul. I'd have  
8 to say I don't have the immediate quantitative answer  
9 to that or a qualitative explanation for the  
10 difference in that model. That's the primary coolant  
11 line break?

12 DR. SCHULTZ: Yes, the other ones are  
13 relatively consistent. That one is not, just in terms  
14 of the relative values.

15 MR. GUINN: Yes, we'd have to look at  
16 event progression, release timing --

17 DR. SCHULTZ: Sure.

18 MR. GUINN: Information for that one and  
19 compare it against, you know, that EAB dose in the  
20 worst two-hour window. LPZ is the full 30-day event  
21 dose, so I would speculate your answer lies somewhere  
22 in that.

23 DR. CORRADINI: The rationale. So it's the  
24 timing?

25 MR. GUINN: My educated guess would be



1 that it would be a timing-related explanation for the  
2 model.

3 DR. CORRADINI: In Rev. 3 the numbers are  
4 different. Does that just mean we're out of date in  
5 Rev. 3 compared to what we have on the vu-graphs?

6 MR. GUINN: Yes.

7 DR. CORRADINI: Okay. Approximately,  
8 they're the same, but they're different in exact  
9 number.

10 MR. GUINN: Yes, there's been upstream  
11 changes to transient analysis based out of Chapter 15  
12 review that has happened in between the three.

13 DR. CORRADINI: That's fine. Thank you.

14 CHAIR KIRCHNER: Paul, just one quick  
15 thing on this chart. I'm missing something, but I  
16 thought the acceptance criteria was 5 rem and 25, 25  
17 for 2 hours and 25 for the duration. So what is 6.3?

18 MR. GUINN: Yes, some of the accidents  
19 have a different acceptance criteria and that's from  
20 the Reg. Guide. 6.3 is the acceptance criteria for  
21 fuel handling accident. Isn't that right? Yes.

22 CHAIR KIRCHNER: I wasn't aware of that.  
23 I thought the regulations trumped the Reg. Guides.

24 MR. GUINN: Oh, the -- well.

25 CHAIR KIRCHNER: You see what I'm saying?

1 You know, you go to 47, 52.47 or 50 point whatever  
2 number it is, and it's 25 rems in 2 hours and --

3 MR. BECKER: Staff can probably speak to  
4 this as well, but I think the implementation of reg.  
5 guidance is that the 25 rem EAB and LPZ --

6 DR. CORRADINI: You've got to get closer.

7 MR. BECKER: I think the history of the  
8 guidance is that the 25 rem EAB and LPZ regulatory  
9 requirement is applicable to eliminating what's been  
10 called the MHA or the radiological LOCA whereas for  
11 other events in their guidance they took the position  
12 that for more frequent events a threshold  
13 substantially less than that was appropriate, 10  
14 percent and 25 percent criteria for various events.

15 CHAIR KIRCHNER: Okay.

16 DR. CORRADINI: And that's only in the  
17 reg. guide. That's not in the regulation.

18 MR. BECKER: Correct.

19 DR. CORRADINI: So okay, fine. We can ask  
20 the staff further. Thank you.

21 MR. SHAVER: All right, Slide 18 is the  
22 Chapter 15 core damage event. And we've had some  
23 discussion about this in the previous presentation  
24 about we take -- accident scenarios derived from  
25 intact containment internal events from PRA and we use

1 those to inform a record under those core damage  
2 events that we also use in conjunction with the other  
3 design basis accidents.

4 As can be seen on Slide 19, these are --  
5 these doses are higher than the doses from the design  
6 basis accident which makes logical sense, if there was  
7 substantial core damage versus no core damage, but  
8 there's also still substantial margins to acceptance  
9 criteria.

10 Moving on to Slide 20, the application of  
11 the source terms in Chapter 19 is the core damage  
12 event is used as the radiation environment for  
13 equipment survivability doses. And as was mentioned,  
14 we looked at what equipment falls under survivability  
15 and we took the dose, the integrated dose from the  
16 core damage event over the necessary life time of each  
17 piece of equipment and compared it to each of the  
18 doses and we picked the higher of the two.

19 There was some discussion in the previous  
20 presentation already about how we would demonstrate  
21 equipment would survive those dose rates that's part  
22 of the survivability program. I don't really have  
23 anything additional to what was stated previously on  
24 that topic.

25 So as I mentioned at the beginning of the

1 presentation, these chapters all have open items in  
2 them for the topical reports most of them were just  
3 pointing to the topical reports saying that acceptance  
4 of these doses in SER is contingent on acceptance of  
5 the topical reports. And largely none of these had  
6 changed since previous presentations to the ACRS.

7 Are there any other questions about source  
8 term implementation in FSAR?

9 All right, the next part of our  
10 presentation will be given by Jim Osborn, I believe.

11 MR. OSBORN: Yes, thank you, Mark. So the  
12 remaining slides of this presentation are dealing with  
13 what we call other topics. These are topics that the  
14 staff attached to their review of the accident source  
15 term topical and so we wanted to address them briefly  
16 here.

17 So the three topics that were talked about  
18 was the post-accident sampling exemption request; then  
19 the second one is the applicability of GDCs, beyond  
20 design basis accident. And the third one had to do  
21 with the radiological consequence from a leak of a  
22 hydrogen monitoring system.

23 So the first one has to do with  
24 post-accident sampling. NuScale had applied for and  
25 requested an exemption from the regulation, 10 CFR

1 50.34(f)(2)(8) which requires post-accident sampling  
2 and it's one of the several TMI-related requirements  
3 that requires consideration of a core melt source  
4 term and so with the NuScale design post-accident  
5 sampling capability is not needed as there are other  
6 means to assess core damage, specifically under the  
7 bioshield radiation monitors and core accident  
8 temperature indicators.

9 And as the exemption request delineates,  
10 there's advantages to this, one that the source term  
11 remains inside containment. You do not in order to  
12 secure a post-accident sampling, you would have to  
13 un-isolate containment and spread the source term out  
14 in other parts of the plant. The advantage of not  
15 having to do that is if you keep the source term  
16 bottled up. And so that reduces chances leaks and  
17 spills and operator doses and offsite doses. So the  
18 exemption request is found in application Part 7,  
19 Chapter 16.

20 MEMBER REMPE: I'm a little slow, but on  
21 the -- I think it was the prior slide at the bottom,  
22 you talk about an RAI 9690, yes.

23 MR. OSBORN: Yes.

24 MEMBER REMPE: I did not see a copy of  
25 that when I looked at the list. Has it been sent to

1 us, Mike? I'd like to request it, please. Thank you.

2 MR. OSBORN: We'll talk about that some  
3 more in our response to that RAI.

4 DR. SCHULTZ: Jim, what is the status then  
5 of the exemption request?

6 MR. OSBORN: As far as I know we didn't --  
7 there's no outstanding issues, so I think the staff is  
8 agreeable to it.

9 DR. SCHULTZ: Okay. It's in front of the  
10 staff and they're agreeable.

11 MR. OSBORN: That's my understanding.

12 DR. SCHULTZ: Thank you. That was my  
13 understanding, too. I just wanted to make sure it was  
14 yours. Thank you.

15 MR. TEFAYE: This is Getachew. That  
16 exemption request evaluation is part of Chapter 9 and  
17 it's in the process of getting approved, so you should  
18 see it soon.

19 DR. SCHULTZ: Thank you.

20 MR. OSBORN: All right, so the second  
21 topic has to do with applicability of general design  
22 criteria and in an RAI response NuScale had stated  
23 this to the staff that general design criteria do not  
24 apply to beyond design basis accidents to which staff  
25 disagrees. So NuScale's understanding is that

1 principal design criteria are the high-level  
2 requirements that underlie the facility's design  
3 basis. 10 CFR 50.34 and its equivalents require an  
4 FSAR to state the principal design criteria for a  
5 facility and describes the design basis quote unquote  
6 the design basis and the relation to the design basis  
7 to the principal design criteria.

8 So 10 CFR 50 of Appendix A, as contained  
9 to the general design criteria, was created in the  
10 1970s to establish the minimum principal design  
11 criteria for similar light water reactors.

12 I have a quote there from a Federal  
13 Register that talks about this development of the GDC  
14 or the accidents used in the development of GDCs are  
15 normally design basis accidents and the NRC believes  
16 that it was not appropriate to address the near  
17 accidents or design requirements in the general design  
18 criteria.

19 So the clearest example I can cite for you  
20 on this is GDC 2 has to do with earthquakes, seismic  
21 events, right. So a facility is designed to  
22 withstand a certain seismic event and that becomes  
23 this design basis. So for a beyond design basis  
24 seismic event, you wouldn't expect the facility  
25 necessarily to be designed to that, too. It can't be

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1 designed to a seismic quality quantity X plus another  
2 seismic event of X plus. It has a design basis and  
3 therefore the GDCs establish what the design basis is  
4 going to be.

5 So the second bullet there is for large  
6 light water reactors, design basis LOCAs, radiological  
7 consequences from SRP 15-65 is based on a core damage  
8 event and so the control room goes to the GDC 19  
9 applied to this assessment.

10 Next slide.

11 CHAIR KIRCHNER: Could you stop for a  
12 moment though? I'm thinking ahead. This is a  
13 digression from your presentation.

14 You are going to make the argument that  
15 basically you can survive a severe accident scenario  
16 for an undetermined length of time, maybe 30 days or  
17 so, based on the performance of, among other  
18 components, your ultimate heat sink or the reactor  
19 pool which is built and qualified to the GDC. So are  
20 you trying to have it two ways? If we're going to  
21 accept that argument for severe accident say beyond 72  
22 hours, it's because the asset components that you  
23 built were built and qualified according to the GDCs.  
24 See where I'm going with this?

25 MR. OSBORN: Yes, not exactly.



1 CHAIR KIRCHNER: Not exactly.

2 MR. OSBORN: I guess you have to define  
3 what the beyond design basis event was, right? So you  
4 design your facility to a magnitude 9 earthquake. You  
5 have a 10 or 11 earthquake, your facility isn't  
6 guaranteed to survive.

7 CHAIR KIRCHNER: No, I understand that.

8 MR. OSBORN: So how could you mitigate an  
9 accident for 72 hours or 30 days in a beyond design  
10 basis event such as that?

11 CHAIR KIRCHNER: Go on. I'll think about  
12 this some more.

13 MR. OSBORN: Okay. So -- next slide.  
14 Thank you.

15 So related to GDC 19 and this is the  
16 discussion centered around this particular GDC. TMI  
17 action item 2(d)(2) which was codified in 10 CFR  
18 50.34(f)(2)(7) required that a design review be done  
19 to ensure that adequate shielding for operator access  
20 and equipment for a degraded core accident. And so if  
21 the GDC is applied to the beyond design basis events,  
22 why was it necessary to add this regulation that  
23 invokes a GDC 19 limit to a post severe accident  
24 condition?

25 So this TMI action required the design

1 review to ensure that the plant could comply with the  
2 5 rem criteria. But in general, the GDCs are not  
3 applicable to beyond design basis accidents unless  
4 they're specifically invoked by other regulations like  
5 GDC 19 was.

6 So NuScale's approach classifies core  
7 damage event as a beyond design basis event, so  
8 normally GDC 19 would not apply. However, because of  
9 the post-TMI action items and 10 CFR 50.34 (f)(2)(7)  
10 the 5 rem limit does apply.

11 DR. CORRADINI: I'm sorry, but I thought I  
12 had you, then you lost me. So the last two dashes are  
13 saying that TMI, the specifics of the TMI Item II.B.2  
14 have to be complied with?

15 MR. OSBORN: Yes.

16 DR. CORRADINI: Okay. But GDC 19 does not  
17 apply in this case, am I understanding this correctly?

18 MR. OSBORN: So --

19 DR. CORRADINI: You went through two pages  
20 of this and I thought I had you, but then you turned  
21 --

22 MR. OSBORN: I lost you at the end?

23 DR. CORRADINI: You lost me --

24 MR. OSBORN: I'm sorry.

25 DR. CORRADINI: -- at the end.

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1 MR. OSBORN: Okay.

2 DR. CORRADINI: I understand the last one,  
3 which is it's a TMI related item that has to be  
4 followed.

5 MR. OSBORN: So, it was specifically, GDC  
6 19 was specifically invoked by that TMI action item.  
7 And so, yes, it does apply and it applies to NuScale.

8 MEMBER BLEY: I forgot where I was going.  
9 So, you're using the TMI example to say that, in  
10 general, the GDCs don't apply for beyond-design-basis  
11 events, unless there's a specific regulation. Where  
12 -- can you summarize, from your point of view, where  
13 this stands between you and the staff now? And I  
14 don't want you to speak for the staff, but have you  
15 come together? Is this -- what's the real impact of  
16 this right now?

17 MR. OSBORN: That's a good question. So,  
18 I'm not sure where the staff stands on this issue --

19 MEMBER BLEY: Okay.

20 MR. OSBORN: -- right now.

21 MEMBER BLEY: Well, we'll hear from them.  
22 But how is it affecting you directly?

23 MR. OSBORN: So, there's no real impact,  
24 other than the fact that there's a general  
25 disagreement that GDCs are applicable to severe

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1 accidents.

2 MEMBER BLEY: Well, I guess we can ask the  
3 staff.

4 CHAIR KIRCHNER: I guess, yes, Dennis, I  
5 find that, boy, that's, you're reaching pretty far, in  
6 my opinion. That's just one member's opinion. The  
7 GDCs weren't just -- didn't evolve just to do  
8 design-basis accidents. The design-basis accident  
9 categorization probably followed the original drafts  
10 of the GDCs.

11 So, in custom, you may be correct that  
12 they are used primarily in analyses, Chapter 15 in  
13 particular, for design-basis accidents, but I would  
14 submit that when the GDCs were developed initially,  
15 this distinction that you're drawing did not exist.  
16 And in particular, the containment is there because it  
17 was envisioned as a barrier between an uncontrolled  
18 release from the core.

19 And then, over time, what evolved was this  
20 essentially leak-tight definition, which then became  
21 subject to interpretation. But it was quite clear  
22 when these general design criteria were developed that  
23 the containment was there for a severe accident.

24 MR. OSBORN: Yes.

25 CHAIR KIRCHNER: So, to say that the GDCs

1 do not apply for severe accidents or  
2 beyond-design-basis, I think is a reach.

3 MR. OSBORN: Okay.

4 CHAIR KIRCHNER: I'll be interested to hear  
5 what the opinion of the staff is.

6 MR. OSBORN: So, let me offer this. So,  
7 I'm not equating the terms, or I don't think we should  
8 equate the terms core damage event and  
9 beyond-design-basis event for everybody, right?

10 Because in the past, other plants, other  
11 applicants have included a core damage severe accident  
12 in their design-basis. That's why this distinction is  
13 important to NuScale, is because we are now  
14 classifying a core damage event as  
15 beyond-design-basis.

16 So, that changes some rules of the game.  
17 So, you can't equate the terms core damage and severe  
18 accident, because in the past, core damage has been  
19 part of the design-basis events.

20 DR. CORRADINI: I think we want to hear --

21 CHAIR KIRCHNER: I think that --

22 DR. CORRADINI: -- from the staff --

23 CHAIR KIRCHNER: -- we disagree --

24 DR. CORRADINI: -- we've heard from you.

25 MR. OSBORN: Okay.

1 CHAIR KIRCHNER: -- but we'll hear from the  
2 staff. But please go on --

3 MR. OSBORN: Okay. All right.

4 CHAIR KIRCHNER: -- with 19. As Mike  
5 observed, you lost me somewhere along the way, that it  
6 doesn't apply. The logic doesn't, for me, hold  
7 together.

8 MR. OSBORN: I understand. That's why I  
9 quoted out of the Federal Register, and that was, to  
10 me, telling that the NRC's position was that the  
11 general design criteria are normally evaluated using  
12 design-basis events.

13 MEMBER REMPE: Your comment about, this  
14 doesn't apply to everybody, the regulator has to have  
15 regulation that is somewhat applicable to people. So,  
16 if it doesn't apply to a subset, where's the cutoff?

17 MR. OSBORN: So, I wouldn't say that the  
18 regulations aren't applicable, that's not my point.  
19 The point is, you have to follow the regulations  
20 within a certain framework.

21 And that framework includes design-basis  
22 events and beyond-design-basis events, and how you  
23 handle that and what the rules for the valuations are.  
24 And so, historically, applicants have done it a  
25 particular way, NuScale is doing it a little bit

1 different way.

2 MEMBER REMPE: Because core damage doesn't  
3 occur in your design-basis.

4 MR. OSBORN: That's right.

5 MEMBER REMPE: And then, with what margin,  
6 what uncertainty? I mean, if they're just right below  
7 ten to the minus four, are we going to say, okay,  
8 you're good, you're not good? I mean, I'm just kind  
9 of wondering about --

10 MR. OSBORN: Yes. So --

11 MEMBER REMPE: -- I kind of had this  
12 problem with this discussion today a couple of times,  
13 I'm just wondering what's going to happen with the  
14 other folks, too? But anyhow.

15 MR. OSBORN: Yes. So, I mean, that goes  
16 into PRA space, right? And I'm sure we could have  
17 Scott Weber fill us in on those kinds of details  
18 related to probabilities and --

19 MEMBER REMPE: Well, it's something --

20 MR. OSBORN: -- how that --

21 MEMBER REMPE: -- the staff's going to have  
22 to deal with, but --

23 MR. OSBORN: Sure.

24 MEMBER REMPE: -- anyway, that's not your  
25 problem today. Just curious, though.

1           MEMBER BLEY: I really lose your argument,  
2           you keep pointing to other plants that have core melt  
3           accident in their design-basis.     None of the  
4           design-basis accidents that are analyzed, Chapter 15,  
5           lead to core damage.   Containment was added, as Walt  
6           was saying, to cover those cases where we go beyond  
7           that, where we have a major release, and not --

8           CO-CHAIR PETTI: Yes, but I --

9           MEMBER BLEY: -- bring reactors closer to  
10          --

11          CO-CHAIR PETTI: -- think we have to be  
12          careful of the terms.

13          MEMBER BLEY: -- populations.

14          CO-CHAIR PETTI: I agree with you when you  
15          say core melt, but does --

16          MEMBER BLEY: Core damage.

17          CO-CHAIR PETTI: Core damage, in my mind,  
18          is not core melt.   Core damage is, I've ballooned some  
19          rods and I've got some gap release and I've failed  
20          some fuel.

21          MEMBER BLEY:   And we don't have a  
22          containment for that purpose?

23          CO-CHAIR PETTI: Correct.   But what they're  
24          saying is, their equivalent core damage occurs at a  
25          probability significantly lower than that in a large



1 light water reactor. That's what I'm understanding.  
2 And so, they're trying -- to have them pull that all  
3 the way in to the edge of design-basis, you're not  
4 just below ten to the minus four, my guess is you're  
5 three to four orders of magnitude below ten to the  
6 minus four in the PRA for those events.

7 MR. OSBORN: Yes.

8 MEMBER BLEY: They're not the only plant  
9 that has --

10 CO-CHAIR PETTI: Yes.

11 MEMBER BLEY: -- that, probably.

12 CO-CHAIR PETTI: Yes.

13 MR. OSBORN: Okay.

14 DR. CORRADINI: It's an interesting  
15 philosophical discussion, but --

16 (Laughter.)

17 CHAIR KIRCHNER: It would seem to me,  
18 you're going to be well below five rem in the control  
19 room, no matter how you do this particular analysis.  
20 You've already shown us one example where you made the  
21 equivalent of a postulated major accident and the dose  
22 was two-point-whatever rem. So, you were comfortably  
23 under the five rem limit. So, I'm not getting the  
24 point of why GDC 19 does not apply.

25 DR. SCHULTZ: Or why you needed to or

1 wanted to not apply.

2 CHAIR KIRCHNER: Yes. Yes, I'm -- and why  
3 do you not want to ensure that the operators, even if  
4 you do not need them to take action, are well  
5 protected and well below five rem? I mean, it's just  
6 good design practice. I'm struggling with why you're  
7 making an issue here.

8 DR. CORRADINI: I guess I'm looking at the  
9 numbers, because they just showed us the numbers --

10 CHAIR KIRCHNER: Yes, in --

11 DR. CORRADINI: -- somewhere in here.

12 CHAIR KIRCHNER: -- earlier presentation.

13 DR. CORRADINI: And for a core damage  
14 event, they're within a factor of two and a half of  
15 limit. So, they're not far away, they're away.

16 CHAIR KIRCHNER: Yes. With some  
17 uncertainty.

18 DR. CORRADINI: With some uncertainty,  
19 which staff is going to get to based on their --

20 DR. SCHULTZ: But this has to do with  
21 access, operator access to equipment outside the  
22 control room.

23 (Simultaneous speaking.)

24 MR. RAD: Hi, I'm behind the column. This  
25 is Zack Rad.

1 (Laughter.)

2 MR. RAD: All right. This is Zack Rad,  
3 NuScale Power. I'm Director of Reg Affairs. So, this  
4 actually goes a little bit more to principle, and this  
5 goes actually all the way back to the beginning of our  
6 presentation on the distinction between  
7 beyond-design-basis and design-basis events.

8 And so, one of the distinctions we  
9 attempted to make in our topical report was that  
10 distinction. And what we heard back was, well, let's  
11 just call this a source term and not make that  
12 distinction. And for the purposes of a source term,  
13 that's kind of okay.

14 But when we start calling it an event and  
15 we don't make that distinction, we also start to bring  
16 in questions like, what about the equipment? If that  
17 equipment is mitigating a design-basis event, it  
18 actually falls into a different category.

19 And the NRC has made this distinction very  
20 clear. Equipment to mitigate design-basis events  
21 falls into a certain category. It actually falls into  
22 a category in this topical report, comes out of  
23 equipment survivability right into 50.49, equipment  
24 mitigating design-basis events, and right back into  
25 EQ.

1           So, it actually has a little circular path  
2           in this topical report. And that's one of the reasons  
3           we were trying to make that distinction.

4           So, there's other complications too. If  
5           you take a look at many of the GDCs and you apply  
6           them, you actually can't get to beyond-design-basis  
7           events.

8           So, if I have diversity and redundancy in  
9           my ECCS system, I never get to a peak cladding  
10          temperature of 2200 degrees and I always have a  
11          coolable geometry and minimum hydrogen production.  
12          So, I never have core melt, I never get to severe  
13          accident, right? I might have core damage --

14          MEMBER BLEY: If those things actually  
15          work.

16          MR. RAD: If -- well, then, I meet -- if I  
17          don't meet them, right? So, if I have a system,  
18          right?, that meets the GDCs, then I assume they work,  
19          right? And that's why I have diversity and  
20          redundancy.

21          So, the point being, when I apply them, I  
22          have systems that are adequate, right?, for protection  
23          of health and safety of the public. So, the  
24          distinction becomes important in principle, and that's  
25          actually how this conversation began, and it was

1 around GDC 19, and that particular element of it.

2 And so, I hope that frames the  
3 conversation in a little broader context, and why  
4 we're concerned about the definition and the legal  
5 aspects of it is because it has broader implications  
6 for us.

7 And I think, actually, probably for the  
8 industry as a whole, as the next applicant comes in  
9 with a design that has a different application and a  
10 different set of SSCs. Because we're only one degree  
11 of separation away from the large lights, you might be  
12 looking at something a lot different in a little bit.

13 MR. OSBORN: All right. Next --

14 CHAIR KIRCHNER: I think the pause is a  
15 sign to go on.

16 MR. OSBORN: Okay, very well. So, the  
17 third issue had to do with hydrogen monitoring system  
18 leak.

19 The staff requested in RAI 9690 that  
20 NuScale provide the methods, models, and assumptions  
21 for calculating the dose contribution from a potential  
22 release to the environment through a leakage of the  
23 systems used in post-accident monitoring of the  
24 hydrogen and oxygen concentration in the containment  
25 atmosphere.

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1                   So, in that RAI response, NuScale stated  
2                   that, one, the hydrogen monitoring capability is only  
3                   provided for the purpose of severe accident  
4                   management.

5                   An evaluation of postulated radiological  
6                   consequences of a potential release from this system,  
7                   as with the radiological consequences of a purge for  
8                   combustible gas control, which would be a very large  
9                   release, is not necessary according to the guidance of  
10                  Reg Guide 1.183.

11                  So, unlike other typical designs, the  
12                  systems used for hydrogen monitoring are used during  
13                  normal operations, such that any excessive leakage  
14                  would become evident. And if the system leaks,  
15                  operators would isolate the leak.

16                  And because this would be an unplanned and  
17                  unexpected activity, after a beyond-design-basis  
18                  event, therefore, a specific dose analysis is not  
19                  required. And it is NuScale's view that such leakages  
20                  have not be included in the past, because it is a low  
21                  risk item. These systems are included in the leakage  
22                  control program.

23                  So, yes, so we continued our response  
24                  saying that Standard Review Plan 15.6.5 and Regulatory  
25                  Guides 1.183 and 1.195 address neither combustible gas

1 monitoring leakage nor post-accident sampling releases  
2 and leakage.

3 Both combustible gas monitoring and  
4 post-accident sampling would be required functions  
5 following a core damage loss of coolant accident  
6 postulated as a design-basis event under a typical  
7 light water reactor Chapter 15.

8 Containment purging for hydrogen control  
9 for severe accident management is explicitly excluded  
10 from consideration in Reg Guide 1.183. Therefore, a  
11 much smaller release from a hydrogen monitoring  
12 related to hydrogen control should also be excluded  
13 for beyond-design-basis events.

14 The other point, NUREG-0737 identifies  
15 potential leakage outside containment as an issue to  
16 address by reducing leakage under TMI Item III.D.1.1,  
17 but it omits this known and potential leakage source  
18 as a source term to be evaluated for doses to  
19 operators under Item II.B.2 and III.D.3.4, which are  
20 the post-accident shielding and control room  
21 habitability items.

22 The hydrogen monitoring loop includes  
23 portions of the containment evacuation, the process  
24 sampling, and core flood and drain systems, and all  
25 these are included in the leakage control program.

1           And precedent also supports NuScale's  
2 position, including the ESBWR, the APR1400, and the  
3 ABWR as design examples, which include systems outside  
4 containment which may contain source terms following  
5 an accident.

6           And the fact that someone's hydrogen  
7 monitoring system may be Seismic Category I is  
8 irrelevant, because in a beyond-design-basis event,  
9 like a seismic event, Seismic Category I items are not  
10 assured to survive. This is why the requirements were  
11 relaxed for hydrogen monitoring in the revision of 10  
12 CFR 50.44 in post-severe accidents.

13           At NuScale -- well, so related to the  
14 other designs. The APR1400 includes a hydrogen  
15 monitoring capability outside containment, which is  
16 included in the scope of the leakage control program,  
17 without quantifying a specific acceptance criterion.

18           The advanced boiling water reactor  
19 includes 13 systems outside containment that may  
20 contain accident source terms, including post-accident  
21 sample, process sampling, containment atmosphere  
22 monitoring, fission product monitoring, et cetera.  
23 Such systems are subject to the leakage control  
24 program per the post-TMI item, but leakage from them  
25 is not included in the dose consequence evaluations.

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1           The NRC's SER for the ABWR concludes that  
2           a requirement for a COL applicant to create a  
3           procedure to reduce detected leakages to lowest  
4           practical levels, quote/unquote, satisfies TMI Item  
5           III.D.1.1.

6           So, NuScale revised their COLA Item 931 to  
7           explicitly identify this as low as practical  
8           requirement from the NUREG as the acceptance criteria  
9           for the leakage control program and to identify, as  
10          within the program scope, the systems and components  
11          used in post-accident hydrogen and oxygen monitoring  
12          of the containment atmosphere.

13          So, arguments of precedent and  
14          inapplicability to NuScale due to remote isolation  
15          capabilities are irrelevant, because isolation of the  
16          valves in a post-severe accident event will be handled  
17          under the severe accident mitigation guidelines.

18          DR. CORRADINI: Can I summarize what I  
19          think you just said, is that others have other methods  
20          of sampling, which have not been considered as part of  
21          the outside source term, and in your case, the leakage  
22          you would prefer not to be allowed to in a similar  
23          fashion?

24          MR. OSBORN: So, traditionally, what plants  
25          have been require to do is, in an accident scenario,

1 they assume containment leakage and consider leakage  
2 from their ESF systems that are routed outside  
3 containment.

4 NuScale has no ESF lines outside  
5 containment. And so, our only contribution to the  
6 offsite dose had to do with containment leakage.  
7 Staff wanted to add this other leakage pathway.

