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

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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654th MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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OPEN SESSION

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THURSDAY

JUNE 7, 2018

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ROCKVILLE, MARYLAND

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The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Michael Corradini, Chairman, presiding.

COMMITTEE MEMBERS:

MICHAEL CORRADINI, Chairman

RONALD G. BALLINGER, Member

DENNIS C. BLEY, Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member

HAROLD B. RAY, Member

JOY L. REMPE, Member
PETER C. RICCARDELLA, Member
GORDON R. SKILLMAN, Member
MATTHEW SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:
ZENA ABDULLAHI

ALSO PRESENT:
BRUCE BAVOL, NRO
MICHAEL BRADBURY, NuScale*
ALLYSON CALLAWAY, NuScale*
TIM DRZEWIECKI, NRO
AZAT GALIMOV, NuScale*
ROBERT GAMBLE, NuScale*
JOSH KAIZER, NRR
BECKY KARAS, NRO
ANDY LINGENFELTER, NuScale*
JEFF SCHMIDT, NRO
JIM TOMKINS, NuScale*
JENNIE WIKE, NuScale*

*Present via telephone

P R O C E E D I N G S

8:30 a.m.

CHAIR CORRADINI: Okay, the meeting will come to order. This is the second day of the 654th meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the Committee will consider the following. NuScale topical reports on codes and method qualification and critical heat flux, future ACRS activities on, of the Planning and Procedures Subcommittee, and preparation of ACRS reports.

The ACRS was established by statute in, and is governed by the Federal Advisory Committee Act, or FACA. As such, this meeting is being conducted in accordance with the provisions of FACA. That means that the Committee can only speak through its published letter reports. We hold meeting to gather information to support our deliberations.

Interested parties who wish to provide comments can contact our offices requesting time after the Federal Register Notice describing the meeting is published. That said, we also set aside ten minutes for extemporaneous comments from members of the public attending or listening in our meetings.

Written comments are also welcome. Ms.

Zena Abdullahi is the Designated Federal Official for the initial portion of this meeting. Portions of the session of the NuScale topical reports on codes and methods qualification and critical influx may be closed in order to discuss and protect information designated as proprietary.

The ACRS section of the US NRC's public website provides our charter bylaws, letter reports, and full transcripts of all full and Subcommittee meetings, including all slides presented at those meetings. We received no written comments or requests to make oral statements from members of the public regarding today's session.

There will be a phone bridge line, and to preclude interruption of the meeting, the phone bridge line will be placed in a listen-in only mode during the presentations and Committee discussion. A transcript of portions of the meeting is being kept, and it is requested that speakers use one of the microphones, identify themselves, and speak with sufficient clarity and volume so they can be readily heard.

I'll note that everybody please check your appliances, make sure that everything's turned off so we don't get beeps or buzzes. And so we'll turn to our first topic, which will be NuScale's topical reports, codes, methods, and qualifications in critical heat flux with Walt Kirchner.

I will note that Member Rempe and Member Ricardella have conflicts in particular parts of these topical reports and will only listen and give information as requested. Walt.

MEMBER KIRCHNER: Thank you, Chairman, I'll keep my remarks brief. We had a meeting of the Subcommittee on May 15. That was a good meeting. And with that, there were some questions and they will be addressed today from members. I'm going to just turn it to Bruce. Bruce.

MR. BAVOL: Yes.

MEMBER KIRCHNER: Go from here.

MR. BAVOL: Okay, my name is Bruce Bavol, I'm a project manager in the Office of New Reactors. We're going to go over briefly in the public portion of the session here just a quick overview of the scope and the timeline for our review.

And again, this is the topic TR-0616-48793, Revision 0, Nuclear Analysis Codes and Methods; and Topical Report 0116-21012, Revision 1, NuScale Power Critical Heat Flux Correlations.

Today, in the public portion like I said, we're going to be going over the staff review timeline, the scope of the review, and the staff conclusion.

The technical contributors today, to my right, Tim Drzewiecki, he was the lead reviewer for this, both topical reports. We also have Becky Karas, who's the Branch Chief of Reactor Systems; Jeff Schmidt, the lead in that branch; and additional support, Josh Kaizer's also here.