8 DR. CORRADINI: That part I got.

9 MR. OSBORN: Yes.

10 CHAIR KIRCHNER: And in your systems, this  
11 would be your containment, I'm forgetting the proper  
12 title of it, evacuation and fill system? It would be  
13 a branch off that, somewhere --

14 MR. OSBORN: Yes, it includes --

15 CHAIR KIRCHNER: -- in the reactor --

16 MR. OSBORN: -- part of that system.

17 CHAIR KIRCHNER: -- support building?

18 MR. OSBORN: All right. So, the core  
19 damage source term stems from a, in the NuScale  
20 design, from a beyond-design-basis event. The staff  
21 has postulated several hypotheticals that result in a  
22 containment vapor space leak through the hydrogen  
23 monitoring system after a severe accident.

24 For a severe accident or severe  
25 beyond-design-basis accident, so now you're

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1 postulating several what ifs. One is what if the  
2 plant decides that they need to unisolate containment  
3 and establish a hydrogen monitoring activity?

4 What if the RCS temperature hangs above  
5 200 degrees for an extended period of time  
6 post-accident, which has some implications on the  
7 containment isolation valve? And then, what if the  
8 system happens to leak excessively?

9 And so, NuScale believes that this  
10 represents a low safety risk and it is not required to  
11 entertain multiple what ifs in a specific GDC 19  
12 criterion.

13 So, as you have seen, NuScale has  
14 substantial margins in the onsite and offsite dose  
15 limits, even when analyzing the beyond-design-basis  
16 core damage source term in the same manner as a  
17 design-basis event.

18 Comparisons of the NuScale containment  
19 volume to other designs show that NuScale is about two  
20 to three times smaller on a per megawatt basis than  
21 the smallest large light water reactor. So, we're a  
22 little bit smaller.

23 However, the range of containment volumes  
24 in the operating fleet span more than a factor of ten,  
25 after you normalize for a megawatt output. So, if

1       there were no additional requirements imposed on the  
2       existing fleet, because of a factor of ten difference  
3       in a normalized containment volume, then NuScale  
4       should not be required to add this additional  
5       requirement, because they're a factor of two or three  
6       smaller than the existing fleet.

7               MEMBER REMPE: I guess I was curious about  
8       what this means, and this is probably my ignorance,  
9       but the staff said that they would exclude it --

10              MR. OSBORN: Yes.

11              MEMBER REMPE: -- from issue resolution,  
12       what does that mean? I thought you were --

13              MR. OSBORN: I'm getting --

14              MEMBER REMPE: -- getting ready to go to  
15       another slide, I --

16              MR. OSBORN: No, I'm getting to that.

17              MEMBER REMPE: -- just wanted you to talk  
18       about it. Okay.

19              MR. OSBORN: No, this is my last slide. I  
20       think anyway, yes.

21              DR. SCHULTZ: But you're going to -- just  
22       to summarize, your RAI response was provided a --

23              MR. OSBORN: Yes.

24              DR. SCHULTZ: -- month and a half ago and  
25       the staff has reviewed that and they still hold this

1 position.

2 MR. OSBORN: Yes.

3 DR. SCHULTZ: And so, now, you're going to  
4 describe what that means to NuScale and whether --

5 MR. OSBORN: Right.

6 DR. SCHULTZ: -- you feel it should be  
7 postponed, the decision should be postponed to a  
8 different time.

9 MR. OSBORN: So, well, NuScale believes  
10 that the existing guidance is adequate,  
11 notwithstanding the design differences, as these  
12 differences are not significant enough to warrant  
13 additional requirements.

14 So, the staff has chosen to address this  
15 issue by, as they coined it, carving it out in  
16 rulemaking, so, effectively deferring this issue to  
17 the COLA applicant. And so, it would require that any  
18 applicant using this design to do this dose analysis  
19 before a license would be granted.

20 DR. CORRADINI: The dose analysis, assuming  
21 hydrogen leakage through your monitoring system?

22 MR. OSBORN: That's correct.

23 DR. CORRADINI: Or leakage through your  
24 monitoring system?

25 MR. OSBORN: Yes, sir.

1 DR. CORRADINI: Okay.

2 MR. OSBORN: So, NuScale disagrees that  
3 this is an appropriate or necessary treatment, based  
4 on the reasons previously stated.

5 DR. SCHULTZ: So, is that -- have you done  
6 such an evaluation? I mean, is that something that  
7 has -- you've just determined that it really shouldn't  
8 be considered a reasonable or feasible evaluation that  
9 needs to be done --

10 MR. OSBORN: Right, no, we --

11 DR. SCHULTZ: -- therefore, you haven't  
12 done it?

13 MR. OSBORN: We have not done it.

14 DR. SCHULTZ: And then, what would a COL  
15 applicant do? I mean, is it --

16 MR. OSBORN: That's a good question. The  
17 --

18 DR. SCHULTZ: It seems like it's not a COL  
19 applicant, I don't see how it can be, because it  
20 really refers back to your design, your design of the  
21 systems.

22 MR. OSBORN: Right.

23 DR. SCHULTZ: So, not to address it in the,  
24 if you will, the design certification stage seems to  
25 be a problem. Not a -- problem, that's the wrong way

1 to put it. You and the designers would need to come  
2 in, or the COL applicant isn't going to put together  
3 a special design for their facility so that they can  
4 pass this issue, pass on this issue, they'll have to  
5 --

6 MR. OSBORN: No, yes --

7 DR. SCHULTZ: -- go back to NuScale and  
8 address it at a later time.

9 MR. OSBORN: Right, no, it just kicks the  
10 can down the road.

11 DR. CORRADINI: That's the best way to put  
12 it.

13 MR. OSBORN: Yes.

14 MEMBER REMPE: So, what are you planning to  
15 do about it? Are you going to try and submit some  
16 sort of request for a decision or are you just going  
17 to kick the can down the road? I mean, there's -- is  
18 it an ITAAC? Or what's going to happen here? Nobody  
19 knows? Okay. So, it's a COL --

20 MS. FOSAAEN: So, as it stands, we do have  
21 a COL item, as Jim described, that would require them  
22 to have leakage controlled as low as reasonable --

23 MR. OSBORN: As low as practicable.

24 MS. FOSAAEN: -- practicable. We  
25 understand the staff's position and their need to move

1 forward in this way, but we also have our position.  
2 So, as I understand it, the staff intends to move  
3 forward carving out the rule and we understand that's  
4 their position and their prerogative.

5 DR. CORRADINI: But I'm not still -- we're  
6 going to ask the staff the same thing, but I want to  
7 make sure I'm clear on your side. Your side of it is  
8 that it just should be left as low as practicable  
9 without a COL item?

10 MS. FOSAAEN: We're fine maintaining the  
11 COL item and we believe that's consistent with --

12 MR. OSBORN: We're talking about two  
13 different --

14 MS. FOSAAEN: Okay.

15 MR. OSBORN: -- COL items.

16 MS. FOSAAEN: Okay.

17 MR. OSBORN: So, the COL item that we  
18 currently have has to do with establishing a leakage  
19 control program to satisfy the TMI item. The COL item  
20 in rulemaking is that they would have to do this --

21 MS. FOSAAEN: Well, it's not a COL item.

22 MR. OSBORN: Well, I --

23 MS. FOSAAEN: It's a carve-out.

24 MR. OSBORN: Yes. So, I call it a COL item

25 --



1 MS. FOSAAEN: Got it.

2 MR. OSBORN: -- but it's really not.

3 DR. CORRADINI: We'll just call it the  
4 thing.

5 MR. OSBORN: The thing?

6 (Laughter.)

7 DR. CORRADINI: Because what, now?

8 MR. OSBORN: The other thing that the  
9 applicant has to do, which is do this dose analysis  
10 for a leaky hydrogen monitoring system.

11 DR. CORRADINI: With leaky, but unspecified  
12 leakage rate?

13 MS. FOSAAEN: They would need to determine  
14 that.

15 MR. OSBORN: Yes, that --

16 MEMBER REMPE: But the leakage rate would  
17 have to result in a dose that's within an appropriate  
18 limit. So, it's --

19 MS. FOSAAEN: Yes.

20 MEMBER REMPE: -- a variable rate, but as  
21 long as you make the dose requirement --

22 MR. OSBORN: Right.

23 MEMBER REMPE: -- it's acceptable, right?

24 MR. OSBORN: Is that a trick -- so, the --

25 MEMBER REMPE: Is that a trick question?

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1 (Laughter.)

2 MEMBER REMPE: I mean, you were basically,  
3 at first, you were saying it has to be as low as  
4 practical. Well, okay, it doesn't have to -- I mean,  
5 you could get down to a leakage rate that's very, very  
6 small. But as long as you meet the dose requirement,  
7 is what they're really saying.

8 MR. OSBORN: Well, that's part of the point  
9 is, these other leakages, these other systems have  
10 never been included in the dose analysis, consequence  
11 analysis. So, it was just as low as practicable, it  
12 wasn't at a dose, at/or, in other applicants.

13 DR. SCHULTZ: You're saying -- yes, that's  
14 what I want to get to. You're talking about other  
15 designs, other applicants, other evaluations.

16 DR. CORRADINI: It was never considered as  
17 at/or in past ones.

18 MEMBER REMPE: But now they have to, with  
19 this other thing, and if they do, the leakage of it  
20 plus the other things that are normally considered  
21 still have to meet the dose requirement. So, it's not  
22 --

23 MR. OSBORN: Yes.

24 MEMBER REMPE: -- as low as practicable, I  
25 mean, we're not trying to get to zero leakage, they

1 just are saying don't --

2 MS. FOSAAEN: Right.

3 MEMBER REMPE: -- increase stuff?

4 MS. FOSAAEN: With the proposed path by the  
5 staff, it would have to meet the dose acceptance  
6 criteria, whatever leak rate that is.

7 MEMBER REMPE: Okay.

8 MR. OSBORN: Right, yes. I'm sorry if I  
9 didn't understand your question.

10 MEMBER REMPE: I just was trying to  
11 understand when I saw the bullet, when the slides were  
12 sent out, I was like, well, jeepers, what does this  
13 mean? And then, understanding now, so thank you.

14 MR. OSBORN: Okay, yes.

15 CO-CHAIR PETTI: Any more questions?  
16 Members?

17 MEMBER BLEY: Yes, but I don't know quite  
18 how to phrase it. I'm going back three or four  
19 slides, back to where you began on hydrogen  
20 monitoring. You say hydrogen monitoring is provided  
21 only for severe accidents, not germane to any  
22 design-basis accident.

23 MR. OSBORN: Yes, that's correct.

24 MEMBER BLEY: But it's there?

25 MR. OSBORN: Yes, it is there, as it's

1 required to be there by 10 CFR 50.44.

2 MEMBER BLEY: But it's there, so given it's  
3 there, your argument about not needing to look at the  
4 dose from leakage, I'm not completely connecting on  
5 the argument. The argument is, it's there,  
6 everybody's got this and nobody else has had to do it?  
7 Is that the argument?

8 MR. OSBORN: Well --

9 MEMBER BLEY: Not quite?

10 MR. OSBORN: Not quite, but that's -- so,  
11 there are other ways that this, quote/unquote, problem  
12 or issue is addressed. One, it's considered -- it has  
13 to be part of the leakage control program.

14 It's a system that we normally use during  
15 normal operations. And so, if it got big leaks in it,  
16 we would know it. And it has not historically been  
17 included into the dose consequence analyses, because  
18 it's been considered low risk.

19 So, we believe that those features that  
20 NuScale has related to the leakage control program and  
21 used during normal operation contributes to the fact  
22 that this is still low risk, even for a NuScale  
23 design.

24 MEMBER BLEY: Okay. I guess I'll focus on  
25 the staff later on this, because that's the real

1 question is, what do they see is different here  
2 compared to other --

3 MR. OSBORN: And you'll probably hear  
4 things like our containment volume is so much smaller.

5 MEMBER BLEY: Yes. Well --

6 MR. OSBORN: And if you compare --

7 MEMBER BLEY: -- we'll see what we'll hear.

8 MR. OSBORN: And if you compare us to the  
9 large dry containments, we're a lot smaller. Not all  
10 designs are that large.

11 CO-CHAIR PETTI: But also, your normal  
12 operations source term is much smaller than assumed in  
13 many of the large PWRs. So, some ratios point in one  
14 direction, but if you correct for source term, it may  
15 in fact flip the other way, in terms of what the real  
16 dose consequence would be of dealing with leakage out  
17 of this valve, out of this system.

18 MR. OSBORN: Yes.

19 MR. RAD: This is Zack.

20 CO-CHAIR PETTI: Oh, okay, very good. Go  
21 ahead, Zack.

22 MR. RAD: Hi, it's me again, Zack Rad,  
23 NuScale Power. So, just at a high level, to backup,  
24 the primary reason we're bringing this up is because  
25 it was actually an RAI that was, and a topic, that was

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1 tied somewhat loosely to this topical report and some  
2 of the issues. So, that's why we want to make sure  
3 that everyone understands the reason that it's part of  
4 this discussion.

5 And also at a high level, one of the  
6 reasons that we have fundamental misalignment is  
7 really sort of simple, and I think it was actually  
8 summed up fairly well. The existence of this system  
9 and systems like it is well-known and the fact that  
10 they would contain accident source term under these  
11 postulated severe accident conditions is also  
12 well-known.

13 And the relative risk of such leakage is  
14 also well-known, and it's captured in NUREG-0737 as a  
15 post-TMI item. They were also not included, the  
16 postulated leakage from them was not included in  
17 calculated source term, like leakage from ESF systems  
18 was.

19 And so, our capturing them in our  
20 licensing basis consistent with other applicants, we  
21 believe to be adequate. And that's where our  
22 fundamental misalignment is.

23 So, again, brought it back because it was  
24 loosely tied to this, we think doing what other  
25 applicants have done is adequate for our application.

1 And that's a pretty good summary. And you'll see this  
2 again in the rulemaking as well, we think. That's our  
3 understanding of it today. Thanks.

4 CO-CHAIR PETTI: Thank you. Any other  
5 questions from the Committee? Okay. Well, let's  
6 change out, let the staff assemble.

7 MEMBER BLEY: Walt?

8 CHAIR KIRCHNER: Yes, sir?

9 MEMBER BLEY: Would you mind if I pose  
10 something to the staff before you get started?

11 CHAIR KIRCHNER: Not at all.

12 MEMBER BLEY: If this upsets your  
13 presentation, just tell me. While this discussion of  
14 the GDCs is fresh in our minds, would you be willing  
15 to talk about that first and let us kind of understand  
16 what you see as the difference between your position  
17 and the applicant's, and what other implications there  
18 might be toward it? Or if you want to put that off  
19 until later, it's okay, but it's fresh in my mind and  
20 it won't be forever.

21 DR. CORRADINI: Are you asking the hydrogen  
22 monitoring system and the relation to GDC or the  
23 broader question of the GDC?

24 MEMBER BLEY: Broader question of the GDC.

25 MS. HART: So, I can talk a little bit

1 about GDC 19 and the discussion that they had about  
2 that. This is Michelle Hart, I'm the Lead Reviewer.  
3 We have a discussion of it in SECY 19-0079, there's a  
4 whole long section about it.

5 And our discussion in there was also,  
6 well, they're doing the dose analysis to meet the five  
7 rem, they didn't provide us enough information from  
8 our standpoint to say that we agree that GDC 19 would  
9 not apply to a core damage event, so we're not making  
10 a finding in that specific event either. Or for that  
11 specific position either.

12 Although, that wasn't a position  
13 specifically that they asked for approval in the  
14 topical report --

15 MEMBER BLEY: Right.

16 MS. HART: -- so we didn't describe it in  
17 the topical report. We do describe it in the SECY.

18 MEMBER BLEY: In the SECY, and I have that.  
19 I'm just -- I kind of stumble over it, because it  
20 seems they meet the GDCs, in any sense I think of, and  
21 I --

22 MS. HART: Well, and it's --

23 MEMBER BLEY: -- I don't understand --

24 MS. HART: Right, it's --

25 MEMBER BLEY: -- the play --



1 MS. HART: -- complicated for --

2 MEMBER BLEY: -- back and forth.

3 MS. HART: -- GDC 19 as well, because the  
4 TMI item does refer you right back to GDC 19 --

5 MEMBER BLEY: Yes.

6 MS. HART: -- for the dose criterion. So,  
7 the dose criterion is not any different. I think  
8 NuScale talked about it being a difference in  
9 principle, we did not see, as the staff, an issue with  
10 some of the changes that they wanted to make,  
11 especially with how they were treating the equipment  
12 survivability versus environmental qualification of  
13 equipment.

14 That we didn't see that bouncing back that  
15 they said that they saw and we did try to chase that  
16 down with them a little bit, and we didn't come to a  
17 common understanding with them on that either.

18 MEMBER BLEY: Thanks.

19 MR. STUTZCAGE: Can I just --

20 MEMBER BLEY: Yes.

21 MR. STUTZCAGE: Can I just add --

22 MEMBER BLEY: Please.

23 MR. STUTZCAGE: -- real quick? The only  
24 real difference in the NuScale design is for the  
25 environmental qualification under the 50.49 and GDC 4,

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1 which also has environmental qualification.  
2 Traditionally, it's been the core damage accident has  
3 been assessed for the radiation dose, and for NuScale,  
4 it's just the iodine spikes source term for their  
5 accident. Besides that, I don't see any other  
6 difference that NuScale has treated.

7 MEMBER BLEY: Okay, thanks.

8 MR. TESFAYE: Back on track, so let me do  
9 the introduction and then, we'll get going. Good  
10 afternoon, again, this is Getachew Tesfaye, I'm NRC  
11 Project Manager, as I indicated this morning. The  
12 second part of our presentation this afternoon will  
13 address the chapters that are impacted by accident  
14 source term methodology. Those who are presenting are  
15 seated here and will introduce themselves and we can  
16 get started.

17 MR. STUTZCAGE: Okay, yes. Ed Stutzcage,  
18 Radiation Protection Reviewer. I'll be talking  
19 through a variety of topics. I'll start with hydrogen  
20 and oxygen monitoring and go on to post-accident  
21 sampling, the exemption, and then, a little bit of  
22 design-basis failed fuel fraction, and then, the  
23 others beside me, Anne-Marie Grady and Michelle Hart,  
24 will pick up the last two topics. So, next slide,  
25 please. Okay.

1           So, on the hydrogen and oxygen monitoring,  
2           the radiological aspects. So, as NuScale discussed,  
3           10 CFR 50.44(c)(4) requires the capability to perform  
4           beyond-design-basis accident hydrogen and oxygen  
5           monitoring.

6           And then, the other items we list here,  
7           regulations we list here, are mostly TMI items. And  
8           they all include the footnote that has been kind of  
9           briefly discussed earlier, that kind of indicates that  
10          core damage accidents should be considered. And  
11          that's the same -- when we say core damage, we  
12          typically think of it as the Chapter 15 MHA-type,  
13          significant core melt accident, where I think a  
14          significant portion of the core is damaged.

15          So, all of these other regulations have  
16          that footnote. So, that was a consideration into how  
17          we evaluated this topic. Next slide, please. Okay.

18          So, just briefly describe NuScale's  
19          hydrogen and oxygen monitoring system. So, when you  
20          have an accident, severe accident, containment would  
21          automatically isolate. And then, if you were to  
22          perform hydrogen and oxygen monitoring, which the  
23          capability is required by the regulations, you have to  
24          unisolate containment.

25          And then, the fluid, the containment

1 atmosphere fluid runs through a portion of the CES  
2 system, then through the sampling system, where the  
3 monitors are located, and then, back through the  
4 containment portion of the containment flood and drain  
5 system to recirculate it into containment, so it isn't  
6 released anywhere.

7 So, in order to unisolate the containment,  
8 to initiate the hydrogen and oxygen monitoring, they  
9 have to -- manual actions are required outside the  
10 control room to unisolate the CES system.

11 So, NuScale evaluated the dose to a worker  
12 to go out to the hydraulic skids to unisolate the CES  
13 system. And there's two separate hydraulic skids,  
14 one's in the hundred foot elevation in the steam  
15 gallery, the other in the mechanical equipment area.

16 And the hundred foot elevation is where  
17 the piping for these systems, some of the piping for  
18 these systems are located. So, you'd expect they'd go  
19 there first to unisolate that and then, up to the  
20 126-foot elevation so that, when the system is  
21 unisolated, there would be lower doses in the area for  
22 the operator. So, next slide, please.

23 So, yes, so this discusses that. And we  
24 evaluated that under (f)(2)(vii), which is the  
25 requirements to be able to access important areas

1 following an accident.

2 And we looked at that, the sources from  
3 the containment vessel, as well as under the  
4 bioshield, what's in the under the bioshield source  
5 that leaks into it, as well as fluid in the HVAC  
6 system, which can ventilate the under the bioshield  
7 area.

8 And we found -- and NuScale's analysis and  
9 our confirmatory scoping calculation showed the dose  
10 to be well under five rem for initiating the hydrogen  
11 and oxygen monitoring. Next slide, please.

12 And then, we get through the discussion on  
13 leakage from the system. So, part of the concern is,  
14 we have information from NuScale indicating that they  
15 can unisolate the system to perform monitoring.

16 We haven't been provided information that  
17 assures us that they have the capability to re-isolate  
18 the system if they needed to, for example, if leakage  
19 was in excess of causing problems or anything like  
20 that.

21 So, we're not sure that they have the  
22 capability to do that. And at that point, you'd have  
23 the fluids in the lines, you could have potential  
24 leakage causing significant airborne activity,  
25 potentially.

1           So this gets us to the COL item. The COL  
2           item they have is for the leakage control program,  
3           ensures an ITP (phonetic), and as I said, it only  
4           ensures that the leakage is as low as practicable.  
5           There's no acceptable amount of leakage specified, so  
6           there's no assurance to us that the dose criteria  
7           could be met. Yes?

8           DR. CORRADINI: So, I'm trying to  
9           understand your logic, your logic is the issue is  
10          whether they can go back in and re-isolate? That's  
11          the issue --

12          MR. STUTZCAGE: Well, that was --

13          DR. CORRADINI: -- there's an -- if the  
14          leakage is too severe?

15          MR. STUTZCAGE: That was something that we  
16          considered. The way we see -- what we have right now  
17          is, we know they can unisolate the system.

18          We're uncertain if they can re-isolate it,  
19          which is part of the concern is that if the system is  
20          unisolated and there's leakage causing offsite or main  
21          control room dose concerns, they may not have a means  
22          to re-isolate it or that hasn't been demonstrated to  
23          us. So, that is part of the reason why we are  
24          concerned about the leakage from the systems.

25          Also, Michelle did some kind of scoping

1 calculations, with a range of information. There  
2 isn't much information in the application on the  
3 design of these systems.

4 But her scoping calculations indicate  
5 there's potential that the leakage from the system  
6 could be more significant than the containment leak  
7 rate and could even potentially result in exceeding  
8 the dose limits. And I get into that a little bit  
9 more on the next couple slides.

10 MEMBER BLEY: Did you ask them if they  
11 could re-isolate it and they just haven't answered?

12 MR. STUTZCAGE: We haven't -- yes, but,  
13 essentially, their response is that's something that  
14 is in the severe accident guidelines that they would  
15 have to figure out. They didn't provide the  
16 information to us.

17 CHAIR KIRCHNER: So, do you have a piping,  
18 an instrumentation diagram for this system?

19 MR. STUTZCAGE: Yes, the three systems,  
20 yes, we do and we could --

21 CHAIR KIRCHNER: Then, I would --

22 MR. STUTZCAGE: -- it's not in the  
23 presentation.

24 CHAIR KIRCHNER: -- expect this particular  
25 system would have two isolation valves. I don't know

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1 if they're manual or electric operated or solenoid,  
2 but it would -- there would be at least two isolation  
3 valves in series, wouldn't there?

4 MR. STUTZCAGE: Right. So, my  
5 understanding is that there's isolation valves for  
6 the, two for the -- well, there's actually --

7 CHAIR KIRCHNER: Two for the --

8 MR. STUTZCAGE: -- technically more than  
9 two.

10 CHAIR KIRCHNER: -- CES, for sure.

11 MR. STUTZCAGE: Two for the CES and then,  
12 two for the containment flood and drain system. My  
13 understanding is the containment flood and drain  
14 system valves can be operated from the main control  
15 room and the CES cannot. So, it requires manual  
16 action out in the field to unisolate or to operate  
17 those. I think Jim's got a --

18 MR. OSBORN: Excuse me, could I add a  
19 clarification?

20 MR. STUTZCAGE: Yes.

21 CHAIR KIRCHNER: Yes.

22 MR. OSBORN: So, Ed's right, the CFDS  
23 containment isolation valves can be operated remotely  
24 from a control room.

25 The containment isolation valves



1 associated with the containment evacuation system can  
2 also be operated remotely from a control room, if the  
3 RCS temperature is below 200 degrees. If the RCS  
4 temperature is above 200 degrees, then they cannot be  
5 remotely operated from the control room and an  
6 operator has to be dispatched.

7 So, there's a condition there. That was  
8 one of my what if's, right? What if the temperature  
9 is above 200 degrees? So, and it --

10 CHAIR KIRCHNER: And that is affected by an  
11 interlock on the valve?

12 MR. OSBORN: Yes, that's part of the I&C  
13 logic that I'm not very smart on. But, yes, that's  
14 the way it was designed.

15 CHAIR KIRCHNER: And are there manual  
16 isolation valves on --

17 MS. GRADY: I believe it's the containment  
18 isolation valve hasn't been reset, so it has to  
19 locally be opened.

20 DR. CORRADINI: Can you repeat that,  
21 please?

22 MS. GRADY: I believe that the signal, the  
23 containment isolation valve has been closed --

24 CHAIR KIRCHNER: Right.

25 MS. GRADY: -- and the signal has not been

1 removed so it could be reopened from the main control  
2 room, if the temperature is above 250 degrees. So, it  
3 has to be done locally, somebody has to go out and  
4 manually open the valve.

5 MR. OSBORN: Yes, the containment isolation  
6 signal will not clear --

7 MS. GRADY: That's right.

8 MR. OSBORN: -- until you're below 200  
9 degrees. And it can't be overridden from the control  
10 room, like the CFDS system can be.

11 CHAIR KIRCHNER: But if you dispatched an  
12 operator, that operator could isolate the system?

13 MR. OSBORN: That's correct, yes. At the  
14 hydraulic skid.

15 CHAIR KIRCHNER: So, wouldn't that address  
16 the leakage, subsequent leakage problem?

17 MR. STUTZCAGE: The concern is the  
18 radiation dose potentially to the worker going in the  
19 field to do that.

20 CHAIR KIRCHNER: I get that.

21 MEMBER REMPE: Periodically, there's got to  
22 be radiation monitors out there, right? And they  
23 could suit up if they needed to do so and go do it?

24 MR. STUTZCAGE: Yes, but as -- and this --  
25 for actions that are normally evaluated, require

1 post-accident, the applicants are expected -- normally  
2 evaluate under the (f)(2)(vii) requirement, to ensure  
3 that they can go out and perform the actions in the  
4 environment.

5 And that's a design requirement to make  
6 sure that there is appropriate shielding that they can  
7 actually do it, because -- following the TMI  
8 requirements, I guess, there's only so much you could  
9 do programmatically to ensure that doses aren't  
10 exceeded.

11 DR. CORRADINI: So, let me ask you a  
12 different question. There was another comment by the  
13 applicant that said that, in past, other advanced  
14 designs, this dose concern was not considered as part  
15 of the dose outside of containment, is that true?

16 MR. STUTZCAGE: Yes, that is true. In the  
17 other designs, like they said, who have EFS systems,  
18 the leakage from those systems is considered. One of  
19 the -- I mean, as pointed out in the next couple  
20 slides, we see several differences here.

21 One of them they mention is the difference  
22 in containment size and the power ratio does -- and we  
23 looked at more of the more recent reactors that we,  
24 new reactors we reviewed. But it results in a higher  
25 concentration of fluid in containment. And also,

1 NuScale has an overall much lower containment leak  
2 rate.

3 So, the potential impacts of the leakage  
4 are potentially higher in the NuScale design. And  
5 there's other design differences too, like they said,  
6 safety-related, the other systems are, they can be  
7 isolated from the control room in the other designs.

8 There's -- this involves three systems, up  
9 to four-inch piping. So, we see it as, there's  
10 potential additional concerns beyond some of what we  
11 saw on other designs.

12 MEMBER BLEY: So, it sounds like a design  
13 issue and putting it off to the COL, what happens out  
14 there? They do a calc and they got the dose you're  
15 worried about, what do they do?

16 MR. STUTZCAGE: Well, part of the -- we  
17 would expect a leakage amount to be specified. And  
18 then, their program would have to provide assurance,  
19 their normal operation program, that the leakage from  
20 the systems wouldn't exceed that amount during an  
21 accident.

22 Or if we had assurance that the system  
23 could be re-isolated, that was something that we would  
24 consider to potentially resolve the issue.

25 DR. CORRADINI: But I guess what Dennis is

1 asking is a design mod that ought to be settled at the  
2 design certification stage, not put it off --

3 MEMBER BLEY: Exactly.

4 DR. CORRADINI: -- to the COL.

5 MEMBER BLEY: If they have to --

6 DR. CORRADINI: If it truly is an issue  
7 that you guys can't come to resolution on, it's got to  
8 be resolved now, not punted down the stream.

9 MEMBER BLEY: Or after it's well along,  
10 they're going to have to make a design mod or --

11 DR. SCHULTZ: Right, it's not an analysis  
12 problem.

13 DR. CORRADINI: Right.

14 MEMBER BLEY: Yes, right.

15 CO-CHAIR PETTI: Can I just for  
16 clarification with the -- I remember reading the  
17 paragraph about the differences in the containment  
18 volumes and the differences in the power of the  
19 plants, but then I thought it said something about  
20 equivalent source terms.

21 MR. STUTZCAGE: Yes, and --

22 CO-CHAIR PETTI: I mean the source term  
23 for NuScale in their CDE event is significantly lower  
24 than the source term --

25 MR. STUTZCAGE: Yes.

1 CO-CHAIR PETTI: -- that would leak out of  
2 a hydrogen system of an ESBWR or one of these others.  
3 So have you accounted for that too when you say you  
4 have a concern about the operators if they open the  
5 valve?

6 MR. STUTZCAGE: I'm sorry, can you -- I'm  
7 not following you exactly. I'm sorry.

8 CO-CHAIR PETTI: So you said you did some  
9 calculations and you said that the doses looked  
10 potentially high, but did you use the NuScale source  
11 terms or the source terms from the other, the larger  
12 reactors?

13 MR. STUTZCAGE: Well, it's -- no, it's  
14 from -- based on the NuScale design, and they're just  
15 kind of scoping calculations, yes.

16 CO-CHAIR PETTI: But it's based on the  
17 NuScale core damage event source terms?

18 MR. STUTZCAGE: Right.

19 CO-CHAIR PETTI: Okay.

20 MR. STUTZCAGE: And what I meant by  
21 equivalent was -- is a similar -- the similar type of  
22 accident, not the same amount of core damage, but the  
23 -- a comparable -- the comparable MHA event, core  
24 damage event that's used --

25 (Simultaneous speaking.)

1 CO-CHAIR PETTI: The wording was just a  
2 little --

3 MR. STUTZCAGE: Yes.

4 CO-CHAIR PETTI: -- wonky for me.

5 MR. STUTZCAGE: Okay.

6 MEMBER MARCH-LEUBA: But, let's talk from  
7 the other point of view. The whole problem is that  
8 these valves cannot be insulated or open or closed  
9 remotely because there is a CIS, containment isolation  
10 signal, just a five-bolt on an open line that is  
11 preventing you from doing it remotely. This is a  
12 logic issue. I mean if you need five keys to be able  
13 to defeat it, you put five keys. But this is a design  
14 malpractice. Yes, we are asking the operator to go  
15 into a radiation area and get the dose, probably five  
16 rem, which I don't want to get, and use a paper clip  
17 to bypass the five-bolt signal. Change the design and  
18 make it logical. I mean it is a cheap design. It is  
19 a minimal design, and let's do it right.

20 MEMBER BLEY: Well, that's kind of where  
21 I'm at, too, but the thing nobody has addressed, a lot  
22 of cases where you have a locked-in signal you lock --  
23 manually operate the valve. As soon as you let go, it  
24 goes back to the position it was in. Is that true  
25 here?

1                   MEMBER MARCH-LEUBA: No, he would use a  
2 paper clip to bypass the signal.

3                   MEMBER BLEY: Oh, he won't. They don't do  
4 that.

5                   (Laughter.)

6                   MEMBER BLEY: But he could. That's why  
7 they don't have a license, yes.

8                   MEMBER MARCH-LEUBA: That's what they do  
9 in BWRs to defeat the MSIB isolation on low level.  
10 They actually have the jumper cable. They don't use  
11 a paper clip, but they have jumper cables in a drawer  
12 in the control room, which I've seen many times. They  
13 go, they pick up the jumper cables and send the guy to  
14 the back to put the paper clip on. They do that in  
15 power plants today.

16                  CHAIR KIRCHNER: Edward, from a physical  
17 dose calculation standpoint, as we discussed, the  
18 source term that evolves into the containment is less,  
19 but the volume is significantly less. So are you  
20 saying that what would come through the system  
21 actually potentially did have higher dose in the  
22 sampling lines than you would see in a large PWR  
23 because it's diluted in the -- in a large dry  
24 containment considerably?

25                  MR. STUTZCAGE: Right, and that's



1 comparing to the --

2 CHAIR KIRCHNER: Yes.

3 MR. STUTZCAGE: Basically the APR1400 is  
4 one example.

5 CHAIR KIRCHNER: Yes.

6 MR. STUTZCAGE: Correct.

7 CHAIR KIRCHNER: But I think what several  
8 of us are struggling with, or I am, is this leakage  
9 control program that you're deferring to the COL, or  
10 some other design modification that -- the applicant  
11 has asserted that for other advanced large LWR designs  
12 this wasn't required, but -- is that correct?

13 MR. STUTZCAGE: The leakage control  
14 program was required. They all have it included. The  
15 difference is is that -- well, and they -- those  
16 designs didn't assess the leakage to the off-site and  
17 main control room dose and they didn't prescribe a  
18 clear leakage criteria for the maximum leakage for  
19 those systems. And here it's essentially the  
20 difference that we're looking for that -- we're  
21 looking for in this potential -- in this carve-out.

22 CHAIR KIRCHNER: So this carve-out though  
23 is being driven by your expectation that an operator  
24 dispatched to deal with this problem would receive a  
25 dose in excess of five rem?

1 MR. STUTZCAGE: They potentially could.  
2 That's -- and that's -- yes, that's part of it. The  
3 other part is that we see it as potentially more  
4 significant than the normal containment leakage for  
5 the NuScale design and --

6 CHAIR KIRCHNER: Yes.

7 MR. STUTZCAGE: -- that's the other part  
8 of the concern.

9 CHAIR KIRCHNER: Now you zeroed in on this  
10 system. Is the potential though greater in other  
11 systems? CBCS comes to mind.

12 MR. STUTZCAGE: For NuScale all those --  
13 that's -- they're isolated.

14 CHAIR KIRCHNER: Right, they're all  
15 isolated.

16 MR. STUTZCAGE: Right.

17 CHAIR KIRCHNER: So it's only because  
18 you're going to un-isolate this particular system?

19 MR. STUTZCAGE: Correct.

20 CHAIR KIRCHNER: I guess I'm with my  
21 fellow members. This is a design fix, not a COL item.  
22 But that's an opinion, not a conclusion.

23 MEMBER BLEY: Neither the staff nor the  
24 applicant have given us any reason this makes sense to  
25 put off to the COL stage.

1 MR. TESFAYE: May I add something? In the  
2 design certification stage what we consider is is this  
3 essentially a complete design? There are certain  
4 designs we defer to the COL stage. I have a couple of  
5 examples of NuScale's design that are going to be done  
6 at the COL stage. So we don't have to have a complete  
7 design for us to certify the design. That's why we  
8 call it a rule carve-out or an interface item.

9 DR. CORRADINI: But it's to no one's  
10 benefit to put this off if you actually think it's a  
11 problem. I don't see how it's a benefit to anybody.  
12 It's not a benefit to the COL applicant. It's not a  
13 benefit to the applicant or to you guys. You're going  
14 to see this all again and it's going to be uglier.

15 MR. TESFAYE: Yes, but we have to review  
16 what we're given. So at this stage we don't have  
17 enough information for us to make a decision on that  
18 particular aspect of the design.

19 DR. CORRADINI: Then you don't proceed  
20 until you get more information.

21 MR. TESFAYE: Yes, but we have essentially  
22 complete design with the exception of this --

23 (Simultaneous speaking.)

24 DR. CORRADINI: But you don't proceed.  
25 You stop.

1           MEMBER REMPE: The other cases with this  
2 wonderful radar-based level of things that we always  
3 -- I always harp about you've got radiation levels,  
4 you've got temperatures and all sorts of things it's  
5 got to meet. Why don't you get rid of as low as  
6 practicable that they've put in and put some sort of  
7 specification that the leakage has to be less than  
8 something so it's easier for the folks who are looking  
9 at what the COL applicant comes in has. Can you do  
10 something like that and give it a little more easy-to-  
11 meet-the-metric flavor?

12           MS. HART: I mean we did have some  
13 discussions about what would be the minimum leakage  
14 that you would expect, or the maximum leakage that  
15 could be allowed, and NuScale did not provide that  
16 kind of response during the discussion about this RAI  
17 response.

18           MEMBER REMPE: They may not like it, but  
19 as far as a certified design, then we know what it is  
20 and any other kind of requirement you want that needs  
21 to be put on it.

22           MR. CAMPBELL: So I'm Andy Campbell. I'm  
23 the deputy in DEX, which is -- because of the merger  
24 we're no longer in the same division, but I was at a  
25 lot of these audits. And the issue was can you

1 specify a leakage rate that would meet a dose  
2 requirement? And that was not done or provided to us.  
3 So the system may or may not be designed well enough  
4 to be able to do that. This is a -- a carve-out has  
5 been used previously. There are examples of that in  
6 the design rules of things that aren't really COL  
7 items, but relate back to the fundamental design of  
8 the system and to meet requirements of the  
9 regulations.

10 MEMBER REMPE: So --

11 MR. CAMPBELL: I think I've said that  
12 right, Ed. Correct?

13 MR. STUTZCAGE: Right.

14 MEMBER REMPE: Educate me. Give me an  
15 example of something that's kind of fuzzy that doesn't  
16 have a specific kind of balance with a design  
17 requirement, like a leakage rate. Give me another  
18 example. Because I don't know.

19 MR. STUTZCAGE: Yes, so another example we  
20 had for the NuScale design is they have the shielding  
21 -- shield walls for the outside of the containment and  
22 bio-shield area. They have significant penetrations  
23 for the HVAC lines and such. And they -- in the  
24 NuScale application they assume that it's a solid wall  
25 and then they say that the wall with have equivalent

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1 shielding as the solid wall. So it's another item  
2 that we propose to have as a carve-out as -- to ensure  
3 that the COL applicant provides either the equivalent  
4 amount of shielding or they assess the consequences of  
5 it not being equivalent. And that's significant for  
6 EQ and stuff -- and operator dose.

7 MEMBER REMPE: So a variable rate that  
8 meets the dose?

9 CHAIR KIRCHNER: So specifically that's  
10 for the shield blocks?

11 MR. STUTZCAGE: That's for the shield wall  
12 on the outside of the --

13 DR. CORRADINI: This was an open item that  
14 we had discussed, if I remember correctly, and it was  
15 supposed to be resolved. So the resolution there is  
16 another carve-out?

17 MR. STUTZCAGE: Correct, and there are  
18 examples of this in the -- and I don't know the -- I  
19 can't speak to the specifics, but there's examples in  
20 the ESBWR design where there were carve-outs. They  
21 weren't related to the radiation protection review.

22 MR. CAMPBELL: It was service water.

23 MR. STUTZCAGE: Okay. Next slide, please.

24 MEMBER REMPE: Actually there's one more  
25 thing, though. I get it if you say that the shielding

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1 or something has to be done so you meet the dose.  
2 This -- I think now I've heard from you it's really  
3 the applicant who said as low as practicable, not you  
4 guys. So in what you do in your carve-out will be a  
5 leakage rate that allows you to meet this dose. It  
6 will be a more specific thing than this as low as  
7 practicable.

8 MR. STUTZCAGE: Right.

9 MEMBER REMPE: That makes me feel a little  
10 bit better.

11 MR. STUTZCAGE: The way it's proposed is  
12 to meet the dose or they provide some kind of design  
13 feature that provides reasonable assurance that the  
14 leakage can be controlled, that it won't result in  
15 exceeding a dose problem.

16 DR. CORRADINI: One clarification: I just  
17 want to make sure I understand. So if they are  
18 required to do hydrogen monitoring and they then  
19 defeat, specifically defeat the needed isolation  
20 signal to go and open so that they can do the  
21 monitoring, and the monitoring is now stopped, have  
22 they now violated another one of the requirements  
23 since they've now stopped the -- they've essentially  
24 shut off -- they've re-isolated because they -- the  
25 source term is large enough that they can't deal with

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1 it? What am I missing?

2 MR. STUTZCAGE: That's a question for  
3 Anne-Marie, I believe

4 MS. GRADY: Yes, this is Anne-Marie Grady.  
5 As far as hydrogen and oxygen monitoring is concerned,  
6 they have to establish it, it has to be continuous,  
7 and it has to be functional. So it shouldn't be  
8 stopped.

9 DR. CORRADINI: It shouldn't be isolated?

10 MS. GRADY: But --

11 DR. SCHULTZ: So re-isolation is not an  
12 option?

13 MS. GRADY: To meet the regulation.

14 DR. SCHULTZ: Okay.

15 DR. CORRADINI: So are you telling me that  
16 you've just invented the catch-22 that I -- that they  
17 -- even if they wanted to re-isolate, they shouldn't  
18 re-isolate? Is that what you just told me?

19 You can say yes.

20 MR. STUTZCAGE: I don't know that that's  
21 the case. I think there was some discussion about the  
22 ability to -- as long as they can get the information  
23 that may be sufficient to re-isolate. I think we'd  
24 have to get back to you on that.

25 MS. GRADY: I can say one thing further:



1 As you know this regulation is in place for hydrogen  
2 and oxygen monitoring as there's been a severe  
3 accident. There's been a lot of hydrogen produced in  
4 the containment. A lot is produced when the cladding  
5 oxidizes in a relatively short period of time. No  
6 matter what scenario, it's all taken place before.  
7 Let's a 48-hours. Okay?

8 But radiolysis continue for days. So we  
9 really have two times when knowing how much hydrogen  
10 and oxygen in containment is available is early on in  
11 the first setting two hours and they covered that. As  
12 long as they establish hydrogen and oxygen monitoring  
13 within 72 hours, the containment can withstand  
14 whatever combustion accident takes place in the  
15 containment.

16 Afterwards they need to establish and  
17 evaluate and find out what kind of conditions they  
18 have, whether they would support combustion. And they  
19 wouldn't need to find out -- they've done an analysis,  
20 we've done a confirmatory calculation that says it's  
21 not for at least 25 days or 45 days. So they got a  
22 lot of time in between the first large quantity of  
23 hydrogen being produced and the first challenge to the  
24 containment. And then the significant hydrogen having  
25 been produced from the radiolysis.

1           So in my view if somebody were asking me  
2           as a reviewer could they re-isolate for a while  
3           safely, as long as they've gotten past the first  
4           amount safely, I would say sure if you need it for  
5           other reasons. That's a practical thing. That's not  
6           what the regulation says. But that's practical.

7           So we have a near-term issue that's  
8           covered by design, the 72 hours. They've got a system  
9           that's going to tell them what's going on whenever  
10          they want to check and they have to establish it  
11          initially to find out what's going on. And then they  
12          could check periodically. It could be continual  
13          rather than continuous, in my opinion.

14          MR. STUTZCAGE: Okay. Next slide, please.  
15          And this is getting -- this is just to discuss a  
16          little bit about Reg Guide 1.183, which as we  
17          indicated specifies that ESF system leakage should be  
18          considered in the off-site and the main control room  
19          dose analysis, but the Reg Guide specifies that, as  
20          NuScale indicated, I think, containment purge need not  
21          be considered if the purge is only -- would only occur  
22          following a severe accident.

23          As he kind of stated, we feel that it's a  
24          little bit different than containment purge because  
25          this is -- the capability to perform hydrogen and

1 oxygen monitoring is required and it is to assess the  
2 conditions where the containment purge is -- you just  
3 determine that it's worth venting to ensure  
4 containment integrity.

5 Next slide, please. And this just gets  
6 into some of the differences of the ESBWR. And we  
7 picked ESBWR here because it was in one of the RAI  
8 responses. And they discussed most of this already.  
9 So I think you have the three systems. They're not  
10 redundant, not safety-related, Seismic Category 3  
11 system, and the system isolation valves cannot be  
12 controlled from main control room.

13 Next slide, please. And then this is more  
14 of what we -- I think we already talked about, about  
15 the containment size being different and the ratio,  
16 the power ratio, and that creating potentially a more  
17 concentrated source.

18 We will say for the APR1400, the 850 times  
19 smaller, I wouldn't necessarily reference that number.  
20 It depends if you're talking about containment free --  
21 the free volume or the total volume, but it's -- the  
22 idea is it's hundreds of times larger. The power  
23 level is only 25 times less.

24 And then the last bullet there on the  
25 piping size. And again these systems have piping from

1 four inches to three-eighth-inch. And there's also  
2 isolation valves to a variety of other systems that  
3 could create potential weak paths like the ventilation  
4 system, the rad waste system and a few others.

5 DR. CORRADINI: So I don't want to design  
6 it here, but I'm confused. Is the sampling system  
7 small?

8 MR. STUTZCAGE: The sampling system is  
9 small.

10 DR. CORRADINI: It's just far away from  
11 the isolation valves such that you have a lot of pipe  
12 volume that could be leaking? Is that the issue?

13 MR. STUTZCAGE: Yes, to take -- to get the  
14 sample in the NuScale design they have to -- it has to  
15 be routed through the CES system. So that's the --  
16 that's where the four-inch pipe is. And then it  
17 narrows down to --

18 DR. CORRADINI: What system?

19 MR. STUTZCAGE: The containment evacuation  
20 system.

21 MR. LAVERA: Ron LaVera. I'm another  
22 radiation protection reviewer. I worked with Ed on  
23 this.

24 So there are a couple of things: One, the  
25 piping size of the CES, containment evacuation system,

1 is four inch. That goes off for a distance until it  
2 gets to the sample system piping which necks down.  
3 Then it comes back to containment flood and drain  
4 system which goes up to I think it's like about an  
5 inch or three-quarters of an inch.

6 So there's another aspect that's not  
7 talked about in the slides in that the containment  
8 evacuation system operates at a hard vacuum during  
9 normal operation. So that's thing number one.

10 Thing number two is downstream where the  
11 sample system normally exhausts to is at very near  
12 atmospheric pressure because it's right before a set  
13 of ventilation ducts. And you're only going to have  
14 inches of water -- DP across the ventilation filter,  
15 so you're not going to have a high filter there. The  
16 containment flood and drain system is not even  
17 normally used. It's only used during containment  
18 refueling.

19 The other aspect of this is we've talked  
20 about this as if it's a severe leakage. When we're  
21 talking severe leakage, the numbers that we use for  
22 our in-house analysis was on the order of 0.3 cfm  
23 causing a problem with the control room dose and  
24 potentially the site boundary dose, and we didn't even  
25 get a chance to take that number and analyze it for

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1 this confined space where the leakage would be  
2 occurring that an individual would have to go into and  
3 presumably at an isolation button. So those are some  
4 nuances on Ed's presentation.

5 CHAIR KIRCHNER: For clarification we're  
6 not in the design business here, but is there -- is  
7 the isolation on -- there's got to be an isolation on  
8 the four-inch line, but does the three-eighths-of-an-  
9 inch line have its own isolation? And typically when  
10 you do that you would have two valves, maybe one  
11 solenoid and one manual?

12 MR. STUTZCAGE: Ron's up there again.

13 MR. LAVERA: Say that again, please?

14 CHAIR KIRCHNER: Is there a separate  
15 isolation system on the three-eighths-of-an-inch  
16 monitoring line? Typically there would be.

17 MR. LAVERA: So you're talking about once  
18 it gets to the sample system --

19 CHAIR KIRCHNER: Yes, for the sample  
20 system. So you're going to take a take-off of a  
21 four-inch pipe, which is the containment evacuation  
22 system line. That has its own isolation. But then  
23 does the three-eighth-inch cap-off -- does that have  
24 its own isolation?

25 MR. LAVERA: So if you did have its own

1 isolation; and I can't remember -- I think it does.

2 CHAIR KIRCHNER: I can't imagine a  
3 plumbing system on a nuclear power plant where you  
4 have a main line four inches and you have a little  
5 tap-off not having its own isolation valves.

6 (Simultaneous speaking.)

7 MR. LAVERA: And I believe you're correct,  
8 but it's not necessarily just leakage from the sample  
9 system that you're concerned about. It's the whole  
10 system. So the containment is going to be at; correct  
11 me if I'm wrong, 60, 70 psi. The four-inch line is  
12 designed to operate at a 29, 28-inch vacuum. The  
13 containment flood and drain system is normally just  
14 pretty close to atmospheric pressure, maybe 10 or 20  
15 pounds when you're pumping back to it when it's in  
16 service. Those isolation valves are going to be open,  
17 both on the CES system and containment flood and drain  
18 system. So all that piping is going to be pressurized  
19 to whatever the containment pressure is at.

20 CHAIR KIRCHNER: Yes. Seems like a design  
21 problem to me. It's solvable.

22 MEMBER MARCH-LEUBA: And all the piping is  
23 non-safety-related.

24 CHAIR KIRCHNER: Well, not -- up to the  
25 isolation valves it's safety-related.

1 MEMBER MARCH-LEUBA: No, once you open the  
2 source above, you connect non-safety-related  
3 components to the containment 30605/60847A.

4 CHAIR KIRCHNER: Yes, that's all B2  
5 piping.

6 MEMBER MARCH-LEUBA: So as far as I'm  
7 concerned you can go to Home Depot and buy some PVC  
8 pipes and put it in there, right? They won't. I  
9 would.

10 MR. STUTZCAGE: All right. Next slide.  
11 And then this slide is just talking about what we  
12 already discussed, performing the dose -- we did a  
13 scoping calculation. And then the inability to  
14 potential -- we don't know if they have the capability  
15 to re-isolate the systems.

16 So next slide. Okay. And then this is  
17 discussing what we already discussed for the path  
18 forward. Since the staff does not have assurance that  
19 the MCR and off-site dose limits will be met, the  
20 staff intends to specify a NuScale design  
21 certification rule that the COL applicant will  
22 demonstrate that the leakage will not result in  
23 exceeding main control room or off-site dose limits  
24 and/or include design features to provide assurance  
25 that the leakage will not result in exceeding the

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1 limits.

2 And then if manual actions are required,  
3 we'd expect to see that they can do it within dose  
4 limits. And that's under -- we view that as being  
5 under the 10 CFR 50.34(f)(2)(vii) requirement.

6 DR. CORRADINI: And there's not been an  
7 analysis that says the leakage rate -- if the leakage  
8 rate is greater than whatever, you've exceeded the  
9 five rem dose limit?

10 MR. STUTZCAGE: No.

11 DR. CORRADINI: Although staff has done  
12 some quick and dirty calculations?

13 MS. HART: Yes, I did some very scoping  
14 calculations. I used the information -- they referred  
15 to an ANSI standard for those type of systems, and it  
16 gives a range of flow rates through the systems. That  
17 could be one liter per minute through 100 liters per  
18 minute. And if I assume like 10 percent of that is  
19 leaking to the outside environment based on no good  
20 information really, at the lower end of the range they  
21 may still be okay. They'd be very close to the  
22 control room dose limits, but at the higher end of the  
23 range they would exceed both at the EAB and in the  
24 control room. And it's not changing that release  
25 rate over time or anything like that. It was a very

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1 rough calculation. And we did discuss this with the  
2 applicant.

3 MEMBER MARCH-LEUBA: And, Michelle, since  
4 I have you there, I'm thinking here that they should  
5 not un-isolate that system because it's all  
6 non-safety-related valves and filters and things that  
7 may or may not work. And you don't want to be dumping  
8 all that severe accident isotopes into that four-inch  
9 valve.

10 So what do we gain by monitoring the  
11 hydrogen there? I mean what -- I mean are we -- is  
12 the solution -- this -- I guess this is probably the  
13 argument that NuScale is doing. Is that -- the  
14 solution is worse than the problem here.

15 MS. HART: So there has been discussions  
16 along those lines. The point that they make is that  
17 they have the capability to do this monitoring if  
18 necessary, however, they will decide at the time if  
19 it's necessary to do the monitoring. However, our  
20 regulation says that you shall have this capability  
21 for continuous monitoring of hydrogen and oxygen. And  
22 that's to protect against massive containment failure  
23 so that you understand what your -- if you're reaching  
24 detonation limits or anything like that.

25 MEMBER MARCH-LEUBA: If you really want to

1 protect you put -- dedicate to the line that is tiny  
2 and this thick so that it won't break and it's -- you  
3 will have to dump it through the four-inch valve with  
4 all those cooling pumps.

5 DR. CORRADINI: Yes, I didn't want to --  
6 I think Jose's going in the same direction I was  
7 thinking, which is I thought there were -- and again  
8 I can't remember the chapter for the meeting we were  
9 at, but I thought there were some fairly conservative  
10 calculations about hydrogen combustion that they will  
11 have no issue here.

12 MS. GRADY: And before 72 hours that's  
13 absolutely correct.

14 DR. CORRADINI: Okay.

15 MS. GRADY: Absolutely. And at that point  
16 it's time to take -- it's time to sample the  
17 atmosphere in a containment to find out what's really  
18 going on, because post-accident there are only two  
19 bits of information we have: this hydrogen and oxygen  
20 monitoring system which will be telling you what's in  
21 containment. And the second thing is the radiation  
22 monitor under the bio-shield. There's no other  
23 indication. Proposed accidents conditions. There's  
24 no -- because there's a sampling exemption request,  
25 they're not going to be sampling any other way in the

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1       containment.

2                   MEMBER MARCH-LEUBA:       I remember you  
3       telling us --

4                   MS. GRADY:     It's an online monitoring  
5       system.

6                   MEMBER MARCH-LEUBA:       I remember you  
7       telling us that there is no -- or mechanism to have  
8       oxygen in containment. I mean we can -- I can tell  
9       you how the hydrogen gets mixed in, but oxygen?

10                  MS. GRADY:   Radiolysis. That's why long  
11       term there's still a concern. Even though in the  
12       first 72 hours there really -- the design will  
13       accommodate whatever conditions are there.

14                  MEMBER MARCH-LEUBA: You've calculated how  
15       much radiolysis you get?

16                  MS. GRADY:   Pardon? Yes, it's a lot that  
17       you get.

18                  MEMBER MARCH-LEUBA:   Okay. Fine.

19                  MS. GRADY:   Like I said, between 25 and 46  
20       days you have enough to support combustion.

21                  MEMBER MARCH-LEUBA:   Yes, you're talking  
22       months.

23                  MS. GRADY:   Yes.

24                  MR. STUTZCAGE:   Okay. That's all I have  
25       on the hydrogen and oxygen monitoring, if there are

1 any more questions on that.

2 No? Next slide, please. Okay. Now to  
3 the sampling exemption. And this is an exemption from  
4 50.34(f)(2)(viii), which requires the ability to  
5 obtain and analyze post-accident samples. And the  
6 regulation specifies certain radionuclides that are  
7 indicated: the degree of core damage, hydrogen in the  
8 containment atmosphere, dissolved gases, chloride and  
9 boron concentration. And when we say sampling, we're  
10 envisioning physically grabbing the sample and taking  
11 it to a lab to analyze it versus monitoring, which is  
12 the actual in-line monitor to provide data. So  
13 NuScale requested they be exempt from this  
14 requirement. And the next several slides I discuss  
15 the -- our evaluation of that.

16 Next slide, please. Okay. So the first  
17 is radionuclides. The regulation specifies that it's  
18 -- the purpose of sampling is to assess the degree of  
19 core damage. As discussed, NuScale has the -- under  
20 the bio-shield radiation monitors and in the core exit  
21 thermocouples which can be used to assess the degree  
22 of core damage. Based on this we found that it is  
23 acceptable to exempt from sampling the radionuclides.

24 And the last bullet here is that we don't  
25 view this exemption to be an exemption from

1 50.47(b)(9) which requires emergency preparedness  
2 ability.