For the timeline for the nuclear codes and methods qualification, that topical report came in August 30, 2016, was docketed in our ADAMS, Agencywide Document Access Management System. We only had one request for additional information, RAI 8807. That RAI was dispositioned in the safety evaluation that was provided to the ACRS members.

The staff does plan to issue, if everything goes according to plan, the final SER in late July 2018 and publish a -A approved version of the topical report in October 2018. And with that, I'll let Tim, turn it over to Tim to go over the scope.

MR. DRZEWIECKI: Okay. So in terms of how these reviews were done, this slide pertains to the codes and methods review, is that findings made for this topical is that you can use this methodology, which is then going to be used later, or is used as part of a design certification application, to make findings with several other GDCs.

We requested the use of the CMS5 software. Some base values, the NRFs, which are used to account for the uncertainty in some of the reactor physics inputs, as well as an update methodology to those NRFs. Things that the staff had looked at was to ensure the code could actually capture the appropriate geometry material and the physics.

We ensured that there was, on the appropriate, on the assessment of the CMS5 code suite, as well as we looked at the update methodology. We also verified that there was a need to manage any uncertainty during the operation. The surveillances, things like that.

Things that the staff had looked at in more detail was to ensure that there was no impact from having several modules that are close together. We ensured the physics modeling was appropriate, as this was a lower flow reactor than we had seen it before. So we ensured the thermal-hydraulics modeling, which is inside of SIMULATE5, was appropriate, and that they had appropriate validation of analyses for these high leakage cores and heavy reflectors.

Things the staff had looked at was of course this topical report itself, but we had also looked at other information that was provided by Studsvik as part of a generic CMS5, the code review. And then of all of the manuals, which we've seen some of that done here today.

We also looked at other reviews that were done in this area. So things that we have found as part of this review is that the CMS5 code suite does apply to the NuScale design.

We had one limitation such that if they were to make any kind of fuel changes or something like that, it would have to be done within the limitations of certain materials, which are bounded and have been reviewed as part of the CMS5 code suite review.

We found the base NRFs are acceptable. For the record, all the NRFs are larger than what, or they account for more uncertainty than what is currently out there, the base NRFs for the entire op fleet.

And as well as the NRF methodology, we imposed a limitation and one condition on this in order to ensure that there are certain features inside of the code suite that are kind of fixed and that you really can't change without the aid of a code vendor. And so they are kind of limited to those.

MEMBER MARCH-LEUBA: Tim, hold it. I've been looking ahead at your closed session slides, and I realized that you are going to agree with some members' suggestions to add a fourth condition.

MR. DRZEWIECKI: That's right, yeah.

MEMBER MARCH-LEUBA: Can that be addressed in the open session, or it that --

MR. DRZEWIECKI: I think it can be.

MEMBER MARCH-LEUBA: In addition, right?

MR. DRZEWIECKI: Yeah. And so I will do that now. Let me make sure I got my language correct here. Okay, so there was a comment in terms of the branch structure, because it was not clear that they had an appropriate branch structure in order to capture things at high void and things like that.

So you know, we had gone back, there will be a change made into the SE. Because the staff did look at the branching structure, and we verified it was appropriate for use and it can capture all the normal and off-normal conditions.

We did not explicitly state that it

captures the void fraction, as you can see, especially during something like an ECCS actuation. So we're going to add that language.

In addition, all of the cases that are supporting this code suite were done using a case matrix which is developed by Studsvik. And what that case matrix is is the histories and the branch structures used to make all the cross sections. And that is the S5C case matrix.

And so we're going to add a limitation saying that you have to use this case matrix because that was used in all their cases.

CHAIR CORRADINI: And we can get into details in the closed session.

MR. DRZEWIECKI: Yeah.

CHAIR CORRADINI: Okay.

MR. DRZEWIECKI: I think I can say that off in the closed session.

CHAIR CORRADINI: Okay, thank you.

MR. DRZEWIECKI: Thank you.