3 Next slide, please. Containment hydrogen.  
4 Purpose of sampling containment hydrogen in the  
5 atmosphere is to ensure that hydrogen and oxygen  
6 concentrations do not support combustion that could  
7 challenge the containment. As we discussed, they have  
8 the ability to monitor for hydrogen and oxygen, so we  
9 found that there wasn't a need to have the ability to  
10 also sample for it.

11 Next slide, please. Dissolved gases.  
12 They're required to assure that natural circulation is  
13 not inhibited. The staff agrees that the -- that it's  
14 okay to be exempt from sampling the dissolved gases  
15 because the NuScale design -- because of the  
16 NuScale-specific considerations of the ECCS system.  
17 That's documented in the staff's section 5.4.5 of the  
18 SER.

19 Next slide, please. Chlorides. NuScale  
20 requests an exemption from the sampling of chlorides  
21 because they're -- chlorides are typically introduced  
22 from external events. The staff finds it acceptable  
23 because NuScale's minimal use of chlorinated cable  
24 insulation monitoring chloride concentration and  
25 make-up inventory sources during normal operations and

1 NuScale limits chlorides by monitoring and control of  
2 reactor water chemistry.

3 Next slide. Last is boron. Purpose of  
4 the sampling boron is to ensure that there's adequate  
5 shutdown margin and to maintain safe shutdown. The  
6 staff agrees with the exemption because only Type B  
7 variables -- the only Type B variables identified in  
8 a NuScale design that provide direct indication and  
9 are used to assess the process of accomplishing or  
10 maintaining reactivity control are neutron flux and  
11 core inlet and exit temperature. The transient and  
12 accident analysis described in 22 Chapter 15 does not  
13 rely on a measurement of RCS boron concentration and is  
14 not expected to be necessary to implement the plant  
15 operating procedures and maintain plant critical  
16 safety functions.

17 Next slide. And since we found that they  
18 could get this information by other means or it was  
19 not necessary, we found the exemption request to be  
20 acceptable and we didn't assess the radiological  
21 consequences to a worker obtaining and analyzing  
22 samples.

23 And that's it for the sampling exemption,  
24 if there are any questions on that.

25 Okay. Next slide, please. Last thing I

1 have is a brief discussion on design- basis failed  
2 fuel fraction. So I think where we were following the  
3 last ACRS meeting, we indicated that the applicant had  
4 changed the design-basis RCS to the 0.066 percent, and  
5 also the realistic to the 0.0 -- 0 -- to 0.0066. And  
6 we just had to -- NuScale had a variety of changes  
7 that they had to make to the FSAR because if the RCS  
8 activity changes, then all the other sources in the  
9 plant change and so forth. And we reviewed all of  
10 those changes and all the downstream effects  
11 throughout Chapter 11 and 12 and found that everything  
12 was completed acceptability.

13 So that concludes my presentation. I  
14 guess I'll turn it over to Anne-Marie Grady to discuss  
15 survivability.

16 MS. GRADY: Equipment survivability.  
17 Following a severe accident the two functions that  
18 have to be maintained are the containment integrity  
19 and post-accident monitoring to provide information on  
20 the severe accident conditions in containment. To  
21 meet the criteria of SECY 90-016 the equipment  
22 required to mitigate severe accidents has to be  
23 identified. The function of the equipment, the time  
24 duration if it's needed and the environmental  
25 conditions of pressure, temperature, humidity and



1 radiological dose for which this function is required  
2 also have to be identified. Environmental conditions  
3 must also, per 50.44(c)(3), include exposure to  
4 conditions created by hydrogen burning.

5 Next slide, please. And NuScale's  
6 methodology for assuring equipment survivability in  
7 terms of radiological dose; and that's all I'm going  
8 to talk about here, involves comparing the severe  
9 accident core damage source term that's been talked  
10 about earlier today with the equipment qualification,  
11 EQ, design-basis source term. If the EQ dose is  
12 larger, no further quantitative survivability  
13 assessment on dose is required. If the severe  
14 accident dose is larger, qualitative assessments,  
15 testing and/or additional analyses may be needed to  
16 perform -- to ensure a survivability. NuScale has  
17 evaluated both the EQ and the severe accident doses  
18 for each component or variable that they have  
19 identified in FSAR table 19.2-11.

20 Now 19.2-11 describes all of the equipment  
21 and the post-accident monitoring variables that are  
22 required per equipment survivability. They have been  
23 identified to us in an RAI response with a mark-up,  
24 which means that in Rev 4 of the DCA we will be able  
25 to see that table, or I can give you a reference for

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1 an RAI response and you can see the table immediately.  
2 The results of the evaluation including that table can  
3 be found in the non-publicly-available version of the  
4 audit report, and there would be a number for the  
5 audit report.

6 Next slide. Specifically identified are  
7 components in their evaluation whose severe accident  
8 dose exceeds or potentially could exceed the EQ dose,  
9 and those components are the electrical penetration  
10 assemblies, the ECCS reset valves, the containment  
11 evacuation system and the containment flooding and  
12 drain system containment isolation valves, the  
13 combustible gas monitors, the containment gas sample  
14 pump and the inside the bio-shield radiation monitor.

15 Now this is important: At the design  
16 certificate state specific components have not yet  
17 been selected. Once selected the COL applicant will  
18 confirm or update the EQ and the severe accident doses  
19 for all of the components identified in table 19.2-11  
20 according to COL Item 19.1-8.

21 Next slide. Oh, okay. One more thing.  
22 So basically in my SE, which you have a copy of, I  
23 have made a finding. I agree with the scope of their  
24 equipment survivability components and post-accident  
25 monitoring variables. I agreed with the durations,

1 that they're needed. I have agreed that the functions  
2 that they provide. However, what has not yet been  
3 done per the note that's down there is that the doses,  
4 both EQ and equipment survivability, have to be  
5 finalized. And the COL applicant will do that.

6 Do you have any questions?

7 Okay.

8 MS. HART: Okay. So to go to our final  
9 topic for our presentation I will describe the SER  
10 section for radiological consequences of accidents and  
11 the core damage event.

12 This dose analysis, they're mostly  
13 described in Chapter 15.0.3 of their FSAR with some  
14 additional information scattered throughout the rest  
15 of Chapter 15 to support those dose analyses plus the  
16 core damage event as described in 15.10 of the FSAR.

17 The radiological consequence analyses  
18 implemented the topical report methodology that we  
19 discussed this morning. We did have some overlap in  
20 the review of the topical report and the FSAR, so we  
21 did use some of the FSAR information to help us  
22 evaluate the methodology. You know, look at how it  
23 was implemented.

24 Mostly what is in the FSAR is they use the  
25 specific design information as input to the topical

1 report methodology and they also included hypothetical  
2 atmospheric dispersion factors which were evaluated  
3 and discussed in SER Section 3.2.4.

4 Next slide, please. So the accidents and  
5 events analyzed are those same ones that we talked  
6 about this morning, the -- except for -- next slide,  
7 please. Well, I'll tell you about the regulatory  
8 criteria first.

9 The regulatory criteria for the off-site  
10 dose of course are the 52.47(a)(2)(iv), 25 rem, any  
11 two-hour period at the exclusionary boundary. And at  
12 the low-population zone, the outer boundary, it's 25  
13 rem TEDE for the duration of the passage of the plume.  
14 DSRS, the design-specific review standard, for NuScale  
15 15.0.3 does provide the accident-specific dose  
16 criteria. This is based on historical -- well, first  
17 off it was taken from Reg Guide 1.183, and that was  
18 also based on historical practice. That was in the  
19 SRP for large light water reactors, that if you have  
20 a more likely event or more likely accident, the dose  
21 criterion is lower.

22 And so for example, for the fuel handling  
23 accident the original Reg Guide for that, Reg Guide  
24 1.25, and the SRP for that, 15.7.4, talk about the  
25 dose criterion for that should be that the doses

1 should be well within the Part 100 limits. At that  
2 time it was Part 100. And so that would be 25 percent  
3 of the 25 rem criterion. And so we have carried that  
4 forward for light water reactors through the Reg Guide  
5 1.183, and NuScale did adopt the same criteria.

6 As another example for the steam generator  
7 tube rupture the more likely event is that you would  
8 be at the equilibrium coolant activity and have the  
9 co-incident iodine spike. And the less likely event  
10 is that you would be at the higher maximum  
11 pre-incident iodine spike technical specification  
12 rate. And so for the normal evaluation, which is more  
13 likely, you would have the -- a -- you would be  
14 required to have your doses be a small fraction of the  
15 25 rem, so that would be 2.5 rem. And then for the  
16 pre-incident spike it would be the full 25 rem, if  
17 that makes some sense. You know, it's more likely  
18 that you would have something so that you would have  
19 the doses be reduced.

20 The PDC that NuScale has asked for an  
21 exemption from GDC 19, in all effect for the dose  
22 requirements it is the same as GDC 19, our principal  
23 design criterion. There's a discussion in another  
24 section of the FSAR in Chapter 1 that describes our  
25 acceptance of their exemption from the principal -- to

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1 design a Principal Design Criterion 19 for control  
2 room. The control room habitability requirements  
3 however are exactly the same as in GDC 19.

4 Underneath that I do refer to the TMI  
5 item, which is 3D34, the five rem TEDE is also  
6 referenced by that. So the five rem TEDE is a dose  
7 criterion for the control room itself.

8 The technical support center habitability  
9 also is five rem, and you get to that through  
10 NUREG-0696, which says that you should have the same  
11 radiological habitability as the control room. Five  
12 rem is not in the regulation for technical support  
13 center habitability.

14 Next slide, please. So here we go through  
15 the dose acceptance criteria. And you see the  
16 different dose acceptance criteria.

17 Next slide, please. So as we had  
18 discussed in the Chapter 6 ACRS Subcommittees, there  
19 are two ventilation systems for the control room, and  
20 so the dose analyses do model operation of both modes  
21 of operation of the control room ventilation system.  
22 So for control room monitoring in addition to intake  
23 and in-leakage to the control room envelope through  
24 the ventilation systems are through unfiltered  
25 in-leakage. We also evaluate direct dose from shine,

1 sky shine and shine from filters.

2 The two-ventilation system operation  
3 cases, as I described this morning in the topical  
4 report methodology, are operation of the control room  
5 ventilation system and a supplemental filtration mode  
6 for the duration of the event, and those have a 99  
7 percent efficient HEPA and charcoal filters assumed.  
8 Or the second case is operation of the control room  
9 habitability system, which is the bottled air system  
10 providing clean air for initial 72 hours. And then  
11 you have the control room ventilation system and  
12 supplemental filtration mode with those 99 percent  
13 efficient filters for the remainder of the duration of  
14 the event.

15 And I've listed the control room  
16 unfiltered in-leakage assumption values. These are  
17 part of the -- they will be tested. The control room  
18 ventilation system during operation with the  
19 filtration in line is 174 cfm through the envelope  
20 plus 5 cfm for opening and closing the door. And  
21 during the bottled air system operation, they said  
22 it's effectively zero, but they're taking an  
23 assumption of 10 cfm through the envelope with 5 cfm  
24 through ingress and egress.

25 Next slide, please. As --

1 CHAIR KIRCHNER: And those numbers,  
2 Michelle, are just backed by precedent? There's no  
3 regulatory basis for those?

4 MS. HART: Which numbers?

5 CHAIR KIRCHNER: The 174 plus 5 and 10.

6 MS. HART: Those are their design numbers.

7 CHAIR KIRCHNER: Yes.

8 MS. HART: Those are the number that --

9 CHAIR KIRCHNER: Yes.

10 MS. HART: -- they are supposed to be able  
11 to assure that they can accomplish.

12 CHAIR KIRCHNER: And for your purposes,  
13 from a regulator standpoint, you either check these  
14 through benchmark analyses or such, and you're  
15 confident they will then stay within the five rem  
16 limit? Is that --

17 MS. HART: Yes, if -- these are the design  
18 values. If you assume these design values, it shows  
19 that you may stay within five rem.

20 DR. SCHULTZ: But there's going to be  
21 pre-operational testing of the leakage?

22 MS. HART: There are ITAAC, there are COL  
23 items and there's a testing protocol.

24 DR. SCHULTZ: Yes, thank you.

25 MS. HART: For the technical support



1 center they do include a technical support center in  
2 their design. The technical support center is part of  
3 the control room envelope and is served by the normal  
4 ventilation system, but it is not served by the  
5 bottled air system. If you lose power or for some  
6 reason the filtration system for the normal  
7 ventilation system cannot operate, the TSC function  
8 can be moved if the TSC is found to be uninhabitable.

9 The operation of the CRVS in supplemental  
10 filtration mode is assumed for the duration of the  
11 event for the technical support center with those 99  
12 percent efficient intake filters, and the TSC  
13 unfiltered in-leakage is 56 cfm through the envelope  
14 and 10 cfm ingress/egress because they do not have an  
15 airlock door.

16 DR. SCHULTZ: What does that mean that it  
17 can be moved if it's uninhabitable? There's a  
18 different location that's further away from the  
19 facility?

20 MS. HART: Yes, and that is not explicitly  
21 defined at this point. It would be up to the COL  
22 applicant --

23 DR. SCHULTZ: Okay.

24 MS. HART: -- to determine where that  
25 function would move.

1 DR. SCHULTZ: Yes, thanks.

2 MS. HART: And there are certain functions  
3 that may have to move to the control room. It's  
4 similar to what we had done for the AP1000.

5 DR. SCHULTZ: Yes.

6 MS. HART: It's a similar situation.

7 DR. SCHULTZ: Right. Thank you.

8 MS. HART: Next slide, please. So we go  
9 to the core inventory. We did take a look at their  
10 core inventory that they developed as the bases for  
11 all these analyses. It's based on 102 percent of the  
12 rated core thermal power. It used the NuScale fuel  
13 design. They calculated the core inventory using the  
14 scale code. They used the TRITON module for  
15 burnup-dependent cross-sections. They developed their  
16 own. And they used ORIGEN-ARP and ORIGEN-S as a  
17 stand-alone for generation and depletion calculations.

18 I did perform some confirmatory  
19 calculations using the ORIGEN-ARP in the previous  
20 version of Scale. I modeled it using the default PWR  
21 17 by 17 fuel, using the design information in the  
22 FSAR on enrichment, maximum burnup cycle length,  
23 amount of uranium, such things. I did get similar  
24 results as the FSAR table, and so therefore I thought  
25 it was consistent with the topical report methodology

1 and the guidance in Reg Guide 1.183, which does  
2 identify ORIGIN as an appropriate code.

3 Next slide, please.

4 MEMBER BLEY: Can I ask you a question  
5 about that? The same thing about moving the TSC. It  
6 just came up in a design cert for somebody else where  
7 an alternative would be to move it to the control  
8 room, but the TSC came about after the Three Mile  
9 Island accident as a place to get people out of the  
10 control room so that we wouldn't interfere with  
11 operations. Moving it back into the control room just  
12 defeats the original purpose. Now it's evolved into  
13 where you run the emergency response guidelines, too,  
14 but it was really to --

15 MS. HART: So --

16 MEMBER BLEY: -- free up the operators.

17 MS. HART: -- not knowing the exact design  
18 you're talking about, but knowing in general terms,  
19 there are some functions that may have to go to the  
20 control room like immediate response help for the  
21 operators if there are actions that may need to be  
22 taken, however other functions of the TSC would move  
23 to another off-site facility like the off-site -- the  
24 OSC or some place like that.

25 MEMBER BLEY: Yes. That makes a lot more

1 sense, yes.

2 MS. HART: So, yes, there are just some  
3 functions that are important that may have to go to  
4 the control room. I don't know that that's the case  
5 for NuScale, but I know that that was the case for  
6 AP1000.

7 DR. SCHULTZ: You could make the argument  
8 that there's time to do that here.

9 MS. HART: There are all kinds of  
10 requirements on incident response or accident response  
11 of facilities.

12 So to go to the primary coolant activity  
13 concentration, we did evaluate that. I did that in  
14 concert with the Chapter 11 reviewer. The FSAR table  
15 11.1-4 primary coolant design source term was adjusted  
16 to the design-specific activity limits to be used as  
17 the accident initial conditions. Those technical  
18 specifications, the general technical specifications,  
19 specific activity limits are given on this slide.  
20 They are as you can see much lower than you are  
21 typically seeing in a PWR.

22 And we've had some discussions about  
23 design-basis failed fuel fractions, so I won't belabor  
24 that point, however, I will say that taking the  
25 technical specification specific activity limits and

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1 using those as the initial conditions for your  
2 accidents where you're moving primary coolant is  
3 consistent with the guidance in Reg Guide 1.183. And  
4 also the way that they used it in the FSAR is  
5 consistent with our topical report methodology, which  
6 also, to remind you, the secondary coolant activity  
7 concentration was not modeled.

8 Next slide, please. So these are the  
9 events that they did -- analyzed that were accidents  
10 without core damage. The FSAR dose analyses followed  
11 the topical report methodology for these events and  
12 the design-specific input values were taken from  
13 relevant FSAR system descriptions and they used the  
14 design-specific core inventory or the primary coolant  
15 activity concentration as required by the specific  
16 analysis. And the dose results calculated by NuScale  
17 do meet the accident-specific dose criteria with quite  
18 a bit of margin.

19 Next slide, please. As they had tried to  
20 describe earlier, the rod ejection accident, the  
21 methodology in the topical report assumes a certain  
22 amount of core failure. Their Chapter 15 analyses did  
23 not predict any core -- or fuel damage for the  
24 limiting rod ejection accident, so therefore they did  
25 not provide a dose analysis. And that's consistent

1 with the topical report methodology which does  
2 describe that, and it also is consistent with our  
3 discussion in Reg Guide 1.183. It also is true that  
4 the rod ejection accident radiological consequences  
5 would be bounded by other events because you have the  
6 same release pathways and the same amount of primary  
7 coolant activity that would be potential to be  
8 released.

9 Next slide, please. For the core damage  
10 event they developed their core damage source term by  
11 implementing the topical report methodology using  
12 design-specific information. The core damage source  
13 term scenario selection was based on the NuScale ERA  
14 internal events and it had five surrogate scenarios  
15 with various failures of the ECCS and the decay heat  
16 removal system was available, and intact containment  
17 was part of those scenarios.

18 They used MELCOR to estimate the release  
19 timing and magnitude for each scenario and they  
20 followed the topical report methodology to  
21 characterize the core damage source term in that they  
22 picked the release onset and duration from the  
23 scenario with minimum time to core damage. And they  
24 took the core release fractions as the median from the  
25 scenarios.

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1 Next slide, please.

2 MEMBER REMPE: Are we going to have a  
3 closed session later for sure? If we have a question  
4 about the proprietary MELCOR confirmatory analysis by  
5 RES? It's kind of a down-in-the-weeds question, but  
6 if we're going to have later time, I'll hold off and  
7 do it. But if we're going to say at the end does  
8 anyone have any question -- I can try and say it in an  
9 open way and let's see where it goes.

10 PARTICIPANT: Why don't you try that?

11 MEMBER REMPE: Okay. In the staff audit  
12 they talk about that the topical report from the  
13 applicant assumed that the release durations into the  
14 containment are estimated as the time when X percent  
15 of the volatile fission products reach the containment  
16 by the end of the accident, but it didn't say whether  
17 it was iodine or cesium. So the staff basically said,  
18 well, we'll take this percentage and they came up with  
19 two numbers and one of them was quite low and then the  
20 other one was quite high.

21 And I -- is there something you can kind  
22 of tell me a little bit more about this, because I was  
23 having trouble figuring out what happened here when I  
24 was reading the table? Does that ring a bell? And  
25 I'm talking about table 5.1.

1 MS. HART: As far as like how we  
2 determined the timing or how we determined --

3 MEMBER REMPE: Yes, just kind of talk a  
4 little bit about what you did and maybe then I'll say,  
5 oh, yes, okay, it makes sense on why you did this,  
6 because I wasn't sure why you picked the two upper and  
7 lower bounds for the values that you did for the  
8 duration.

9 MS. HART: Is there something that we can  
10 provide in open session?

11 MR. SCHAPEROW: Joy, could you clarify our  
12 question about duration and the staff's calculations?

13 MEMBER REMPE: Okay. So I'm looking at  
14 Section 5.2 of this report and basically --

15 MR. SCHAPEROW: What report. Excuse me.

16 MEMBER REMPE: The MELCOR confirmatory  
17 analysis that RES completed.

18 MR. SCHAPEROW: So we did a subsequent  
19 report providing MELCOR fission prior releases to the  
20 environment and RADTRAD dose analysis of that.

21 MEMBER REMPE: This is table 5.1. We're  
22 just talking about releases into the containment. And  
23 there was some -- I guess some uncertainty the staff  
24 had about what was going on and what the applicant had  
25 submitted. And so they basically -- the applicant



1 said, well, we took a time when X percent of all  
2 fission products were released, but apparently the  
3 staff couldn't figure out whether they were talking  
4 about cesium or iodine. Is this starting to ring a  
5 bell?

6 MR. SCHAPEROW: I think you're referring  
7 to an earlier staff report that was done last year.

8 MEMBER REMPE: Again, this is the only one  
9 that we were given to prepare for this meeting. Does  
10 that mean that the report we've been reviewing is not  
11 the right one?

12 MR. SCHAPEROW: So --

13 MEMBER REMPE: I'm looking at the name of  
14 the report. It's FSCB19-01. And where I was now I've  
15 lost the page. But am I looking at the wrong version  
16 here? April 2019?

17 MR. SCHAPEROW: That report was done and  
18 we actually discussed it at a previous ACRS meeting.  
19 And what we've done since that time was we took the  
20 MELCOR predictions from that report, from those  
21 calculations and we put it into RADTRAD to calculate  
22 doses off-site to compare back to the applicants. So  
23 I'm not sure what time --

24 MEMBER REMPE: So did you not use this  
25 result then to do that?

1 MR. SCHAPEROW: We're using the  
2 radiological releases out of the containment from that  
3 report. The report doesn't actually describe it --  
4 those releases.

5 MEMBER REMPE: Yes.

6 MR. SCHAPEROW: Those releases are  
7 described in a subsequent report that we did that I  
8 think are referenced in the SER that was sent to the  
9 Committee a month ago.

10 MEMBER REMPE: And I did have the SER, but  
11 they referenced this. And so this was what we were  
12 given to look at, so I looked at it. And, yes --

13 MR. SCHAPEROW: The reference should be to  
14 another report, but again it's the same MELCOR  
15 calculation.

16 MEMBER REMPE: Is there something else  
17 that I missed?

18 MR. CAMPBELL: Have we provided to them --

19 MR. SCHAPEROW: Yeah, of course.

20 (Simultaneous speaking.)

21 MEMBER REMPE: That would be good. Okay.

22 MR. SCHAPEROW: Yes.

23 MEMBER REMPE: I'll look at it, and I'll  
24 come back if I have questions.

25 (Simultaneous speaking.)

1 MR. CAMPBELL: -- this is Andy Campbell.

2 MEMBER REMPE: Okay. Thank you. Sorry to  
3 interrupt.

4 MR. SNODDERLY: This is Mike Snodderly.  
5 You also might want to go back to the SER. That's the  
6 report that's referenced in the SER.

7 MR. TESFAYE: We'll take a look at it.

8 MR. SNODDERLY: Thanks.

9 MS. HART: Okay. Oh, yes. So next slide,  
10 please. So the iodine modeling in the core damage  
11 event, they did use the Reg Guide 1.183 iodine  
12 chemical forms as was put in their topical report.  
13 And they did some calculations using their  
14 temperature-dependent DH calculation code. They  
15 modeled it over a 30-day period and they did find that  
16 the post-accident pH inside the containment was  
17 between six and seven and the iodine -- given that the  
18 iodine re-evolution was assumed negligible consistent  
19 with the topical report methodology. There's no  
20 credit for elemental iodine removal within the  
21 containment and the staff did find these particular  
22 aspects to be consistent with the topical report  
23 methodology.

24 Next slide, please. For the aerosol  
25 modeling in the containment they did model aerosol

1 natural deposition in containment. They provided  
2 time-dependent aerosol removal rates calculated using  
3 the STARNAUA code implementing their topical report  
4 methodology. And you saw in their slides earlier the  
5 actual results of that. Those are the same results  
6 that are in the FSAR.

7 The calculation did use design-specific  
8 input. They used the thermal hydraulic conditions  
9 calculated by MELCOR for the surrogate scenario with  
10 the minimum time to core damage and the staff found  
11 implementation of the topical report aerosol natural  
12 deposition methodology acceptable. We also evaluated  
13 -- you know, when we had evaluated that as part of the  
14 topical report with the integrated methodology to  
15 satisfy our concerns, we didn't have any additional  
16 concerns based on the information in the FSAR.

17 Next slide, please. For the containment  
18 leakage pathway the release from the core was assumed  
19 to be well-mixed within the containment air volume and  
20 the design-basis containment leakage rates were taken  
21 from the FSAR technical specification testing that  
22 they're going to do. And so that's 0.2 percent per  
23 day for the first 24 hours of release and 0.1 percent  
24 per day for the remainder of the 30 days, which is  
25 consistent with the topical report methodology and

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1 also consistent with the guidance in Reg Guide 1.183.

2 Next slide, please. We talked about the  
3 combustible gas monitoring, the hydrogen and oxygen  
4 monitoring pathway. The NuScale analysis does not  
5 include potential dose from this -- leakage from this  
6 pathway because we could not make a finding that the  
7 application -- that the applicant has evaluated all  
8 the potential pathways for radioactivity that could  
9 lead to control room habitability problems as required  
10 under the TMI Item 28.

11 We're proposing in the design  
12 certification rule to have that carve-out that we  
13 talked about earlier so that the COL applicant would  
14 be required to provide additional information to show  
15 that that requirement is met and that the dose results  
16 either are held to be still acceptable, but the  
17 leakage is so low that it does not affect the dose  
18 results that are currently in the FSAR or that they  
19 make changes to either the dose results or to the  
20 design to accomplish that.

21 Next slide, please.

22 DR. CORRADINI: This connects back up to  
23 what we started the whole discussion with, right?

24 MS. HART: Yes, but I'm not quite done  
25 yet.

1           Next slide, please. So I did do some  
2       confirmatory analysis for each of the accidents. I  
3       used the topical report methodology and  
4       design-specific input from the design certification  
5       application and some RAI responses and audit -- you  
6       know, confirmed that I had the right information  
7       through audit of their calculations.

8           I also did control room ventilation system  
9       modeling sensitivity because I did have some concern  
10      about giving credit for two of the non-safety-related  
11      ventilation systems that was there. An impact of  
12      assuming that the ventilation system with filtration  
13      did not come on after the initial bottled air system,  
14      and there was not a big change in the results, so I  
15      therefore found NuScale's dose analyses were  
16      acceptable and within the applicable accident-specific  
17      dose criteria off site and in the control room.

18          Next slide, please. So in conclusion, for  
19      the SER Section 15.0.3, the staff found reasonable  
20      assurance that the SMR design meets the off-site dose  
21      criterion and the accident-specific off-site dose  
22      acceptance criteria given in the DSRS. We found  
23      reasonable assurance that the main control room  
24      habitability requirements in PDC 19 are met and we  
25      found reasonable assurance that the TSC habitability

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1 requirements are met. And those are all with the  
2 caveat of the carve-out to ensure that those -- that  
3 additional leakage pathway does not change any of  
4 those findings. And that concludes my presentation.

5 DR. CORRADINI: Can we go back to one that  
6 we started with on No. 38? This one. This is what we  
7 started with relative to the hydrogen monitoring.

8 So I want to make sure I understand. You  
9 did some site calculations that if there was leakage  
10 from the hydrogen monitoring system it could be  
11 problematic.

12 MS. HART: Correct.

13 DR. CORRADINI: How does that leakage rate  
14 compare to the containment leakage rate of 0.2 percent  
15 per day?

16 MS. HART: So it is much larger, the  
17 assumptions that I made. In fact the lower end of the  
18 leakage rate -- and I don't have those numbers in  
19 front of me. I do have to --

20 (Simultaneous speaking.)

21 DR. CORRADINI: What I'm trying to get an  
22 idea is just the relative --

23 MS. HART: It's --

24 DR. CORRADINI: -- comparison point.

25 MS. HART: -- at the -- if you assume the

1 lower end, that one liter per minute, and then 10  
2 percent leakage, that's approximately about the same  
3 amount of leakage from the containment is what my  
4 calculation shows. Like I said, it was a very rough  
5 calculation.

6 DR. CORRADINI: It was one liter per  
7 minute -- I'm sorry. Can you repeat it again? I'm  
8 sorry.

9 MS. HART: One liter per minute flow  
10 through the system. Ten percent of that leaks.

11 DR. CORRADINI: Okay.

12 MS. HART: So 0.1.

13 DR. CORRADINI: So 0.1 liter per minute  
14 leakage rate?

15 MS. HART: Right.

16 DR. CORRADINI: And that's approximately  
17 two percent per day?

18 MS. HART: Correct.

19 DR. CORRADINI: It's like doubling --

20 MS. HART: Well, it's around the same  
21 value. It's not exact.

22 DR. CORRADINI: Around? I understand.  
23 Okay.

24 MS. HART: It's comparable.

25 DR. CORRADINI: No, I was just trying to



1 get a feeling, because when you quoted the numbers  
2 earlier in the discussion, I was trying to get a  
3 comparison to --

4 MS. HART: Yes.

5 DR. CORRADINI: Okay. Thank you.

6 MEMBER REMPE: Let me try again with my --  
7 what I wanted to be a MELCOR question. But if I go to  
8 slide 36, you did use the MELCOR calculations at some  
9 point to do this, right? Whether the report is an old  
10 one and you've put into RADTRAD --

11 MS. HART: Right.

12 MEMBER REMPE: -- you did use the MELCOR  
13 stuff? How sensitive was your finding to how long you  
14 picked? Because there were a couple of times there  
15 clearly was some uncertainty whether you picked the  
16 right time on what the applicant had done. And if  
17 you'd done like --

18 MS. HART: So what we --

19 MEMBER REMPE: -- let's say 48 days versus  
20 an hour, would that have been a big difference in your  
21 finding?

22 MS. HART: So I did not calculate over 30  
23 days. I used 30 days as the duration.

24 MEMBER REMPE: Okay, but if you did --

25 MS. HART: So we'll start that as a start.

1                   MEMBER REMPE:  -- no, 48 hours versus an  
2                   hour?  Let's do two days versus one hour.  What would  
3                   be the difference?

4                   MS. HART:  So what we did for the two  
5                   MELCOR cases is we took that scenario.  We didn't  
6                   adjust it at all.  We said what it was was what it  
7                   was.  It included the release timing that it had, the  
8                   release fractions.  It included the thermal  
9                   hydraulics.  And it did not model containment release  
10                  the same way that is in the topical report or in the  
11                  FSAR.  It modeled MELCOR's evaluation of what the  
12                  release would be to the environment.  So it also -- it  
13                  adjusted that.  And then we used the MAEROS-based  
14                  model within MELCOR to do the aerosol deposition  
15                  within containment.  And so those were -- there were  
16                  -- all the differences were there.

17                 MEMBER REMPE:  So you picked a particular  
18                 scenario, not an average from the five or whatever --

19                 MS. HART:  Correct.

20                 MEMBER REMPE:  -- that the applicant  
21                 picked?

22                 MS. HART:  Correct.  We had those two  
23                 scenarios --

24                 MEMBER REMPE:  Okay.

25                 MS. HART:  -- that were described in the

1 topical report methodology that included failure of  
2 the ECCS with the reactor vent valve.

3 MEMBER REMPE: Okay.

4 DR. CORRADINI: According to Jason cases  
5 3 and 5.

6 MR. SCHAPEROW: Yes, so -- and you're  
7 correct. Those calculations were run for a specific  
8 period of time. I believe we ended them after two  
9 days. And you could see it there. So the release  
10 that went to the control room and off site, that was  
11 over the period of those two days.

12 Now MELCOR doesn't produce a lot of --  
13 MELCOR is -- well, MELCOR is supposed to be a  
14 realistic model and it doesn't have this assumption of  
15 five percent of the iodine being vapor for 30 days.  
16 We don't do that. We do whatever the code calculates.  
17 So after a couple of days you'll find pretty much all  
18 the fission products have deposited somewhere.

19 MEMBER REMPE: Right.

20 MR. SCHAPEROW: So we felt that a couple  
21 of days was a fine surrogate for this.

22 MEMBER REMPE: Okay.

23 MR. SCHAPEROW: Hopefully that answers the  
24 question.

25 MEMBER REMPE: I'll look at it some more.

1 Thank you.

2 MS. HART: Any other questions?

3 CHAIR KIRCHNER: Members, any further  
4 questions of the staff?

5 MS. HART: Oh, wait. Jason has something  
6 else he would like to say.

7 MR. SCHAPEROW: Yes, one other  
8 clarification. One of the early concerns we had was  
9 on the assumption of the leak rate, a steady leak rate  
10 of two percent per day for the first 24 hours after  
11 the accident and then one percent per day after that.  
12 So again with MELCOR we set a hole size to reflect the  
13 two percent per day leak rate with the design pressure  
14 of the containment filled with air. And we selected  
15 that hole size and we let MELCOR calculate the actual  
16 leak rate during the accident. And the leak rate of  
17 course varies based on what it's -- whether there's  
18 air in there or hydrogen or steam. So again, we  
19 endeavored to kind of just let MELCOR do its thing.  
20 But we had to fix the hole size. So that's how we  
21 fixed the hole size.

22 CHAIR KIRCHNER: Any further questions  
23 from the Committee?

24 So at this point I think a break is long  
25 overdue, but I think we're at closure. At this point

1 we would turn to the public for comments.

2 So if there's anyone in the room who  
3 wishes to make a comment please come forward.

4 MR. BECKER: Excuse me. Just before the  
5 public comment session I wanted to take the  
6 opportunity -- I'm Gary Becker on behalf of NuScale.  
7 Because of all the discussion on the hydrogen  
8 monitoring and everything else and the back and forth  
9 and -- between us and now with the staff, I just  
10 wanted to take an opportunity to restate and clarify  
11 our position to make sure it's clear for the record.

12 Several members have taken -- noted that  
13 this really is a design issue. We agree it's a design  
14 issue in the sense that it's something that should be  
15 addressed by a design certification. We don't think  
16 the carve-out is necessary or appropriate. We do not  
17 believe that there is a design issue in the sense that  
18 there's a problem with the design or an inadequacy.

19 The basic premise of our position, setting  
20 aside all the details, just to go to high-level  
21 principles, is that we've done what is necessary to  
22 demonstrate reasonable assurance of adequate  
23 protection. That is the fundamental question here is  
24 reasonable assurance of adequate protection.

25 So as we pointed out in our presentation,

1 the staff have found reasonable assurance of adequate  
2 protection based on a leakage control program that  
3 required leakage as low as practical for past  
4 applicants. We believe that same finding can apply to  
5 us based on our same leakage control requirement.

6 The core damage event is a very, very  
7 unlikely event that we've classified as a  
8 beyond-design-basis event, orders of magnitude below  
9 the safety goals. And so essentially our position is  
10 that we don't think it's necessary to account for the  
11 -- to quantitatively account for the leakage from the  
12 hydrogen monitoring program -- excuse me, hydrogen  
13 monitoring system within our core damage event  
14 radiological consequence analysis because we're around  
15 half of the regulatory limits, less than half of the  
16 regulatory limits for doses for this extremely  
17 unlikely event, and therefore we don't think it's  
18 necessary to perform this quantitative analysis which  
19 has never been required for a past applicant. I hope  
20 that's a -- captures the high-level summary. And if  
21 I can -- if there's any follow-up questions, I'd be  
22 happy to --

23 CHAIR KIRCHNER: Well, for me it begs the  
24 question why don't you ask for an exemption from that  
25 TMI requirement, post-TMI requirement? I don't have

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1 the number in front of me, but I have it written down  
2 somewhere.

3 MR. BECKER: I'm sorry?

4 (Simultaneous speaking.)

5 CHAIR KIRCHNER: -- you're saying?

6 MR. BECKER: No, I don't.

7 CHAIR KIRCHNER: Well, you've asked for  
8 exemptions in other cases. Why not -- if your  
9 argument is predicated on having substantial margin to  
10 meeting the safety goals and not having a severe  
11 accident essentially, then why not ask for an  
12 exemption for that -- from that specific rule --

13 MR. BECKER: I don't think there's a --

14 CHAIR KIRCHNER: -- for the hydrogen  
15 sampling?

16 MR. BECKER: The hydrogen monitoring,  
17 50.44?

18 CHAIR KIRCHNER: Yes.

19 MR. BECKER: We have considered it in the  
20 past. The -- and the basis for that would be the time  
21 frame in which hydrogen monitoring might become  
22 necessary is very long.

23 CHAIR KIRCHNER: Right.

24 MR. BECKER: All I could say is that in  
25 some of our early interactions with the staff

1 preapplication period we had some feedback that that  
2 exemption wouldn't be viewed favorably, and so we  
3 decided not to pursue that path.

4 CHAIR KIRCHNER: I was just looking for  
5 closure. That's all. It was a rhetorical question.

6 MR. BECKER: Okay.

7 CHAIR KIRCHNER: Thank you again.

8 MR. BECKER: You're welcome.

9 CHAIR KIRCHNER: All right. Now I'll  
10 repeat myself. Any member of the public present  
11 wishing to make a comment? Come forward.

12 Okay. Then we'll just wait a moment here  
13 for the public line to be open.

14 PARTICIPANT: The light's on.

15 PARTICIPANT: You can ask.

16 CHAIR KIRCHNER: Oh, I can go ahead and  
17 ask?

18 PARTICIPANT: Yes.

19 CHAIR KIRCHNER: Going to have to open it  
20 out of the control room? Yes, that's what I thought.

21 MR. SHAVER: So if there's anyone from the  
22 public that would like to make a public comment, you  
23 just need to take your phone off mute and make your  
24 comment. Is there anyone on the line, the public  
25 line?



1 MS. FIELDS: Yes, this is Sarah Fields and  
2 just one comment I have is it's apparent that NuScale  
3 is going to in some way supplement their DCA to a  
4 standard design approval application. They expect to  
5 submit that in 2021. And they've had an initial  
6 meeting with the NRC and now they've put forward to  
7 the NRC a NuScale SMR standard design approval  
8 regulatory engagement plan. They have a new docket  
9 for this. Since the application isn't submitted it's  
10 not a Part 52 docket, but it's a docket No. 99902078.  
11 So someone can check that.

12 I have a concern that this plan, which is  
13 a regulatory engagement plan, is proprietary. So a  
14 member of the public can't view this plan. And they  
15 did not submit a non-proprietary version of this plan  
16 to the NRC. It's unclear why a regulatory engagement  
17 plan would contain so many elements that would put it  
18 under a proprietary designation. But moving forward  
19 maybe some of the things that you've discussed today  
20 that have not been resolved NuScale might resolve  
21 these in the SDA process because they -- they did  
22 mention because they did have some slides that they  
23 presented to the NRC and they are available. But I  
24 would think that a regulatory engagement plan would be  
25 something that should be accessible to the public, at

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1 least a non-proprietary version be submitted. Thank  
2 you.

3 CHAIR KIRCHNER: Thank you, Sarah. Any  
4 other members of the public wishing to make a comment?

5 Hearing none, we can go ahead then and  
6 close the public line. In summary, Dave, do you want  
7 to make any comments?

8 CO-CHAIR PETTI: Yes, I mean we just want  
9 to thank NuScale and the staff. We did two things  
10 here: We reviewed the technical -- the topical  
11 report, but we also did a lot to focus a review of the  
12 source term for NuScale. You guys are the first that  
13 have gone through the focus area review and so it's  
14 very helpful the way things were organized I think to  
15 see how they were all closed out. So good job.

16 CHAIR KIRCHNER: Yes. Let's go around the  
17 table anticipating a letter on this topic as well as  
18 a full Committee meeting. So we pretty much have  
19 almost the full Committee here.

20 Vesna?

21 MEMBER DIMITRIJEVIC: I have no additional  
22 comments. Thanks. That was interesting.

23 CHAIR KIRCHNER: Dennis?

24 MEMBER BLEY: Nothing additional.

25 MEMBER MARCH-LEUBA: Nothing more.

1 MEMBER BROWN: Nothing more.

2 MEMBER REMPE: I have to remember which  
3 mic to use. There were a couple items I requested and  
4 I would like to see those happen. I don't think I  
5 need to reiterate them, though. Thanks to everyone  
6 who presented.

7 CHAIR KIRCHNER: Ron?

8 MEMBER BALLINGER: Nothing more. Thank  
9 you.

10 CHAIR KIRCHNER: Mike and Steve?

11 DR. CORRADINI: You want to go first?

12 DR. SCHULTZ: My stopping point in the  
13 reviews in preparation for the meeting was the  
14 hydrogen sampling monitoring issue, and we've  
15 discussed that thoroughly. I wish we had been able to  
16 contribute more fully to the resolution of that today,  
17 but we certainly have the information on the table for  
18 consideration by the committee.

19 DR. CORRADINI: Is -- I wanted to -- I  
20 guess I'm in agreement with Steve on this about  
21 hydrogen sampling. I guess I didn't anticipate this  
22 to be what turned out to be the discussion.

23 I wanted to ask Walt and Dave, is it the  
24 plan to essentially address that as part of the  
25 letter?

1 CO-CHAIR PETTI: Yes, as it stands now.  
2 It's all in one letter unless we decide when we get to  
3 letter writing that we should separate it into two  
4 letters: one on the topical report and one on the  
5 focus areas.

6 DR. CORRADINI: Okay. So my feeling is --  
7 my worry is, not feeling -- since I experienced the  
8 ESBWR and the AP1000 I'm concerned about leaving a  
9 design issue on the table unresolved to a COL  
10 applicant. It just -- it -- we've done that before  
11 and it didn't go well, so I would prefer we not do  
12 that this time. Either staff or applicant's got to  
13 sit down and figure out how to resolve this from a  
14 design standpoint, or as Walt requested, they ask for  
15 an exemption from the requirement.

16 CO-CHAIR PETTI: So is it appropriate for  
17 us to recommend and encourage them to come to  
18 resolution?

19 DR. CORRADINI: It's appropriate for us to  
20 do whatever we feel like.

21 CO-CHAIR PETTI: Okay.

22 CHAIR KIRCHNER: Well, I think certainly  
23 it's going to show up prominently in our letter based  
24 on the full Committee's input, but it was our intent,  
25 as Dave said, to have one letter to cover --

1 DR. CORRADINI: Okay.

2 CHAIR KIRCHNER: -- this broad area.

3 DR. CORRADINI: Okay.

4 CHAIR KIRCHNER: And it's not our  
5 responsibility to redesign --

6 DR. CORRADINI: Nope.

7 CHAIR KIRCHNER: -- the system, but we can  
8 flag a concern that we have certainly.

9 MEMBER BROWN: Could you articulate the  
10 design issues so that it's clear in our transcript?  
11 I heard all the --

12 DR. CORRADINI: You're asking me?

13 MEMBER BROWN: Huh?

14 DR. CORRADINI: You're asking me?

15 MEMBER BROWN: Yes, you're the one that  
16 brought it up, so --

17 DR. CORRADINI: Okay.

18 MEMBER BROWN: I agree with you, but  
19 you're --

20 DR. CORRADINI: Well, I think --

21 MEMBER BROWN: -- I'm not adequate to --

22 DR. CORRADINI: Well, Steve and I have  
23 been kibitzing, but I think; Steve can correct me if  
24 I'm wrong, my understanding is that with the hydrogen  
25 sampling issue they're in the conundrum that they've

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1 -- they -- if they follow the TMI rule, they just do  
2 monitoring both early time and late time, and that  
3 creates a potential issue about leakage through that  
4 system. And at least based on some of the staff's  
5 initial estimates, that leakage rate could be  
6 significant enough that it could challenge the five  
7 rem habitability issue, relative control and  
8 operators.

9 And if they choose to isolate, that  
10 essentially then goes in -- goes against the  
11 requirement, the TMI requirement for continuous  
12 monitoring of hydrogen long term into the accident.  
13 Given Anne-Marie's suggestion, they could start it and  
14 stop it and start again. It's still is against the  
15 spirit of it. So it seems to me we've got two things  
16 butting up against each other that could potentially  
17 either be solved by design or by, as Walt suggested,  
18 a requested exemption.

19 So somehow the letter has got to address  
20 this because this is the only one that I saw amongst  
21 all the view graphs that seems to be unresolved from  
22 the staff and the applicant. Did I say it close to  
23 right?

24 DR. SCHULTZ: Yes, that's correct. The  
25 applicant has indicated that they don't feel they need

1 to go in that direction in the first place, that the  
2 ALARA principle associated with the ability to monitor  
3 leakage during normal operation should be sufficient  
4 because of the low likelihood of the event to occur in  
5 the first place.

6 MEMBER BROWN: Would it be useful then for  
7 the challenge to the staff to at least during full  
8 Committee to frame this issue so that we can have it  
9 clearly characterized during the full Committee  
10 meeting with the full set of -- we've got what 8 of  
11 the what 10 or 11 here?

12 CHAIR KIRCHNER: And Pete was on this  
13 morning as well --

14 MEMBER BROWN: Okay. I missed that.

15 CHAIR KIRCHNER: -- so we have almost --  
16 no, but of course.

17 MEMBER BROWN: I just think we need to try  
18 to characterize it during the full Committee meeting  
19 so we know and get it clear into the letter what we're  
20 doing. That's all. That's just my suggestion.  
21 That's all I had.

22 CHAIR KIRCHNER: We'll try and do that.

23 MEMBER DIMITRIJEVIC: Well, I want to  
24 point out also in addition what you identified there  
25 is problem with the insulation because it has to be

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1 .done locally and it will run into the exposure issue  
2 for whoever has to go and insulate it when they  
3 insulate. That's also part of the problem.

4 CHAIR KIRCHNER: Getachew, is it  
5 reasonable for us to ask you at the full Committee to  
6 frame your proposed carve-out as -- I think that was  
7 the terminology you all were using for this particular  
8 issue how you're going to not get closure, but as it  
9 stands right now it appears that it's going to be the  
10 COL applicant who has to resolve it. But you'll need  
11 to have a crisp clear definition for your carve-out,  
12 right, of what is required? Or do you feel you  
13 already have that in the SER at this point?

14 MR. DUDEK: So this is Michael Dudek, the  
15 branch chief. We do have a pretty concise description  
16 in the SER and I think we're working with -- currently  
17 working with the lawyers on that, on what the actual  
18 carve-out language would say.

19 CHAIR KIRCHNER: Okay.

20 MR. DUDEK: And understanding if after  
21 extensive discussions with NuScale it will -- it was  
22 our ultimate goal to come to resolution on this, but  
23 we just weren't able. And understanding that the  
24 carve-out is a little clunky, but that was the best  
25 way that the staff envisioned its way to move forward



1 and not providing finality.

2 MEMBER REMPE: But, yes, you had the text  
3 of what your carve-out was in your slide, right? Is  
4 it going to be different than that?

5 MR. DUDEK: That's not the exact language  
6 that will be in the rule.

7 CHAIR KIRCHNER: Yes, so we would -- to  
8 the best of your ability give us whatever that  
9 carve-out is, because then we as a Committee will  
10 respond to that accordingly.

11 DR. CORRADINI: So Steve and I were --  
12 just to get to -- somebody moves on -- so is the top  
13 of page 18 the only place we found it on the SER in  
14 terms of the discussion? We're looking for the clear  
15 and concise thing that's in the SER. I'm trying to  
16 make sure that I didn't miss it somewhere else. It's  
17 not here. It's just hinted at here.

18 PARTICIPANT: Yes.

19 DR. CORRADINI: Okay. Fine. All right.  
20 I misunderstood.

21 CHAIR KIRCHNER: That's what I thought.

22 DR. CORRADINI: Because this is the only  
23 thing that -- okay.

24 (Simultaneous speaking.)

25 CO-CHAIR PETTI: I'm mixing this up in my

1 mind. I just reviewed Chapter 12. It is in Chapter  
2 12, I can tell you that.

3 PARTICIPANT: Yes. It's in the Chapter 12  
4 SER, yes.

5 CO-CHAIR PETTI: Because it's in the draft  
6 letter I'm writing on Chapter 12. And I will let you  
7 know that since I'm -- try to get things done quickly,  
8 there is a draft on SharePoint of my thoughts about  
9 this stuff that serves as a starting point for our  
10 letter. So, because with the holidays coming up I  
11 just had to get it done early.

12 CHAIR KIRCHNER: Further discussion,  
13 anyone?

14 MR. SNODDERLY: So, Walt, if I could -- so  
15 right now we have -- the first date of the meeting is  
16 Wednesday, November 4th, 2:45 to 5:00. So that's two  
17 hours and fifteen minutes. I guess the first question  
18 I'd ask of you is do you want NuScale and the staff,  
19 or the staff and just NuScale there in support? And  
20 to what depth? What did you want them to cover? I  
21 think --

22 CO-CHAIR PETTI: Yes, I think the staff  
23 and just say -- let's call it local NuScale support.  
24 And then if Corvallis can be on the line. I think you  
25 want -- we want to hear their side of it. So whatever

1 they think they need to have their side of it there.

2 MR. SNODDERLY: Okay. So I'll work with  
3 Getachew and NuScale to figure out the proportions,  
4 but it will be mostly the staff with NuScale in  
5 support with their bridge line and there to answer  
6 questions. And obviously they'll probably want to  
7 weigh in on this -- on the hydrogen monitoring issue.  
8 That one will clearly be covered.

9 Since we -- since the holiday is coming up  
10 and everything -- so one thing I -- so I had the two  
11 action items for Joy was RAI 9690, right, and the  
12 subsequent MELCOR analysis report, the one that  
13 apparently --

14 MEMBER REMPE: Well, apparently --  
15 (Simultaneous speaking.)

16 MR. SNODDERLY: -- for 2019. So --

17 MEMBER REMPE: Is it MELCOR or is it -- he  
18 was talking about -- it was RADTRAD.

19 MR. SNODDERLY: MELCOR -- so MELCOR --  
20 there's the MELCOR analyses and there's the RADTRAD  
21 analyses. And then I can't speak for Jason. I don't  
22 know specifically what report he was talking about.

23 MEMBER REMPE: Whatever it is he's saying.  
24 And if that MELCOR report is not the right one, please  
25 do give us whatever. I mean it's dated April.

1 MR. SNODDERLY: The MELCOR -- right.  
2 That's why I think we're talking past -- let me work  
3 with them, but the point is what I'm asking, Joy, is  
4 because we're running out -- because we don't have as  
5 much time, should we just have the staff prepared to  
6 go over RAI 9690 --

7 MEMBER REMPE: Well, it's the hydrogen --  
8 it's whatever was the response on this hydrogen  
9 monitoring. So --

10 MR. SNODDERLY: Okay. So then that --

11 MEMBER REMPE: But I'd just like to see  
12 the text.

13 MR. SNODDERLY: -- so why don't you plan,  
14 Getachew -- why don't -- as part of the hydrogen  
15 monitoring issue please be prepared to summarize the  
16 staff's acceptance or resolution of the RAI 9690.

17 MEMBER REMPE: I think they did.

18 MR. SNODDERLY: They put it together.

19 MEMBER REMPE: It's just we never saw the  
20 original text. That's what I was asking for.

21 MR. SNODDERLY: Right.

22 MEMBER REMPE: I think that both sides  
23 presented. I just would like to see what it was they  
24 sent you back.

25 MR. SNODDERLY: All right. Then we'll get

1 -- then I'll work with Getachew. We'll get you the  
2 RAI -- or the response to 9690.

3 MR. CAMPBELL: We took it as an IOU --

4 MR. SNODDERLY: Okay.

5 MR. CAMPBELL: -- Mike, and Committee  
6 members, to --

7 MR. SNODDERLY: Right.

8 MR. CAMPBELL: -- get you the right report  
9 that you were looking for --

10 MR. SNODDERLY: Okay.

11 MR. CAMPBELL: -- that has the RADTRAD  
12 results.

13 MR. SNODDERLY: Thank you.

14 CHAIR KIRCHNER: Okay. Finished?

15 Okay. We are adjourned.

16 (Whereupon, the above-entitled matter went  
17 off the record at 4:12 p.m.)

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# **Accident Source Term Methodology Topical Report Staff Review**

**NuScale Design Certification Application Review**

**Presentation to the ACRS Subcommittee**

November 20, 2019

# Staff Review Team

- Technical Staff
  - ◆ Michelle Hart, NRR
  - ◆ Jason White, NRR
  - ◆ Jason Schaperow, NRR
  - ◆ Tony Gardner, NRR
  - ◆ Ed Stutzcage, NRR
  - ◆ Ron LaVera, NRR
  - ◆ Shawn Campbell, RES
  - ◆ Hossein Esmaili, RES
  
- Project Managers
  - ◆ Getachew Tesfaye – Chapter PM
  - ◆ Greg Cranston – Lead PM

# Contents

- Background and review approach
- Atmospheric dispersion
- Accident source terms that do not include core damage
- Core damage event
- Conditions and limitations
- Conclusion



## **TR-0915-17565**

- Methodology for NuScale SMR accident dose analyses including development of source terms
- NuScale developed codes
  - ◆ In-containment  $\text{pH}_T$
  - ◆ Site characteristic atmospheric dispersion factors
- Requested approval of 15 specific positions listed in TR Section 1.2

## TR Revision 3

- NuScale proposed to make changes, starting Jan. 2018
  - ◆ Issues with environmental qualification (EQ)
- Changes
  - ◆ Originally described a postulated maximum hypothetical accident (MHA) to meet offsite dose regulatory requirements
  - ◆ Now instead describes a core damage event (CDE)
  - ◆ Added iodine spike design basis source term
  - ◆ EQ source based on coolant activity release instead of core damage
- SECY-19-0079
  - ◆ Describes staff review approach to evaluate accident source terms for both the TR and the DCA
  - ◆ Basis for using source term without core damage for EQ, in contrast to other accident dose-related regulatory requirements

# Atmospheric Dispersion

- PAVAN and ARCON96 are NRC computer codes approved for calculating atmospheric relative concentration values (also known as atmospheric dispersion factors or  $\chi/Q$  values)
- PAVAN implements the guidance provided in Regulatory Guide (RG) 1.145 for determining offsite  $\chi/Q$  values at the Exclusion Area Boundary (EAB) and outer boundary of the Low Population Zone (LPZ)
- ARCON96 implements the guidance provided in RG 1.194 for determining onsite  $\chi/Q$  values for the control room (CR)

# Atmospheric Dispersion

- NuScale's topical report describes the applicant's methods for determining accident  $\chi/Q$  values for the EAB and LPZ using a methodology that differs from the NRC's guidance
- The staff reviewed the topical report in accordance with NUREG-0800, Section 2.3.4

# Atmospheric Dispersion

- NuScale proposes using the ARCON96 computer code methodology for calculating offsite atmospheric dispersion values rather than using the PAVAN computer code (Position 2)
- ARCON96 is a general code for assessing atmospheric relative concentrations in building wakes under a wide range of situations
- The ARCON96 dispersion algorithms are based on field measurements taken out to distances of 1,200 meters
- Typical EAB and LPZ distances range from 800 to 6,000 meters

# Atmospheric Dispersion

- NuScale postulates an EAB and LPZ at the site boundary, which is estimated to be in the range of 80 to 400 meters
- NuScale states that the ARCON96 computer code, which was developed to model shorter distances, is more appropriate for modeling NuScale EAB and LPZ  $\chi/Q$  values
- NuScale plans to use its version of ARCON96, NARCON, instead of PAVAN to calculate EAB and LPZ  $\chi/Q$  values

# Atmospheric Dispersion

- NRC staff conducted an audit of the topical report
- Staff reviewed the documentation for the NARCON computer code
- Staff executed several independent runs of the NARCON computer code

# Atmospheric Dispersion

- The staff concluded that, subject to the conditions and limitations specified in Section 6.0 of the SER, the NuScale methods described in the topical report are acceptable for calculating offsite  $\chi/Q$  values for the EAB and LPZ in a NuScale design or in a COL application referencing the NuScale design
- Therefore, staff finds TR Position 2 acceptable



# Accident Source Terms That Do Not Include Core Damage

- Following accidents same as typically evaluated for PWRs
  - ◆ Fuel handling accident (FHA)
  - ◆ Rod ejection accident (REA) with fuel damage
  - ◆ Main steam line break (MSLB) outside containment
  - ◆ Steam generator tube failure (SGTF)
  - ◆ Failure of small lines carrying primary coolant outside containment
- Iodine spike design basis source term (DBST)
- Staff found that the methodology for each is consistent with guidance in RG 1.183 for PWR accidents
  - ◆ Some specifics further discussed in following slides

# Iodine Spike DBST

- Not based on any specific scenario
- Bounding source term for any release of coolant without core damage
  - ◆ Instantaneous release to containment of all radioactivity in primary coolant
  - ◆ Iodine spiking, 2 cases
  - ◆ Iodine chemical form 97% elemental, 3% organic (arbitrary)
  - ◆ Homogenously mixed throughout free air volume
  - ◆ Design basis containment leak rate for first 24 hours, then half that rate
- Basis for EQ radiation source
- Analysis assumptions consistent with similar accidents in RG 1.183
- Staff found acceptable

# Primary Coolant Iodine Spiking

## ➤ Coincident iodine spike

- ♦ Design basis limit equilibrium activity concentration for iodine and noble gases
- ♦ 8-hr coincident iodine spike at **x** times the equilibrium appearance rate

## ➤ Pre-incident iodine spike

- ♦ Design basis limit maximum allowed activity concentration

Accident	Coincident Spike	Spiking Factor	Pre-incident Spike
MSLB, Iodine spike DBST	Yes	500	Yes
SGTF	Yes	500	Yes
Small line break	Yes	335	No

## ➤ Staff found acceptable because consistent with RG 1.183

- ♦ TR Position 11 is acceptable

# Assumptions on Secondary Coolant Volume

- REA, MSLB and SGTF modeling neglects the small secondary side volume that could contain activity from primary to secondary leakage (Position 13)
- For NuScale SMR design
  - ◆ Ratio of secondary coolant volume to primary is approximately 1%
  - ◆ Initial secondary coolant activity concentration is an order of magnitude less than primary
- Staff finds acceptable because initial secondary coolant activity would not add significant dose
  - ◆ TR Position 13 is acceptable

# Pool Effective Iodine Decontamination Factor

- FHA assumes effective iodine DF of 200 for release from the reactor pool consistent with RG 1.183 (Position 12)
- NuScale fuel is similar to fuel covered by RG 1.183
- Water depth above damaged fuel is greater than 23 feet used as basis for DF in RG 1.183
- Staff found this assumption to be consistent with RG 1.183 and acceptable
  - ◆ TR Position 12 is acceptable

# Containment Shine

- Dose analyses assume that containment shine is negligible (Position 15)
- Staff audited proprietary calculations for shine contribution assuming the CDE source term in containment and confirmed negligible
  - ◆ TR Position 15 is acceptable

## EQ Source

- RGs 1.89 and 1.183 indicate that a core damage source term should be considered for EQ
- However, since design basis events do not result in core damage in the NuScale design, the staff determined that core damage need not be assumed for EQ (see SECY 19-0079)
  - ◆ 10 CFR 50.49(e)(4) states that the post-accident radiation environment must be based on the most severe design basis accident
  - ◆ GDC 4 requires that structures, systems, and components important to safety be compatible with the environmental conditions associated with postulated accidents, including loss-of-coolant accidents
    - Note: This is different than some other requirements which indicate that beyond design basis events or core damage accidents should be considered (e.g. 10 CFR 52.47(a)(2)(iv), 10 CFR 50.44(c)(4) and some TMI requirements)
- NuScale EQ source
  - ◆ Reactor coolant spiking event

## **Iodine Spike Source for EQ**

- TR provides the iodine spike source term and dose methodology used for EQ inside containment and under the bioshield
- The source term does not consider the spiking of radionuclides other than iodine
  - ◆ Same source term assumptions for spiking as MCR and offsite dose
  - ◆ Information available to staff indicates that other radionuclides may spike following reactor transients (e.g. cesiums and noble gases)
  - ◆ These other radionuclides could be significant to the EQ dose
- The applicant included numerous conservatisms in the EQ dose methodology inside containment (both liquid and vapor regions considered) and under the bioshield, but it was unclear that these conservatisms bound the potential spiking of other radionuclides



# Resolution of Only Spiking Iodines

- Due to the small containment size in the NuScale design, dose rates inside containment and under the bioshield are high during normal operation
- The staff performed a conservative independent analysis with spiked source terms
  - ◆ Based on the staff's independent review and analysis the staff found that the normal operation dose provided sufficient margin over the accident doses, that the potential spiking of other radionuclides would not be a significant contributor to the total integrated dose (normal operation plus accident)
- Staff approves the use of the EQ dose methodology for calculating doses inside containment and under the bioshield and direct radiation shine from the associated sources for the NuScale design

# FSAR EQ Dose

- EQ source term and total integrated dose values in the FSAR are more conservative than the values that would be obtained using the TR methodology
- Since the use of these values would result in equipment being qualified to a higher value than indicated in the approved methodology, the staff found the FSAR source term and total integrated dose values to be acceptable

# Equipment Survivability

- While core damage was not assessed for EQ, certain equipment associated with containment integrity and combustible gas monitoring is designed to function to withstand core damage events, in accordance with 10 CFR 52.47(a)(23), 10 CFR 50.44, SECY 90-016, and SECY 93-087
  - ◆ Equipment survivability is discussed in more detail in the afternoon session

## Core Damage Event (CDE)

- 10 CFR 52.47(a)(2)(iv), DCA safety analysis
  - ◆ Footnote 3 fission product release:

*The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.*
- TMI-related requirements in 10 CFR 50.34(f)(2) with similar source term footnote
  - ◆ Shielding for vital area access (vii)
  - ◆ Post-accident sampling (viii) - requested exemption
  - ◆ Leakage control outside containment (xxvi)
  - ◆ Control room habitability problem pathways (xxviii)

## TR Position 1

- “Treatment of the CDE, postulated as a major accident for purposes of site analysis pursuant to Footnote 3 of 10 CFR 52.47(a)(2)(iv), is an appropriate beyond design-basis event for the NuScale design.”
  - ◆ CDE not a specific accident scenario
  - ◆ Source term derived from range of accident scenarios that result in significant damage to the core
  - ◆ Core damage is beyond the design basis
  
- Staff makes **no finding** on Position 1
  - ◆ Applicable regulations do not require classification of source terms as “design basis” or “beyond design basis” to demonstrate compliance with the requirements
  - ◆ Classification of CDE is not material to the staff’s findings under these regulations

## **CDE Scenario Selection**

- Core damage source term (CDST) representative of core damage events, not a specific scenario
- Derived from spectrum of surrogate accident scenarios that involve substantial damage to the core and release to the containment
  - ◆ Informed by the NuScale PRA
    - Single module internal events at full power
    - Intact containment
  - ◆ Intended to be representative or bounding of a dominant majority of intact containment CDEs for the NuScale nuclear power module
- Consideration of a range of core damage scenarios is consistent with the source terms described in NUREG-1465 and RG 1.183
- Staff found acceptable because consistent with Footnote 3 to 10 CFR 52.47(a)(2)(iv) for the fission product release

# CDST Release to Containment

- CDST based on NuScale-specific analyses using MELCOR to be performed by applicant referencing the TR
  - ◆ Characterized by magnitude, timing, chemical grouping and iodine chemical forms
  
- Release magnitude (TR Position 5)
  - ◆ Fraction of core inventory released to containment by chemical group
  - ◆ Use median release values over all scenarios, for each group
  - ◆ **Not** consistent with NUREG-1465, which used mean values with some adjustments
    - Staff evaluated sensitivity of release magnitude in integrated fashion for the CDST using independent confirmatory analyses
  
- Release timing (TR Position 4)
  - ◆ Use minimum release onset of any scenario evaluated
  - ◆ Use associated release duration from same scenario
  - ◆ Consistent with NUREG-1465 and RG 1.183

# CDST Release to Containment (cont.)

- Chemical grouping (TR Position 6)
  - ◆ Radionuclides grouped based on 2011 Sandia report for LWR source terms for high-burnup or MOX fuel
  - ◆ Consistent with state-of-the-art in severe accident modeling
  - ◆ TR position 6 acceptable
  
- Iodine chemical form fractions
  - ◆ Same as NUREG-1465 and RG 1.183
  - ◆ PHEBUS tests demonstrate assumption is conservative
  
- Staff found methodology to characterize the CDST release to containment to be acceptable
  - ◆ Similar to development of LWR source terms in NUREG-1465 and RG 1.183
  - ◆ Integrated evaluation of CDE methodology by staff independent confirmatory analyses



# Iodine Retention In Containment

- Post-accident  $\text{pH}_T$  calculation
  - ◆ Methods consistent with the information in NUREG/CR-5950, “Iodine Evolution and pH Control, and guidance in RG 1.183 and SRP 6.5.2
  - ◆ NuScale  $\text{pH}_T$  code implements TR Section 4.4
  
- Iodine re-evolution is negligible for  $\text{pH}_T$  values between 6.0 and 7.0 (Position 14)
  - ◆ RG 1.183: iodine re-evolution need not be considered from in-containment water pools with a pH of 7 or greater
    - Based on NUREG/CR-5950
  - ◆ NuScale estimated less than 1 percent of the aqueous iodine is converted to elemental iodine for a  $\text{pH}_T$  value of 6.0 using Figure 3.1 of NUREG/CR-5950
  
- Methods consistent with guidance, therefore acceptable to the staff
  - ◆ TR Position 14 is acceptable

# Aerosol Removal In Containment

- Aerosol removal in containment through natural deposition phenomena
- Uses STARNAUA code
  - ◆ Includes models for natural deposition in containment (e.g., gravitational settling)
  - ◆ Staff has previously approved new LWR applications that used STARNAUA to calculate aerosol deposition factors
    - STARNAUA code not reviewed by the staff for approval or endorsement
- Staff found that the topical report methodology describes the modeling of applicable aerosol natural deposition phenomena in containment

# Staff Integrated Evaluation of CDE Analysis Methodology

- Objective: Confirm NuScale's methodology for CDE radiological consequence analyses
- NuScale adopted two assumptions from Regulatory Guide 1.183
  - ◆ Containment leak rate
    - Design leak rate for first 24 hours of release, then half that rate for the remainder of 30 days
  - ◆ Iodine gaseous fraction
    - 5% of iodine release assumed to be gaseous and not deposit in containment
- For other aspects, NuScale developed their own methodology
  - ◆ For example, release timing (delayed) and magnitude (lower)

# Staff Concerns With CDE Methodology

- Initial staff concerns with NuScale assumptions/parameters
  - ◆ Containment leak rate depends on containment atmosphere composition (hydrogen vs. air)
  - ◆ Uncertainty in extent of severe core damage
  - ◆ Aerosol deposition calculation
    - Non-radioactive aerosol mass assumption
    - Thermal hydraulic conditions
  - ◆ Magnitude of release (core release fraction) and aerosol deposition rate taken from different scenarios
- Related to TR Positions 3, 4, 5, 7, 8 and 10
- Staff evaluated NuScale positions in an integrated fashion by in-house independent confirmatory analysis with MELCOR and RADTRAD

# Staff MELCOR Modeling of Core Damage Releases

- Best-estimate prediction of release to environment for two core damage scenarios
  - ◆ Stuck open reactor vent valve (RVV) with subsequent opening of the remaining two RVVs (LEC-06T)
  - ◆ CVCS break inside containment with subsequent opening of all RVVs (LCC-05T)
  
- Staff independently developed MELCOR input models using DCA information

# Staff RADTRAD Dose Analyses

- Staff developed RADTRAD model using staff MELCOR-predicted releases to the environment for each scenario
- Other dose analysis assumptions from TR methodology and DCA
  - ◆ Atmospheric dispersion factors
  - ◆ Breathing rates
  - ◆ Dose conversion factors
  - ◆ Control room modeling – 2 ventilation system cases
    - Case 1: control room habitability system (CRHS) operation for 72 hours, then normal control room HVAC system (CRVS) filtered operation for the remainder of the accident duration
    - Case 2: CRVS filtered operation for duration
- Staff's independent analyses predict doses comparable to applicant results and below regulatory dose criteria

## **CDST Issue Resolution**

- Staff determined that the issues did not require additional follow-up
- Considering the technical bases, along with the staff's analysis of the sensitivity of the overall dose results to the uncertainty in the dose analysis modeling of these phenomena, the staff found that the methodology to develop the CDST is acceptable
- TR Positions 3, 4, 5, 7, 8 and 10 are acceptable

## Conditions and Limitations

- Approval applies only to the NuScale SMR design
  - ◆ Maintain the same fundamental size, geometry, and safety features of the design docketed in 52-048
- No finding on the treatment of the CDE as a beyond-design-basis event for the NuScale design
- Limited to specific assessments
- Conditions related to use of the atmospheric dispersion methodology by COL applicant



## Conclusions

- Staff found 14 of the 15 positions specified by NuScale in Section 1.2 of the topical report to be acceptable
- Staff found acceptable the methods for developing accident source terms and performing accident radiological consequence analyses to be referenced by the NuScale SMR design
- Staff approves the use of NuScale topical report TR-0915-17565, Revision 3, subject to the conditions and limitations specified

# Abbreviations

CDE	core damage event	rem	Roentgen equivalent man
CDST	core damage source term	RG	regulatory guide
COL	combined operating license	RVV	reactor vent valve
CRHS	control room habitability system	SECY	Commission paper
CRVS	normal control room HVAC system	SGTF	steam generator tube failure
CVCS	chemical and volume control system	SMR	small modular reactor
DBST	design basis source term	SSCs	structures, systems and components
DCA	design certification application	TEDE	total effective dose equivalent
DF	decontamination factor	TR	topical report
EQ	environmental qualification		
FHA	fuel handling accident		
HVAC	heating ventilation and air conditioning		
LWR	light water reactor		
MHA	maximum hypothetical accident		
MSLB	main steam line break		
pH <sub>T</sub>	temperature dependent pH		
PWR	pressurized water reactor		
REA	rod ejection accident		

November 13, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission  
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11555 Rockville Pike  
Rockville, MD 20852-2738

**SUBJECT:** NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale Topical Report – Accident Source Term Methodology," PM-1119-67927, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on November 20, 2019. The materials support NuScale's presentation of the "Accident Source Term Methodology" topical report.

The enclosure to this letter is the nonproprietary version of the presentation titled "ACRS Subcommittee Presentation: NuScale Topical Report – Accident Source Term Methodology," PM-1119-67927, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Carrie Fosaaen at 541-452-7126 or at [cfosaaen@nuscalepower.com](mailto:cfosaaen@nuscalepower.com).

Sincerely,



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Enclosure: "ACRS Subcommittee Presentation: NuScale Topical Report – Accident Source Term Methodology," PM-1119-67927, Revision 0

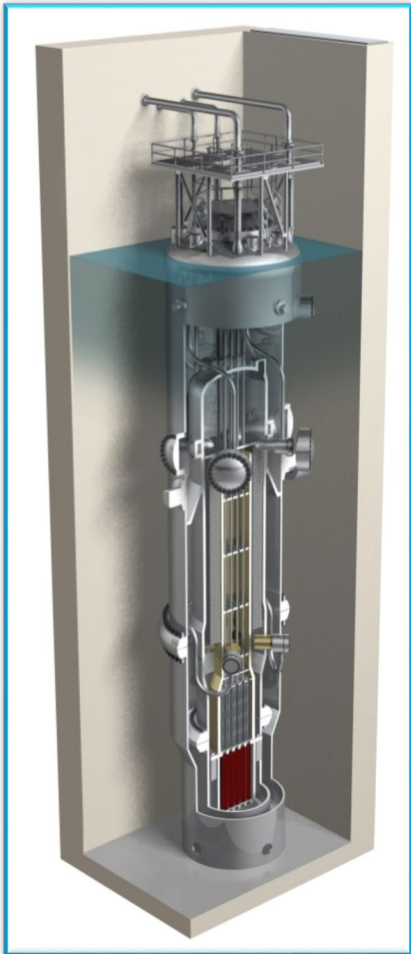
**Enclosure:**

“ACRS Subcommittee Presentation: NuScale Topical Report – Accident Source Term Methodology,”  
PM-1119-67927, Revision 0

# ACRS Subcommittee Presentation

## NuScale Topical Report Accident Source Term Methodology

*November 20, 2019*



# Presenters

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**Paul Guinn**

Radiological Safety Analyst

**Mark Shaver**

Radiological Engineering Supervisor

**Carrie Fosaaen**

Licensing Manager

**Jim Osborn**

Licensing Engineer

**Gary Becker**

Regulatory Affairs Council

# Agenda

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- NuScale AST Methodology Overview
- Closed Session

# Acronyms

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Term	Definition
AST	accident source term
CDE	core damage event
CDST	core damage source term
CNV	containment vessel
DBA	design basis accident
DBST	design basis source term
EAB	exclusion area boundary
EQ	environmental qualification
FHA	fuel handling accident
LPZ	low population zone
MSLB	main steam line break
NEI	Nuclear Energy Institute
PCA	primary coolant activity
REA	rod ejection accident
SAS	surrogate accident scenario
SGTF	steam generator tube failure



# NuScale AST Methodology

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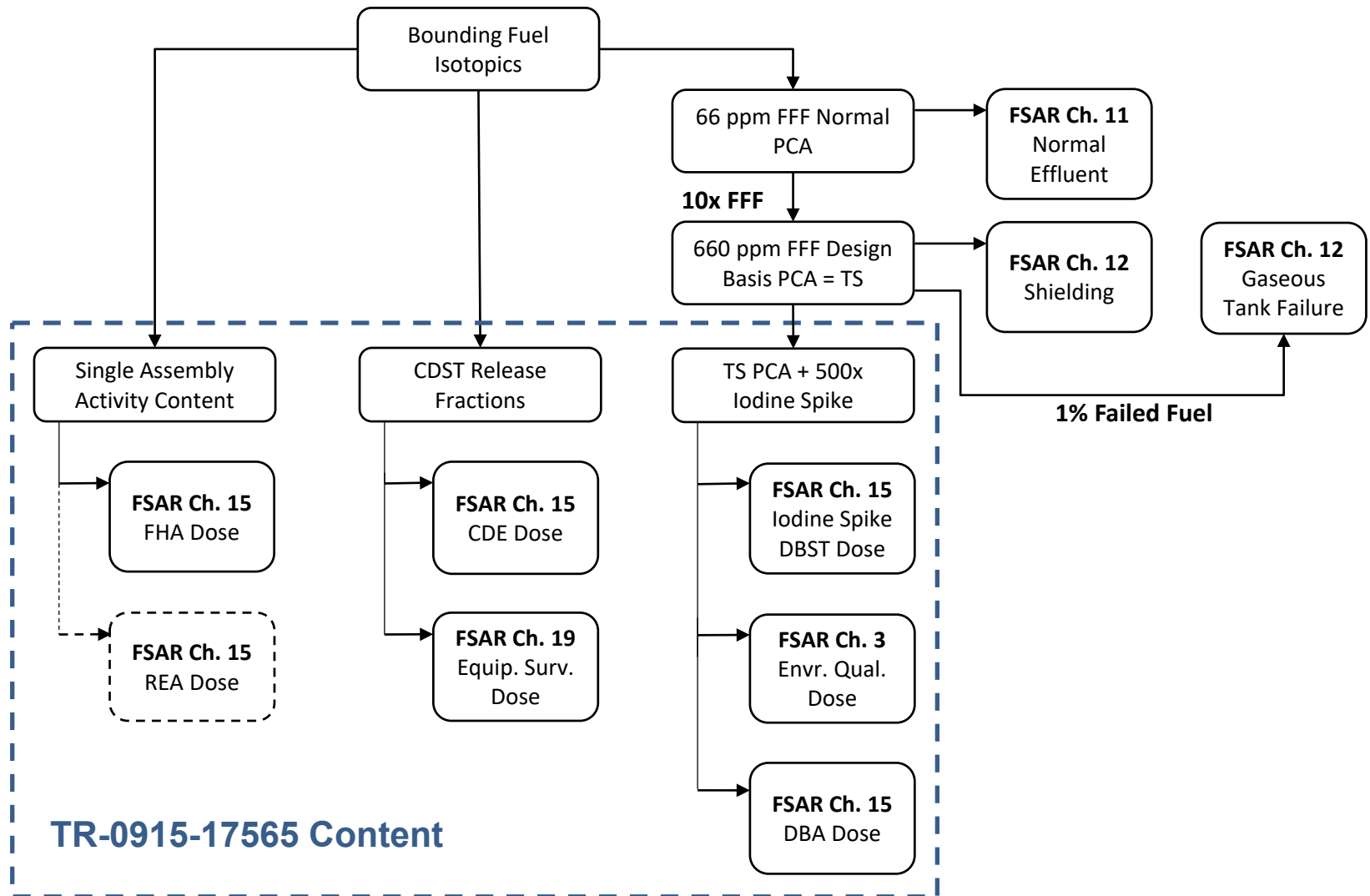
- Unique radiological consequence analysis methods within TR-0915-17565 include:
  - Atmospheric dispersion
  - Core damage source term
  - Containment aerosol transport and removal
  - Post-accident  $\text{pH}_T$
  - Iodine spike design basis source term (DBST)
  - EQ dose within CNV and bioshield envelope
- Industry standard radiological consequence analysis methods within TR-0915-17565 include:
  - Design basis accident source terms
    - REA, FHA, MSLB, SGTF, and small line break accident methodologies

# NuScale AST Methodology

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- Application of core damage event (CDE) limited by classifying it as a beyond design basis event
  - CDE used for control room dose evaluation and 10 CFR 52.47(a)(2)(iv) offsite dose evaluation
  - CDE not used for EQ
  - CDE used for equipment survivability evaluation
- Unique design basis event “iodine spike source term”
  - Surrogate for LOCA in containment without fuel damage events
    - Assumes 100% of radionuclides in primary (plus an iodine spike) are in containment
    - Used in conjunction with CDE to evaluate MHA radiological consequences for acceptability
    - Used in EQ

# Source Term Overview



# Software Utilized

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- General application software:
  - SCALE 6.1, TRITON, and ORIGEN-S – used to calculate the time-dependent isotopic source term for all evaluated radiological events
  - ARCON96 – used to calculate onsite and offsite atmospheric dispersion factors all evaluated radiological events
  - RADTRAD – used to estimate radionuclide transport and removal for all evaluated radiological events
- DBA application software:
  - NRELAP5 – used to provide event-specific thermal-hydraulic data for design basis events
- CDE application software:
  - MELCOR – used to model the progression of severe accidents for the CDE
  - STARNAUA – used to perform aerosol removal calculations for the CDE
  - NuScale pH<sub>T</sub> Code – used to calculate post-accident aqueous molar concentration of hydrogen ions for iodine re-evolution evaluation in the CDE
  - MCNP6 – used for evaluating potential shine radiological exposures or doses to operators in the control room following a radiological release event

# Atmospheric Dispersion

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- TR-0915-17565 position: ARCON96 (RG 1.194) methodology used for the calculation of offsite atmospheric dispersion factors
- ARCON96 and PAVAN compared to demonstrate conservative application of ARCON96 for NuScale site distances
- Unique NARCON atmospheric dispersion model created to apply ARCON96 model results which incorporate RG 1.145 (PAVAN methodology) modeling conservatisms

# Core Damage Source Term

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- Methodology based on state-of-the-art (post-RG 1.183) SAND2011-0128 severe accident modeling and the approach of the 2012 Nuclear Energy Institute (NEI) position paper on SMR source terms
- TR-0915-17565 positions:
  - CDE treated as beyond-design-basis event for the NuScale design, but evaluated in concert with iodine spike DBST to constitute a bifurcated maximum hypothetical accident
  - CDST derived from range of five surrogate accident scenarios taken from Level 1 PRA intact-containment internal events
    - SAND2011-0128 radionuclide groups for the CDST release groups
    - CDST release fractions taken as medians from SAS spectrum
    - CDST release timing from scenario of quickest core damage onset

# CDST Evaluation

- SAND2011-0128-based representative (median) release fractions from the spectrum of surrogate accident scenarios used as the CDST release fractions

SAND2011-0128 Radionuclide Groups		
Group Number	Chemical Group Name	Elements in Group
1	Noble Gases	Kr, Xe
2	Halogens	Br, I
3	Alkali Metals	Rb, Cs
4	Tellurium Group	Se, Sb, Te
5	Alkaline Earths	Sr, Ba
6	Molybdenum Group	Mo, Nb, Tc
7	Noble Metals	Ru, Rh, Pd, Co
8	Lanthanides	La, Nd, Eu, Pm, Pr, Sm, Y, Cm, Am
9	Cerium Group	Ce, Pu, Np, Zr

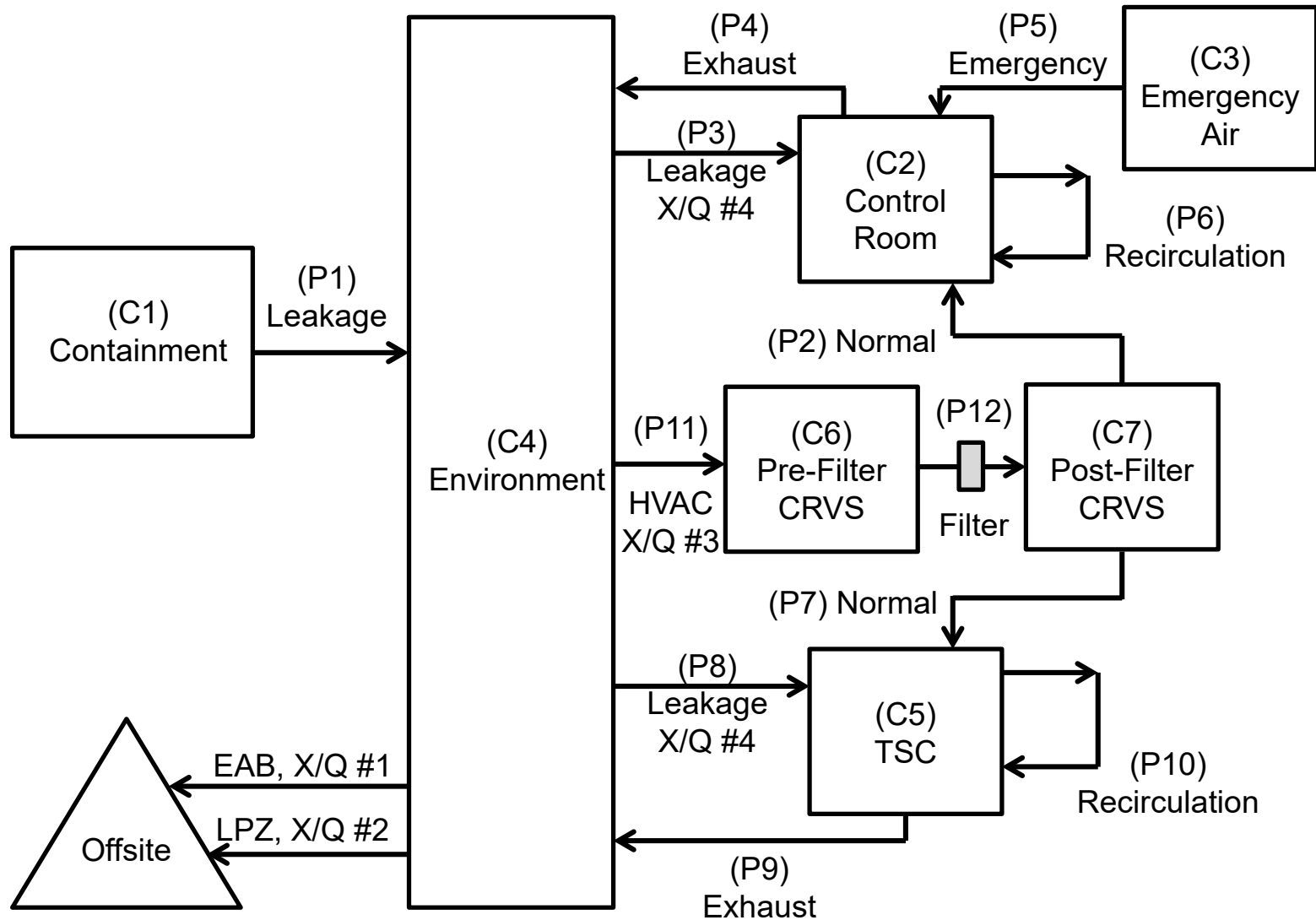
# CDST Evaluation (continued)

Comparison of release timing and magnitudes of example surrogate accident scenario cases

Description		Case 1	Case 2	Case 3	Case 4	Case 5	Median Value	RG 1.183 Current	SAND 2011-0128
Onset of gap release (hr)		17.6	<b>3.8</b>	8.1	6.2	21.3	8.1	30 sec	30 sec
Duration of gap plus early in-vessel release (hr)		12.0	<b>1.0</b>	9.0	1.3	14.0	9.0	1.8	5.63
Fraction of initial core inventory released into containment	Noble Gases	0.39	0.19	0.41	0.19	0.48	<b>0.39</b>	1	0.872
	Halogens	0.21	3.5E-2	0.16	1.9E-2	0.14	<b>0.14</b>	0.4	0.307
	Alkali Metals	0.25	5.9E-2	0.22	3.1E-2	0.20	<b>0.20</b>	0.3	0.235
	Alkaline Earths	5.9E-3	2.8E-3	6.7E-3	2.4E-3	5.3E-3	<b>5.3E-3</b>	0.02	0.0054
	Tellurium Group	0.22	3.8E-2	0.16	2.3E-2	0.15	<b>0.15</b>	0.05	0.267
	Molybdenum Group	6.4E-2	1.3E-2	5.3E-2	5.8E-3	4.9E-2	<b>4.9E-2</b>	0.0025	0.1
	Noble Metals	1.2E-3	1.2E-4	1.5E-3	4.9E-5	7.9E-4	<b>7.9E-4</b>	0.0025	0.006
	Lanthanides	3.3E-8	2.6E-9	3.1E-8	1.1E-9	2.1E-8	<b>2.1E-8</b>	0.0002	1.1E-7
	Cerium Group	3.3E-8	2.6E-9	3.1E-8	1.1E-9	2.1E-8	<b>2.1E-8</b>	0.0005	1.1E-7



# RADTRAD Model Nodalization



# Aerosol Transport and Removal

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- TR-0915-17565 positions:
  - STARNAUA is appropriate for modeling natural removal of containment aerosols for the NuScale design
  - Utilizing thermal-hydraulic data associated with the surrogate accident scenario with the minimum time to core damage is appropriate for use in STARNAUA
  - No maximum limit on iodine decontamination factor for natural removal of containment aerosols
- Methodology credits sedimentation, diffusiophoresis, and thermophoresis removal mechanisms
- Accident-specific natural deposition coefficient outputs of STARNAUA provided to RADTRAD dose transport model

# STARNAUA Removal Rate Error

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- Error identified when evaluating RAI 9224 (12/2017)
  - Time-dependent aerosol removal rate values appeared nonphysical
- Immediate actions:
  - Condition extent evaluation initiated (12/11/2017)
  - Notified NRC staff of identification (~12/15/2017)
  - Error discovery letter transmitted to code vendor (12/21/2017)
- Subsequent actions:
  - Completed extent evaluation to identify removal rate output issue as only impact
  - Developed alternative output postprocessing workaround
  - Rededicated code

# STARNAUA Error Extent

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- Limited to incorrect post-processed aerosol removal ( $\lambda$ ) values; internal NAUA subroutines not affected
- STARNAUA built as an extension of NAUAHYGROS 1.0
  - Thermophoretic deposition and spray removal models added
  - Changed output plot file generation to include calculated  $\lambda$  values
- NAUA-based codes historically demonstrated to predict aerosol mass concentration behavior reasonably well
- STARNAUA confirmed to predict aerosol mass concentration well by multiple experiment benchmarks
  - LACE 4, LACE 6, ABCOVE 5, ABCOVE 7 included
- Removal calculation validated by manual calculation

# Post-Accident $\text{pH}_T$

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- TR-0915-17565 position: For  $\text{pH}_T$  values of 6.0 or greater, the amount of iodine re-evolution that could occur between  $\text{pH}_T$  values of 6.0 and 7.0 is negligible and not included in the dose calculation
- Considers acids and bases expected in post-accident environment, including:
  - lithium hydroxide, cesium hydroxide, hydrochloric acid, nitric acid, hydriodic acid, and boric acid
- Given  $\text{pH}_T$ , amount of iodine re-evolution estimated using NUREG/CR-5950 methods and shown to be negligible (<1 %) with respect to other iodine modeling conservatisms

# Iodine Spike DBST

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- Iodine spike design basis source term:
  - Bounding surrogate source term for any design-basis event involving primary coolant loss inside containment; postulated to enable deterministic evaluation of the response of a facility's engineered safety features.
    - Assumes 100% of radionuclide inventory within 100% of primary coolant volume instantaneously, homogenously mixed release
    - Primary coolant inventory contains radionuclide concentrations at tech spec limits, plus iodine spike
    - Conservative evaluation-dependent leakage treatment
    - Additional conservative treatments available for discussion in closed session

# Environmental Qualification Dose

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- TR-0915-17565 position: methodology described in TR-0915-17565, Appendix B is appropriate for calculating environmental qualification doses in the containment vessel (CNV) and bioshield envelope regions
- Surrogate source term bounding for all primary coolant loss design basis events
  - Assumes 100% of radionuclide inventory within 100% of primary coolant volume instantaneously released inside containment
  - Primary coolant inventory contains radionuclide concentrations at tech spec limits, plus iodine spike
  - Multiple additional conservative treatments summarized in TR-0915-17565, Appendix B

# Other Positions

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- TR-0915-17565 positions:
  - Utilizing the iodine spiking assumptions of RG 1.183 is appropriate
  - Utilizing the iodine decontamination factor assumptions of RG 1.183 for the fuel handling accident is appropriate
  - Secondary coolant source negligible with regards to primary coolant source
  - Containment shine to the environment is negligible for the NuScale design



# Design Basis Accidents

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- Small line break outside containment
  - Follows SRP § 15.6.2
- Steam generator tube failure
  - Follows RG 1.183 App. F
- Main steam line break
  - Follows RG 1.183 App. E
- Rod ejection accident
  - Follows RG 1.183 App. H
- Fuel handling accident
  - Follows RG 1.183 App. B

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# Backup Slides

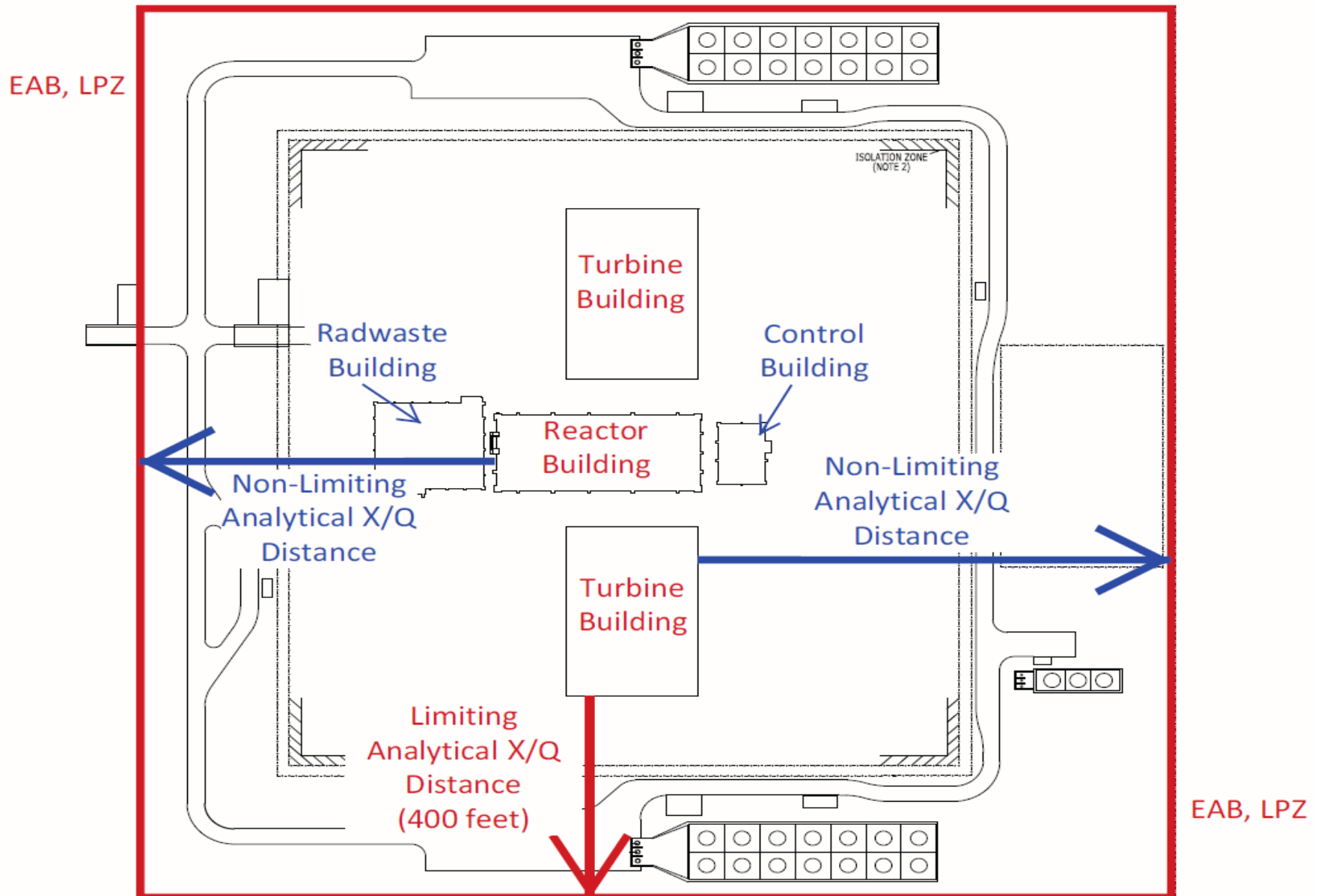
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# Overview of Different Boundaries

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- Restricted Area Boundary
  - 10 CFR 50 Appendix I normal releases
- Low Population Zone (LPZ) and Exclusion Area Boundary (EAB)
  - 10 CFR 50.34(a)(1) and 52.17(a)(1) accident releases
- Emergency Planning Zone (EPZ)
  - Independent of DCA

**Figure 2.3-1: Limiting Analytical Distance to EAB and LPZ Outer Boundary**



November 13, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
One White Flint North  
11555 Rockville Pike  
Rockville, MD 20852-2738

**SUBJECT:** NuScale Power, LLC Submittal of Presentation Materials Entitled "ACRS Subcommittee Presentation: NuScale Source Term Methodology Application," PM-1119-67928, Revision 0

The purpose of this submittal is to provide presentation materials to the NRC for use during the upcoming Advisory Committee on Reactor Safeguards (ACRS) NuScale Subcommittee Meeting on November 20, 2019. The materials support NuScale's presentation of the "Accident Source Term Methodology" topical report.

The enclosure to this letter is the nonproprietary version of the presentation titled "ACRS Subcommittee Presentation: NuScale Source Term Methodology Application," PM-1119-67928, Revision 0.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Carrie Fosaaen at 541-452-7126 or at [cfosaaen@nuscalepower.com](mailto:cfosaaen@nuscalepower.com).

Sincerely,



Zackary W. Rad  
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Enclosure: "ACRS Subcommittee Presentation: NuScale Source Term Methodology Application," PM-1119-67928, Revision 0

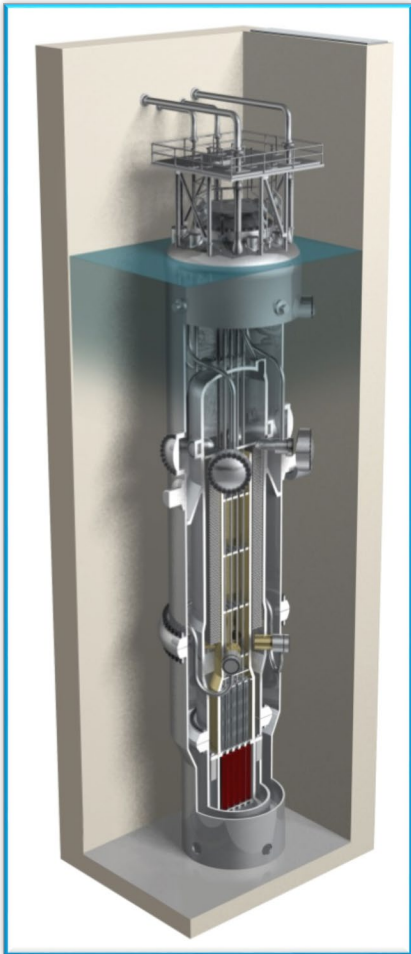
**Enclosure:**

“ACRS Subcommittee Presentation: NuScale Source Term Methodology Application,” PM-1119-67928,  
Revision 0

# ACRS Subcommittee Presentation

## NuScale Source Term Methodology Applications

*November 20, 2019*





# Presenters

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**Mark Shaver**

Radiological Engineering Supervisor

**Paul Guinn**

Radiological Safety Analyst

**Carrie Fosaaen**

Licensing Manager

**Jim Osborn**

Licensing Engineer

**Gary Becker**

Regulatory Affairs Council

# Agenda

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- Source-term-related open items
- Accident source terms applications
- Other Topics

# Acronyms

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Term	Definition
AST	accident source term
BTP	branch technical position
CDE	core damage event
CDST	core damage source term
CR	control room
DBA	design basis accident
DBFFF	design basis failed fuel fraction
DBST	design basis source term
EQ	environmental qualification
ESF	engineered safety feature
FFF	failed fuel fraction
MHA	maximum hypothetical accident
PAM	post-accident monitoring
PAS	post-accident sampling
PSCT	pool surge control tank
TID	total integrated dose

# Source-Term-Related Open Items

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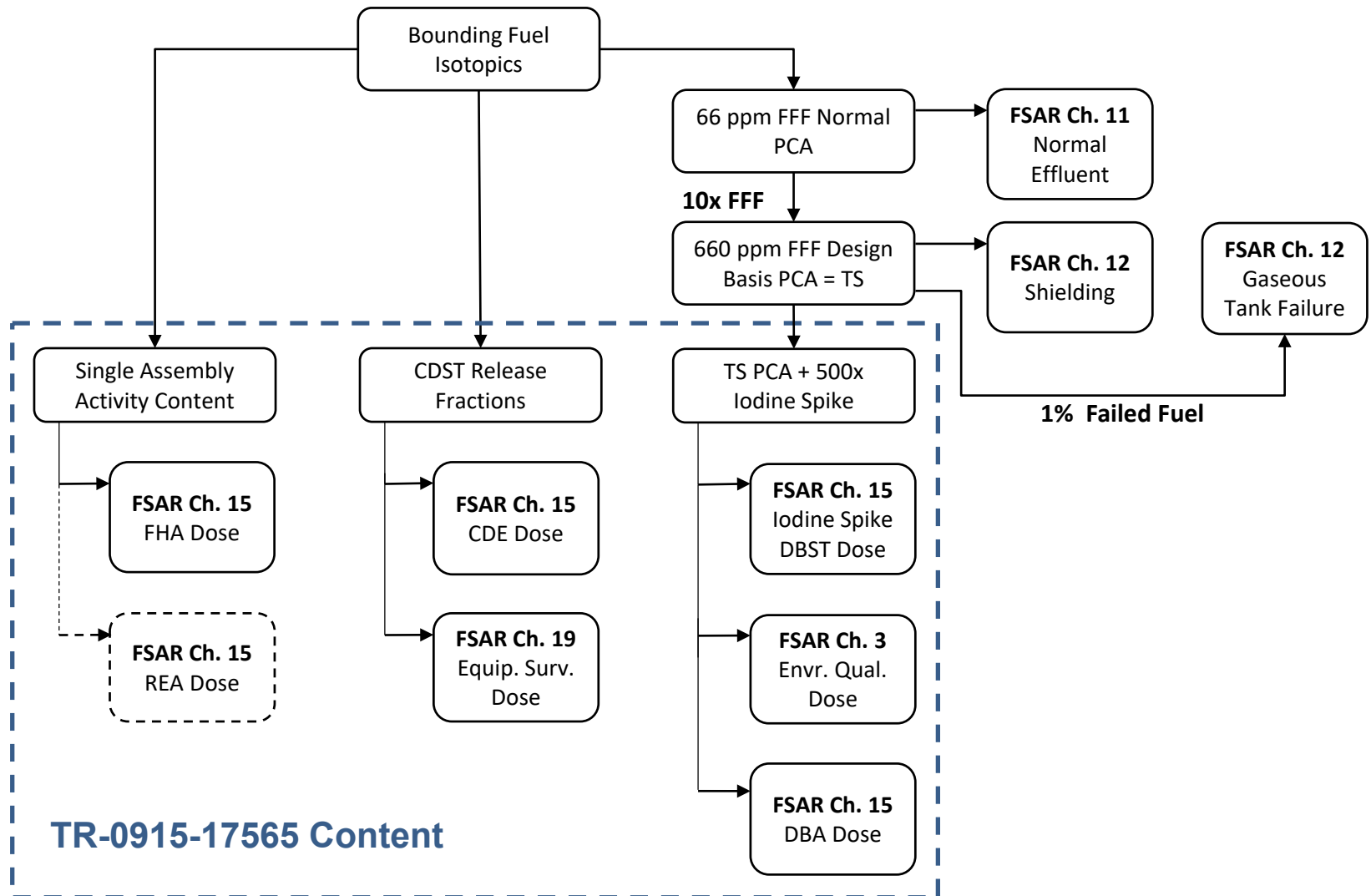
- FSAR Chapter 2
  - Open Item 02.03.04-1: staff evaluation to determine if TR-0915-17565 is acceptable for calculating accident offsite  $\chi/Q$  values
- FSAR Chapter 3
  - RAI 8837, multiple questions: staff request for clarification of TID calculation methodology for DCA Part 2, Appendix 3C, Table 3C-8
- FSAR Chapter 11
  - RAI 9161, Question 11.01-1: staff evaluation of DBFFF as application in source terms for radiation shielding, ventilation systems, and radiation zoning
  - RAI 9253, Question 11.01-2: staff request for inclusion of COL Item 11.2-3: evaluation of PSCT for BTP 11-6

# Source-Term-Related Open Items

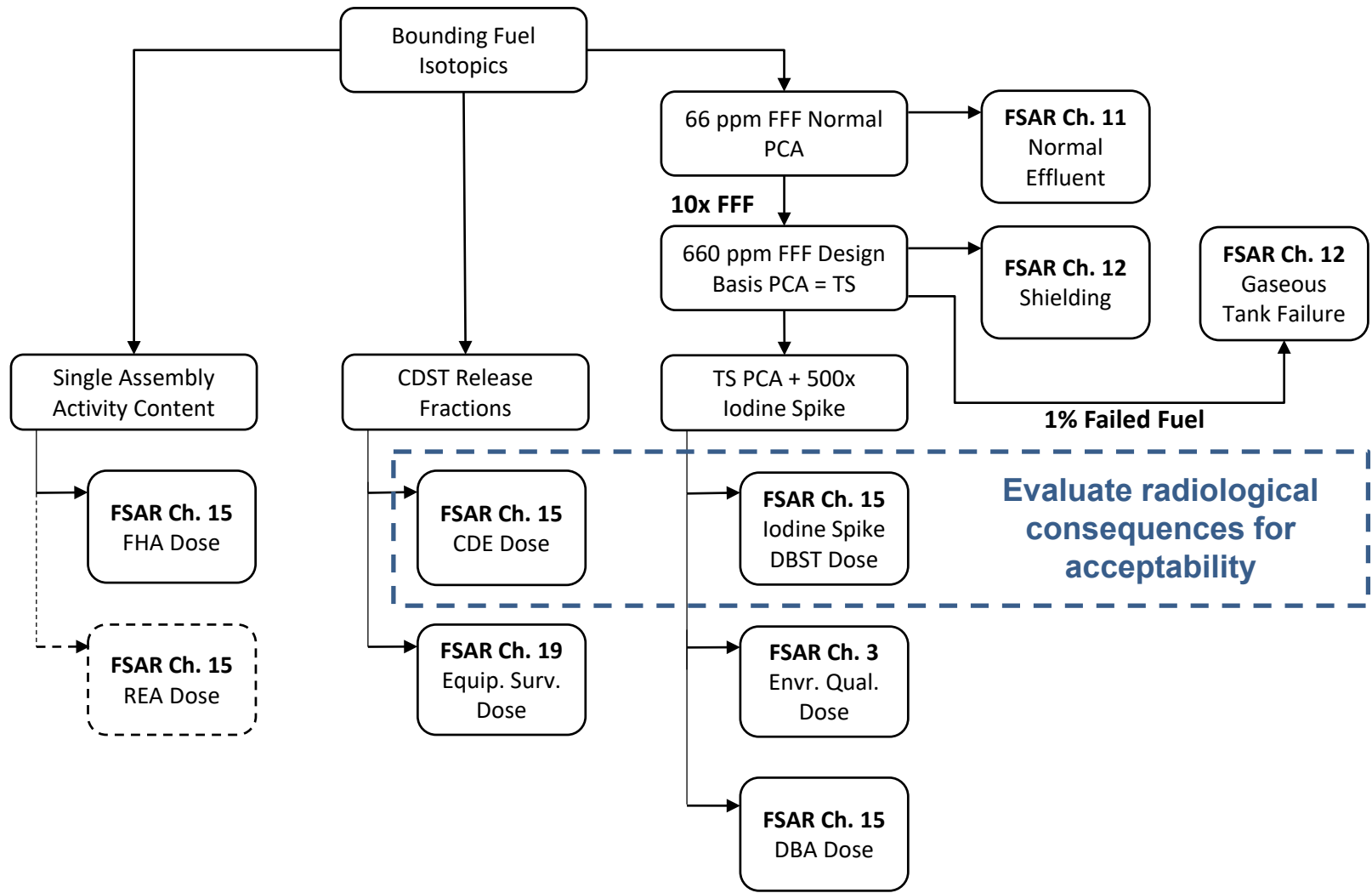
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- FSAR Chapter 12
  - Multiple items
- FSAR Chapter 13
  - RAI 9825, Question 13.03-1: staff evaluation of process sampling system
- FSAR Chapter 15
  - Open Item 15.0.2-6: staff review of the use of ARCON96, STARNAUA, and  $\text{pH}_T$  as part of NuScale methodology (described in TR-0915-17565) for performing DBA radiological consequence analyses
- FSAR Chapter 19
  - Open Item 19.2.4-1: Possible inadequate description of equipment survivability in Ch.19; addressed by Ch. 19 revision and RAI 9705

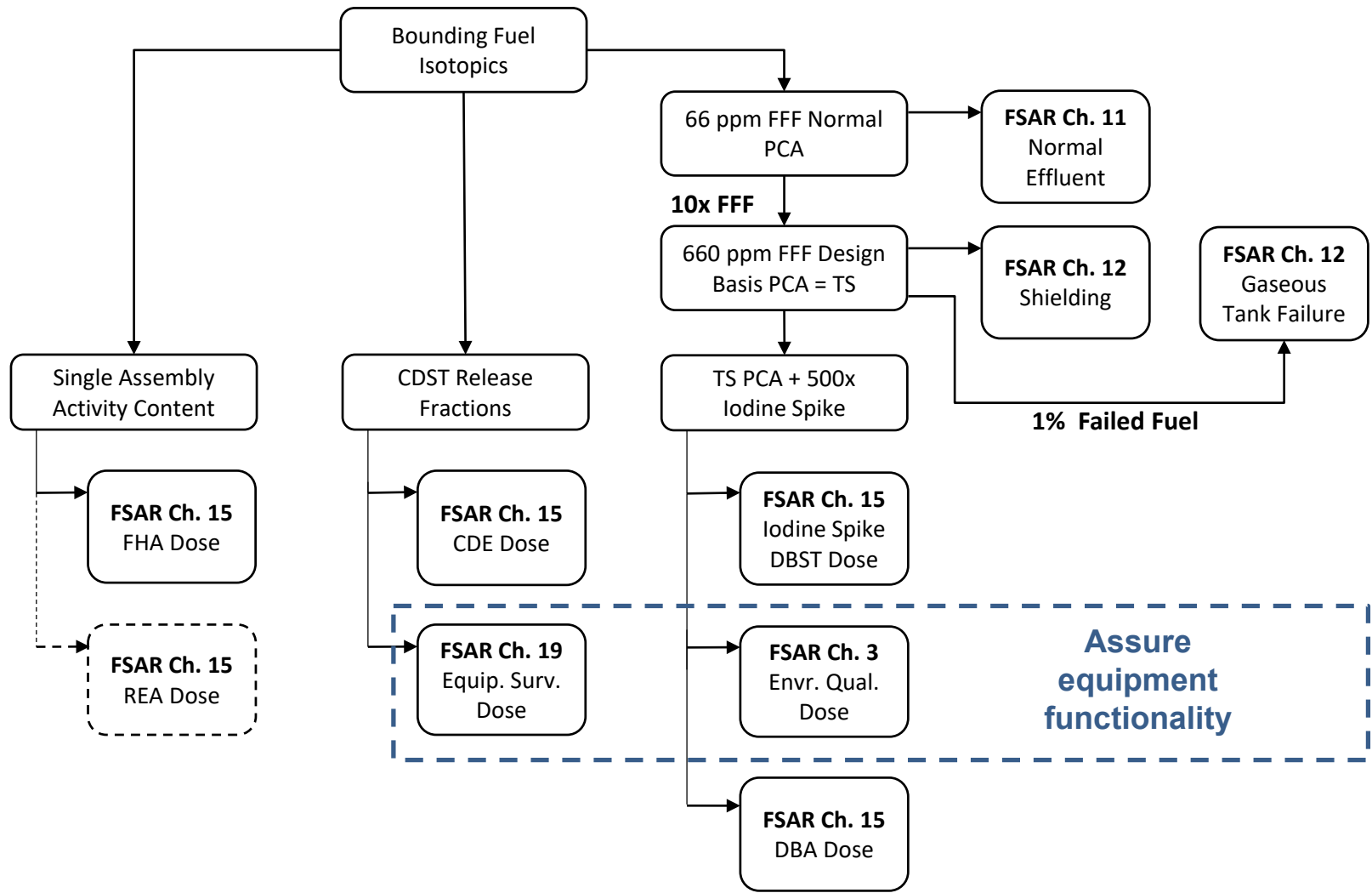
# Source Term Overview



# Source Term Overview



# Source Term Overview





# Chapter 2 AST Application

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- In general, NuScale's site parameters are consistent with past applicant precedents and the EPRI ALWR URD.
- Notable differences are
  - Much smaller site footprint
    - Less atmospheric dispersion
- Atmospheric dispersion (X/Q) methodology based on ARCON96
  - NuScale site boundary (~140m) vs traditional LWR (~800-6000m)
  - ARCON96 used in control room X/Q analyses is closer to NuScale distances and empirically proven to produce more accurate results than PAVAN at shorter distances
  - Methodology described in AST LTR

# Chapter 3 Normal EQ Dose

---

- FSAR Appendix 3C describes environmental qualification (EQ) program for qualifying equipment
  - Normal operation dose for EQ derived from direct gamma emitted by design basis source term (~6-7 failed rods/core)
  - Integrated dose for conservative 60-year equipment life
- Environmental Qualification (EQ) program includes equipment in 10 CFR 50.49 scope: safety-related electric equipment and certain PAM equipment specified in RG 1.97

# Chapter 3 Accident EQ Dose

---

- Accident EQ dose for FSAR Appendix 3C derived from both gamma and beta emitters from design basis source term (~6-7 failed rods/core + iodine spike)
- AST LTR Rev. 4 expanded scope to provide accident EQ dose methodology
  - Iodine spike DBST is a design basis event and thus addressed by EQ per 10 CFR 50.49
  - CDST is a beyond design basis event, and thus beyond the scope of EQ
    - Per SECY-90-016: stringent safety-related requirements, including 10 CFR 50.49, were not “commensurate with the importance of the safety functions to be performed” during severe accident mitigation. Equipment survivability applied instead.

# Chapter 11 Source Terms

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- Two source term models are developed for both primary and secondary coolants:
  - Design Basis and Normal Effluent (“Realistic”) coolant source terms have three components:
    - Water activation products
      - » Calculated from first principles
      - » The same concentration for both Normal Effluent and Design Basis
    - Corrosion activation products
      - » Utilized ANSI 18.1-1999, adjusted to NuScale plant parameters
      - » The same concentration for both Normal Effluent and Design Basis
      - » The only component that strictly used the regulatory guidance provided
    - Fission products
      - » Developed using first principles physics in SCALE 6.1 for core inventory

# Chapter 11 Source Terms (cont.)

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- Normal Effluent (“Realistic”) source term fission products
  - <1 failed rod per core (66 ppm) failure rate is assumed
  - Supported by industry experience from large PWRs, which shows much improvement since the 1970s
    - » 90-95% of US LWRs are zero-defect since 2010
  - Industry data (1987-2010) shows that most failures (90%) are due to grid-to-rod fretting (77%) and debris (13%)
    - » NuScale uses natural circulation, which mitigates these mechanisms
    - » Technical Report TR-1116-52065, Rev. 1
- Design Basis source term fission products
  - ~6-7 failed rods per core (660 ppm) failure rate is assumed
    - » 10x normal effluent source term
    - » Also, supported by Tech Spec 3.4.8 value based on this fuel failure rate

# Chapter 12 Application

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- 12.2 Source Terms
  - Chapter 11 Design Basis Source Terms (660 ppm) used for normal operations design and shielding
  - AST Design Basis Accident Iodine Spike source term for equipment qualification evaluations
- 12.3 Radiation Protection Features of the design accounting for Design Basis Source Terms (660 ppm)
- 12.4 Dose Assessments are informed by Design Basis Source Terms (660 ppm)

# Chapter 15 Design Basis Events

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- Small line break outside containment (FSAR § 15.0.3.8.1)
  - Iodine-spiked primary source
- Steam generator tube failure (FSAR § 15.0.3.8.2)
  - Iodine-spiked primary source
- Main steam line break (FSAR § 15.0.3.8.3)
  - Iodine-spiked primary source
- Rod ejection accident (FSAR § 15.0.3.8.4)
  - Damaged fuel source
- Fuel handling accident (FSAR § 15.0.3.8.5)
  - Damaged fuel source
- Iodine spike DBST (FSAR § 15.0.3.8.6)
  - Iodine-spiked primary source

# Chapter 15 DBE Dose Results

Event	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
Iodine Spike Design Basis Source Term (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Iodine Spike Design-Basis Source Term (coincident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	<0.01
Main Steam Line Break (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	<0.01
	CR	5.0	0.01
Main Steam Line Break (coincident iodine spike)	EAB	2.5	<0.01
	LPZ	2.5	<0.01
	CR	5.0	<0.01
Steam generator tube failure (pre-incident iodine spike)	EAB	25.0	0.08
	LPZ	25.0	0.08
	CR	5.0	0.20
Steam generator tube failure (coincident iodine spike)	EAB	2.5	<0.01
	LPZ	2.5	<0.01
	CR	5.0	<0.01
Primary coolant line break	EAB	6.3	0.02
	LPZ	6.3	0.04
	CR	5.0	0.08
Fuel handling accident	EAB	6.3	0.55
	LPZ	6.3	0.55
	CR	5.0	0.89



# Chapter 15 Core Damage Event

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- Core damage event:
  - A special event (beyond design basis) with radionuclides from core damage released into an intact containment
  - Postulated to enable deterministic evaluation of the response of the facility and site to the maximum hypothetical accident (i.e. a “substantial meltdown” event)
- Five surrogate accident scenarios derived from intact-containment internal events in the Level 1 PRA were used to establish the CDST
  - The minimum onset time for fission product release from the gap, the release duration associated with minimum release onset time, and the median value of the release fractions determined from the spectrum of surrogate accident scenarios are used as the CDST.

# Chapter 15 CDST Dose Results

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Event	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
Core Damage Event	EAB	25.0	0.63
	LPZ	25.0	1.37
	CR	5.0	2.14

# Chapter 19 Application

---

- Functionality of equipment that is necessary for mitigating a severe accident is, commensurate with the importance of the safety functions to be performed, reasonably assured by demonstrating equipment survivability
  - The core damage source term (CDST) is considered in the equipment survivability evaluation to demonstrate necessary equipment is available in a severe accident for its required functional duration
- Following a severe accident, containment integrity and post-accident monitoring must be maintained
  - Post-accident monitoring is not relied upon for mitigating severe accidents, but is intended only to provide information on severe accident conditions.

# Other Topics

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## 1. PAS exemption request

- NuScale requested exemption from 10 CFR 50.34(f)(2)(viii) based on alternate means to assess core damage.

## 2. Application of GDCs to beyond design-basis accidents

- In general, NuScale maintains that 10 CFR 50 Appendix A does not apply to severe, beyond design-basis accidents, unless specifically invoked (e.g., GDC 19 via NUREG-0737, Item II.B.2 – 10 CFR 50.34(f)(2)(vii)).

## 3. Radiological consequence contribution from potential leaks in non-safety hydrogen monitoring lines in 10 CFR 52.47(a)(2)(iv) analysis.

- NRC RAI 9690 Question No. 01.05-40
- NuScale Response submitted 9/5/2019

# 1. Post-Accident Sampling Exemption Request

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- One of several TMI-related requirements that expressly considers a core melt source term.
- PAS capability is not needed because NuScale design ensures the capability to assess core damage by other means.
  - Under the Bioshield radiation monitors
  - Core exit temperature indicators
- Advantages
  - Source term remains contained within module
  - Reduced opportunity for leaks and spills
  - Reduced operator doses
- Exemption request is found in DCA Part 7, Chapter 16

## 2. Applicability of GDC 19

---

- The GDCs define and establish acceptance criteria for design basis events for LWRs.
  - 68 FR 54123, “Combustible Gas Control in Containment:  
“The postulated accidents used in the development of [the GDCs] are normally design-basis accidents. The NRC believes it is not appropriate to address severe accident design requirements in the General Design Criteria.”
  - For Large LWRs, the “design basis LOCA” radiological consequences assessment (FSAR 15.6.5) is based on a core damage event. Thus, the control room dose limits of GDC 19 apply to this assessment.

## 2. Applicability of GDC 19 (cont.)

---

- TMI Action Item II.B.2 (10 CFR 50.34(f)(2)(vii))
  - Required a design review to ensure adequate shielding for operator access and component protection for “degraded core” accidents “beyond the design basis.”
  - Assured the design and licensing basis of then-operating plants was in-line with current guidance. The RG 1.3 and 1.4 source terms and the operator dose limits of GDC 19 were prescribed.
  - Thus, the operator access requirements of II.B.2 are redundant to GDC 19 under current guidance for Large LWRs.
- NuScale’s approach classifies the CDE as a beyond design basis event
  - GDC 19 does not apply
  - But TMI Item II.B.2--which expressly addresses core damage events--prescribes the GDC 19 operator dose criteria for the CDE

# 3. Hydrogen Monitoring System Leak

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- NRC issued RAI 9690 requesting that NuScale postulate a leak from the hydrogen monitoring system and analyze the resultant radiological consequences.
- NuScale believes accounting for such leakage in the CDST analysis is unnecessary to reasonably assure adequate protection
  - Hydrogen monitoring capability is provided only for severe accidents and is not germane to any DBA.
  - If the system leaks excessively, operators will isolate the leak, but this would be an unplanned and unexpected post-accident activity, and therefore does not require a separate dose analysis.
  - Such potential leakage contributors are not included in guidance or past applications, apparently due to low risk.



# 3. Hydrogen Monitoring System Leak

---

- NuScale followed established guidance provided in RG 1.183, Appendix A to evaluate offsite doses following CDE.
- NRC guidance excludes these known potential leakage pathways from the design basis accident radiological consequence analysis.
  - RG 1.183 and SRP 15.6.5 include only containment and ESF system leakage for PWRs.
  - TMI Item II.B.2: “Leakage of systems located outside of containment need not be considered for” the shielding review; leakage from those systems is “treated under Item III.D.1.1.”
  - Item III.D.1.1: requires a Leakage Control Program to “minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.”

# 3. Hydrogen Monitoring System Leak

---

- NuScale's CDST evaluation is a beyond design basis event analysis.
- NuScale does not believe potential leakage represents a significant safety risk.
- Therefore, the existing guidance is adequate for NuScale to provide reasonable assurance that the worker and public are protected.
- However, NRC staff have stated that they cannot reach a finding on the issue, and therefore intend to exclude the hydrogen monitoring leakage from issue resolution in the NuScale DC rulemaking.

# Questions?

---

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# **Accident Source Term Methodology Related Topics**

**NuScale Design Certification Application Review**

**Presentation to the ACRS Subcommittee**

November 20, 2019

# Staff Review Team

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- ◆ Ron LaVera, NRR
- ◆ Amanda Marshall, NSIR

## ➤ Project Managers

- ◆ Getachew Tesfaye – Chapter PM
- ◆ Greg Cranston – Lead PM

# Contents

- Hydrogen and Oxygen Monitoring Radiological Review
  - ◆ Regulatory Basis
  - ◆ Post-accident hydrogen and oxygen monitoring design overview
  - ◆ Dose to a worker establishing monitoring
  - ◆ Leakage
  - ◆ Resolution
- Post Accident Sampling (PAS) Exemption
  - ◆ Exemption Request
  - ◆ Staff Review of Exemption Request
  - ◆ Conclusion
- Additional Technical Topic of Interest
  - ◆ Realistic and Design Basis Failed Fuel Fraction (DBFFF) Source terms
  - ◆ Equipment Survivability
  - ◆ SER Section 15.0.3 Radiological Consequences of Accidents

# Hydrogen and Oxygen Monitoring Radiological Review

## Regulatory Basis

- 10 CFR 50.44(c)(4) – requires the capability to perform beyond design basis accident hydrogen and oxygen monitoring
- 10 CFR 50.34(f)(2)(vii) – requires that applicants perform post-accident radiation and shielding design reviews and design as necessary to permit adequate access
- 10 CFR 50.34(f)(2)(xxvi) – requires for leakage control and detection for systems outside containment that contain (or might contain) accident source term
- 10 CFR 50.34(f)(2)(xxviii) – requires that applicant's evaluate potential pathways that may lead to control room habitability problems.
- 10 CFR 52.47(a)(2)(iv) – requires that the barriers that must be breached as a result of accident be assessed before a release of radioactive material can occur. The expected demonstratable leak rate from containment should be considered.



# Hydrogen and Oxygen Monitoring Radiological Review

## Post-accident Monitoring Design

- Requires the containment to be unisolated and the use of three systems
  - ♦ Containment Evacuation System (CES)
  - ♦ Sampling System
  - ♦ Containment flood and drain system
- Manual actions to override the containment isolation signals at hydraulic control skids to unisolate containment to establish monitoring
  - ♦ Steam gallery (100' elevation)
  - ♦ Mechanical equipment area (126' elevation)

# Hydrogen and Oxygen Monitoring Radiological Review

## Dose to a Worker Establishing Monitoring

- 10 CFR 50.34(f)(2)(vii) requires the ability to access important areas in the post-accident radiation environment
- Containment isolation valves are in series
  - ♦ Allows workers to access potentially higher dose rate room (steam gallery area) before fluid is removed from containment
- Applicant and staff performed dose analyses considering containment and under the bioshield radiation sources and found dose to worker to be less than 5 rem

# Hydrogen and Oxygen Monitoring Radiological Review

## Leakage from monitoring

- COL Item 9.3-1
  - ♦ Leakage control program
- However, the applicant did not specify an acceptable amount of leakage and did not assess the leakage from these systems in the main control room or offsite dose assessment
- Applicant specified that leakage from these systems need not be considered in the main control room or offsite dose assessment for various reasons
  - ♦ RG 1.183
  - ♦ Similarities to other designs

# Hydrogen and Oxygen Monitoring Radiological Review

## RG 1.183 and comparison to other designs

- RG 1.183 specifies that ESF system leakage should be considered in the offsite and MCR dose analysis but that the dose from containment purge need not be considered if containment purge would only occur following severe accidents
  - ♦ The combustible gas monitoring systems are not ESF systems in the NuScale design. However, the staff does not believe performing hydrogen and oxygen monitoring is equivalent to purging containment
- Staff reviewed several other designs with hydrogen and oxygen monitors outside of containment and found significant differences between these designs and the NuScale design

# Hydrogen and Oxygen Monitoring Radiological Review

## Comparison to other designs

- Combustible gas monitoring system for ESBWR
  - ♦ single system
  - ♦ redundant
  - ♦ safety-related
  - ♦ seismic category 1
  - ♦ system isolation valves can be controlled from the main control room
- NuScale combustible gas monitoring system.
  - ♦ three systems
  - ♦ not redundant
  - ♦ not safety-related
  - ♦ seismic category 3
  - ♦ system isolation valves cannot all be controlled from the main control room

# Hydrogen and Oxygen Monitoring Radiological Review

## Comparison to other designs (continued)

- Additional reasons that leakage from the monitoring systems may be more significant in the NuScale design
  - ♦ Compared to large LWRs, the NuScale containment is much smaller, resulting in a lower total leakage rate to the environment
  - ♦ The difference in containment size to reactor power ratio results in the potential for a more concentrated source term of leaked fluid than large LWRs with a similar percentage of fuel damage
    - The containment of one NuScale module is about 850 times smaller than the APR1400 containment, but the power level is only about 25 times less
  - ♦ Some of the piping sizes of these systems in the NuScale design appears larger than that identified by other designs and has connections to numerous other systems, which presents the potential for additional leakage paths.
    - Piping ranges from 4" to 3/8" in the monitoring loop

# Hydrogen and Oxygen Monitoring Radiological Review

## Potential Significance of Leakage

- Staff calculations using the limited amount of available information indicates the potential for leakage from these system to be a significant contributor to offsite and MCR dose limits and could potentially result in exceeding dose limits
- The applicant has not demonstrated a capability to re-isolate the systems, so it is unclear if unacceptable leakage can be mitigated
  - ♦ Potentially very high dose rates at the hydraulic skids due to direct radiation and leakage from the systems

# Hydrogen and Oxygen Monitoring Radiological Review

## Resolution

- Since the staff does not have assurance that the MCR and offsite dose limits will be met, the staff intends to specify in the NuScale design certification rule that the COL applicant will demonstrate that the leakage will not result in exceeding MCR or offsite dose limits and/or include design features to provide assurance that the leakage will not result in exceeding MCR or offsite dose limits
  - ♦ If manual actions outside of the control room are required, the COL applicant will demonstrate the ability to perform the actions in the radiation environment



## **PAS Exemption Request**

- 10 CFR 50.34(f)(2)(viii) requires that applicants provide the capability to promptly obtain and analyze post-accident samples from the reactor coolant system and containment atmosphere. Materials to be analyzed include the following:
  - ♦ Certain radionuclides that are indicators of the degree of core damage
  - ♦ Hydrogen in the containment atmosphere
  - ♦ Dissolved gases
  - ♦ Chloride
  - ♦ Boron concentrations
- DCA Part 7, Exemption 16
  - ♦ Request to be exempt from post-accident sampling (10 CFR 50.34(f)(2)(viii)) in entirety, since the NuScale plant design meets the underlying purpose of the rule by ensuring the capability to assess the presence and extent of core damage during an accident by other means

## **PAS Exemption - Radionuclides**

- The regulation specifies the purpose of sampling for radionuclides is to assess the degree of core damage
  - ♦ The NuScale design includes radiation monitors under the bio-shield and core exit thermocouples which can be used to assess core damage
  - ♦ Based on this, the staff found it acceptable to be exempt from sampling radionuclides
- The staff notes that 10 CFR 50.47(b)(9) requires that the COL applicant must address that they have adequate methods, systems, and equipment for assessing and monitoring actual or potential consequences of a radiological emergency but this can be accomplished by other means than sampling

## **PAS Exemption - Containment Hydrogen**

The purposes of sampling hydrogen in the atmosphere is to ensure that hydrogen and oxygen concentrations do not support combustion that could challenge the containment.

NuScale requests an exemption from sampling hydrogen during post-accident conditions.

- The staff agrees with the exemption because NuScale has the capability to monitor hydrogen and oxygen concentrations as required by 50.44(c)(4) as discussed in Section 6.2.5 of this SER.

## **PAS Exemption - Dissolved Gases**

Sampling for dissolved gases has generally been required to ensure that natural recirculation is not inhibited.

NuScale requests an exemption from sampling for dissolved gases during post-accident conditions.

- The staff agrees with the exemption because of NuScale design specific considerations of the NuScale ECCS system, as documented in the staff's evaluation of the exemption for high point vents in Section 5.4.5 of the SER.

## **PAS Exemption - Chlorides**

NuScale requests an exemption from sampling chlorides during post-accident conditions.

Chlorides typically are introduced from external sources. The staff finds that NuScale exemption acceptable based on the following:

- NuScale's minimal use of chlorinated cable insulation.
- Monitoring chloride concentration in makeup inventory source during normal operation.
- NuScale limits chloride by monitoring and control of reactor water chemistry based on industry guidelines contained in EPRI, "PWR Primary Water Chemistry Guidelines," during normal operations.

## **PAS Exemption - Boron**

The purpose of sampling boron concentration of the RCS is to ensure that there is adequate shutdown margin to achieve and maintain safe shutdown.

NuScale requests an exemption from sampling for boron during post-accident conditions.

- The staff agrees with the exemption because The only Type B variables identified in the NuScale design that provide direct indication and are used to assess the process of accomplishing or maintaining reactivity control are neutron flux and core inlet and exit temperature.
- The transient and accident analyses described in DCD Part 2, Tier 2, Chapter 15 does not rely on the measurement of RCS boron concentration and is not expected to be necessary to implement the plant operating procedures and maintain the plant critical safety functions.

## **PAS Exemption Conclusion**

- Since equivalent information to that provided by sampling can be provided by other means, the staff determined that post-accident sampling need not be required. Therefore, the staff approves the exemption from post-accident sampling for the NuScale design
- The staff did not assess the radiological dose consequences to a worker obtaining and analyzing post-accident samples and does not consider the ability to take samples following an accident within the design
- This does not exempt a COL applicant from complying with the emergency preparedness planning standard in 10 CFR 50.47(b)(9)

## **Realistic and Design Basis Failed Fuel Fraction (DBFFF) Source terms**

### ***SER Section 11.1***

- **Open Item 11.01-1 (RAI 9161, Question 11.01-01):** Changes to realistic and design basis failed fuel fraction (DBFFF) source terms. The response to this RAI resulted in the change to all listed effluent release results and component inventories. Response revised the following information:
  - Revision to all component source terms contained in chapter 11 and 12.
  - Revision to the calculated doses for compliance with 10 CFR 50 Appendix I.
  - Revised NuScale Technical Report on the Effluent Release Methodology.
- The staff reviewed all changes related to DBFFF through out chapters 11 and 12 and found the source term changes to be acceptable.



# Equipment Survivability

Following a severe accident, the two functions that must be maintained are:

- containment integrity, and
- post-accident monitoring, to provide information on severe accident conditions in containment

To meet criteria of SECY 90-016,

- equipment required to mitigate severe accidents must be identified.
- Function, time duration and the environmental conditions of pressure, temperature, humidity, and radiological dose for which this function is required must also be identified.
- Environmental conditions must also include exposure to the conditions created by hydrogen burning, per 10 CFR 50.44(c)(3).

# Equipment Survivability

NuScale's methodology for assuring equipment survivability (ES) in terms of post-accident radiological dose involves comparing the severe accident (CDST) dose with the equipment qualification (EQ) design basis source term (DBST) dose.

- If the EQ dose is larger, no further quantitative survivability assessment is performed.
- If the severe accident dose is larger, qualitative assessments, testing, and/or additional analyses may need to be performed to assure survivability.

NuScale has evaluated both the EQ and the severe accident doses for each component or variable identified in FSAR Table 19.2-11, Equipment Survivability List. The results of this evaluation can be found in the non publicly available version of the audit report, ML19308A944.

## **Equipment Survivability**

Specifically identified are the components whose severe accident dose exceeds, or potentially could exceed, the EQ dose:

- Electrical Penetration Assemblies (EPA)
- ECCS reset valves
- CES, CFDS containment isolation valves (CIV)
- Combustible gas monitors
- Containment gas sample pump
- Inside the Bioshield Radiation Monitor

At the design certification stage specific components have not yet been selected. Once selected, the COL applicant will confirm or update the EQ and the severe accident doses for all components identified in FSAR Table 19.2-11, according to COL Item 19.1-8.

## **SER Section 15.0.3**

# **Radiological Consequences of Accidents**

### **Radiological Consequence Analyses**

- Implemented dose analysis and source term methodology in NuScale topical report TR-0915-17565, “Accident Source Term Methodology”
  - ◆ TR reviewed concurrently
- Design specific input to TR methodology
- Hypothetical atmospheric dispersion factors (SER Section 2.3.4)

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Accidents and Events Analyzed**

- Failure of small lines carrying primary coolant outside containment
- Steam generator tube failure (SGTF)
- Main steam line break (MSLB) outside containment
- Fuel handling accident (FHA)
- Iodine spike DBST
- Core damage event (CDE)

# **SER Section 15.0.3**

## **Radiological Consequences of Accidents**

### **Regulatory Criteria**

- 10 CFR 52.47(a)(2)(iv) offsite dose: **25 rem TEDE**
  - ◆ Exclusion area boundary (EAB), any 2-hour period
  - ◆ Low population zone (LPZ), duration of passage of plume
  - ◆ DSRS 15.0.3 – accident-specific dose criteria
    - Fraction of criterion based on likelihood of accident
- PDC 19 control room habitability: **5 rem TEDE**
  - ◆ 10 CFR 50.34(f)(2)(xxviii)
- Technical support center (TSC) habitability: **5 rem TEDE**
  - ◆ NUREG-0696 – same radiological habitability as control room
  - ◆ 10 CFR 50.47(b)(8) and (b)(11)
  - ◆ Paragraph IV.E.8 of Appendix E to 10 CFR Part 50

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Dose Acceptance Criteria TEDE (rem)**

<b>Accident</b>	<b>EAB (2-hr)</b>	<b>LPZ (duration)</b>	<b>CR/TSC (duration)</b>
Small line break	2.5	2.5	5
SGTF coincident iodine spike	2.5	2.5	5
SGTF pre-incident iodine spike	25	25	5
MSLB coincident iodine spike	2.5	2.5	5
MSLB pre-incident iodine spike	25	25	5
FHA	6.3	6.3	5
Iodine spike DBST (both cases)	25	25	5
CDE	25	25	5

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Control Room Modeling**

##### ➤ Pathways

- ♦ Intake and inleakage to control room envelope (CRE)
- ♦ Direct shine
- ♦ Sky-shine
- ♦ Shine from filters

##### ➤ Two ventilation system operation cases

- ♦ Operation of CRVS in supplemental filtration mode for duration
  - 99% efficiency HEPA and charcoal filters
- ♦ Operation of CRHS for initial 72 hours, then CRVS in supplemental filtration mode for remainder of duration

##### ➤ CR unfiltered inleakage

- ♦ CRVS operation: 174 cfm through CRE + 5 cfm ingress/egress
- ♦ CRHS operation: 10 cfm through CRE + 5 cfm ingress/egress



## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **TSC Modeling**

- TSC not served by CRHS
  - ◆ TSC function can be moved if TSC uninhabitable
- Operation of CRVS in supplemental filtration mode for duration
  - ◆ 99% efficiency HEPA and charcoal filters
- TSC unfiltered inleakage
  - ◆ 56 cfm through ventilation envelope + 10 cfm ingress/egress

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Core Inventory**

- Based on 102% of rated core thermal power
  - ◆ NuScale fuel design
- Calculated using SCALE code
  - ◆ TRITON for burnup-dependent cross sections
  - ◆ ORIGEN-ARP and ORIGEN-S for generation and depletion
- Staff confirmatory calculation using ORIGEN-ARP
  - ◆ Modeled as PWR 17x17
  - ◆ Using fuel design information described in FSAR
    - Enrichment, maximum burnup, cycle length
  - ◆ Similar results as FSAR table
- Calculation of core inventory consistent with TR methodology and guidance in RG 1.183

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Primary Coolant Activity Concentration**

- FSAR Table 11.1-4 primary coolant design basis source term adjusted to the design specific activity limits for accident initial conditions
  - ◆ Based on design basis failed fuel fraction
- Technical Specification (TS) 3.4.8 specific activity limits
  - ◆  $3.7\text{E-}02 \text{ } \mu\text{Ci/gm DE I-131}$  (equilibrium)
  - ◆  $2.2 \text{ } \mu\text{Ci/gm DE I-131}$  (maximum)
  - ◆  $10 \text{ } \mu\text{Ci/gm DE Xe-133}$
- Consistent with RG 1.183
- Consistent with TR methodology
  - ◆ Secondary coolant activity concentration not modeled

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Accidents Without Core Damage**

Small line break

SGTF

MSLB outside containment

FHA

Iodine spike DBST

- FSAR dose analyses follow TR methodology
  - ◆ Design specific input values taken from relevant FSAR system descriptions
  - ◆ Design specific core inventory or primary coolant activity concentration
- Dose results meet accident-specific dose criteria with margin

## **SER Section 15.0.3**

# **Radiological Consequences of Accidents**

## **Rod Ejection Accident**

- No fuel damage predicted for limiting REA
- No dose analyses provided
  - ◆ Consistent with TR methodology
  - ◆ REA radiological consequences bounded by other events

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Core Damage Event**

- Core damage source term (CDST) implemented TR methodology using design specific information
- CDST scenario selection
  - ◆ Based on NuScale SMR PRA, internal events
  - ◆ 5 surrogate scenarios
    - Various failures of ECCS, with decay heat removal system available
    - Intact containment
- Used MELCOR to estimate release timing and magnitude for each scenario
- Followed TR methodology to characterize CDST
  - ◆ Release onset and duration from scenario with minimum time to core damage
  - ◆ Core release fractions taken as median of scenarios

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Iodine Modeling in CDE**

- Iodine chemical form
  - ◆ 95% CsI
  - ◆ 4.85% elemental
  - ◆ 0.15% organic
- pH<sub>T</sub> calculations
  - ◆ Modeled over 30-day period
  - ◆ Post-accident pH<sub>T</sub> inside containment between 6.0 and 7.0
  - ◆ Iodine re-evolution assumed negligible, consistent with TR
- No credit for elemental iodine removal in containment
- Staff found acceptable consistent with TR methodology

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Aerosol Modeling in CDE**

- Aerosol natural deposition in containment
  - ◆ Time-dependent aerosol removal rates calculated using STARNAUA code implementing TR methodology
- Calculation used design-specific input
  - ◆ Thermal hydraulic conditions calculated by MELCOR for surrogate scenario with minimum time to core damage
- Staff found implementation of TR aerosol natural deposition methodology acceptable



## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Containment Leakage Pathway**

- Release from core assumed to be well mixed within containment air volume
- Design basis containment leakage to environment
  - ◆ 0.2% per day (first 24 hours of release)
  - ◆ 0.1% per day (remainder of 30 days)
- Consistent with TR methodology and RG 1.183

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

#### **Combustible Gas Monitoring Pathway**

- 10 CFR 50.44(c)(4) requirement for capability to monitor hydrogen and oxygen concentrations in containment following a significant beyond design basis accident
- NuScale design routes containment atmosphere through systems outside containment
  - ◆ No assurance provided that containment can be re-isolated post-accident to mitigate potential leakage from systems outside containment
  - ◆ CDE analysis does not include dose contribution from this potential release pathway
- Staff is unable to reach a finding that the applicant has evaluated all potential pathways for radioactivity that could lead to control room habitability problems under accident conditions (10 CFR 50.34(f)(2)(xxviii))
- Per the design certification rule, COL applicant will be required to provide additional information to show 10 CFR 50.34(f)(2)(xxviii) is met

## **SER Section 15.0.3**

# **Radiological Consequences of Accidents**

### **Confirmatory Analyses**

- Staff performed independent confirmatory analyses for each of the accidents
  - ◆ TR methodology
  - ◆ Design-specific input from DCA
  - ◆ Control room ventilation system modeling sensitivity
- Staff audited NuScale calculation documents to verify methodology assumptions and input values
- Staff results confirm doses calculated by NuScale are within the applicable accident-specific dose criteria

## **SER Section 15.0.3**

### **Radiological Consequences of Accidents**

### **Conclusion**

- Staff found reasonable assurance that the NuScale SMR design meets 10 CFR 52.47(a)(2)(iv) dose criteria and the accident-specific offsite dose acceptance criteria given in NuScale SMR DSRS Section 15.0.3
- Staff found reasonable assurance that the main control room habitability requirements of PDC 19 are met
- Staff found reasonable assurance that TSC habitability requirements are met

## Abbreviations

CDE	core damage event	PWR	pressurized water reactor
CDST	core damage source term	REA	rod ejection accident
COL	combined operating license	rem	Roentgen equivalent man
CRHS	control room habitability system	RG	regulatory guide
CRVS	normal control room HVAC system	RVV	reactor vent valve
CVCS	chemical and volume control system	SECY	Commission paper
DBFFF	design basis failed fuel fraction	SGTF	steam generator tube failure
DBST	design basis source term	SMR	small modular reactor
DCA	design certification application	SSCs	structures, systems and components
DF	decontamination factor	TEDE	total effective dose equivalent
EQ	environmental qualification	TR	topical report
FHA	fuel handling accident		
HVAC	heating ventilation and air conditioning		
LWR	light water reactor		
MHA	maximum hypothetical accident		
MSLB	main steam line break		
pH <sub>T</sub>	temperature dependent pH		
PAS	Post Accident Sampling		