MR. BAVOL: Okay, moving on to the critical heat flux topical report. The original Revision 0 was submitted in October 2016, and in November 30, a Revision 1 was provided to the NRC and the revised submittal implements an additional NSP4 critical heat flux correlation and incorporates some of the changes that the NRC requested and requests for addition information.

There were two RAIs that were submitted under this review for this particular topical report, and those two responses and evaluations were dispositioned in the safety evaluation. Again, the staff plans to provide a final SER in late July 2018 with the codes and methods at the same time, and then publish the -A approved version early October 2018.

MR. DRZEWIECKI: Okay, so in terms of the staff review, again, on the CHF review itself, you don't make any findings associated with any of GDCs, but it used to support findings associated with things like GDC10, 12, as well as any kind of dose consequences analysis. Because this is used to verify that, you know, you may or may not have a fuel damage.

They had requested the approval for the NSP2 and NSP4 CHF correlations with very specific requirements. It's only for new fuel, HTP2 fuel, over a specified range of application, as well as certain CHFR limits. So that's the approval that was being sought.

Key areas of the review, there's really three large ones, and that's the data itself, the model generation, and a large emphasis was the model error, which is actually used to develop the CHFR limits.

So areas in which the staff spent a little more time was really on the development of the 95/95 limits. We also wanted to make sure that we understood some of the correlation behavior. As discussed before, traditional PWR on a DMVR correlation you would expect to see at a higher pressure that you would gain from a margin.

In other words, if you had, you know, say like, an event in which pressure was reduced, you would usually see less thermal margin. And you'd be closer to getting a DMVR on the event. This correlation shows a different trend.

So as you increase pressure in your system, you get less thermal margin, or you're closer to CHFR. But they had data to back that up. And that's also been seen in other correlations and other data, historical data.

Things that the staff had looked at was obviously this correlation itself, as well as there was an inspection at Stern Laboratories, that was some of the early testing that was done for a fuel type that was not the prototype, which is currently being used in the plant, but for an early prototype for the NuScale design.

There was a test report that was out in 2014, as well as staff did an audit of the testing at the KATHY facility in Germany. And we also built an appendix as part of the CHF SE, which was based on information obtained from the review of other CHF correlations.

MEMBER SKILLMAN: Tim, back to your explanation of margin change for increasing pressure. Is there a brief explanation for why that phenomenon occurs the way it does?

MEMBER BLEY: The physics, not the data, the physics.

MR. DRZEWIECKI: So, okay, well, we -- so I will give you an explanation of why I think it is. Now I could say, on the basis for the finding, it was the fact that they had data. That was the real basis.

But what I think it is is that as you -- so in terms of, you know, of thermodynamics, and you know, please step in if you disagree with me, Mr. Cardini, but as I get to a lower pressure, my latent heat of vaporization gets larger.

And so if you are more akin to like a dry-out scenario, then it takes more energy to boil off that layer. And so that's why I think it is. So does that make sense? Like if you think of, like a TS diagram in thermodynamics, there's that dome, and as you go to higher and higher pressures, it takes less energy to boil water. And I think that's what it is.

MEMBER SKILLMAN: Except that the pin temperature must be at or above a certain value to drive the heat across that vapor. So it's the pin temperature, specifically the clad temperature I would think, that is of greatest concern.

So in order to drive that heat across that film, you've got to have a much higher fuel temperature, and across the gap inside the pen, and then across the clad.

So this kind of ties into our discussion yesterday about how the NsScale design sometimes acts like a P and sometimes acts like B. And what you're pointing to here is a case where as you higher in pressure, the thermal margin decreases. It just almost, to me it's counterintuitive. It may be absolutely accurate and based on data, but it sure seems backwards to me.

MR. DRZEWIECKI: Well, that was the first impression that we had when we first saw this correlation. Because we had heard of that phenomena before we had gone to the audit at KATHY. And that --

CHAIR CORRADINI: I'm sorry.

MR. DRZEWIECKI: Oh, please, if you want to interject.

CHAIR CORRADINI: No, no.

MEMBER KIRCHNER: Actually, I do. So I pondered this too, because it is somewhat counterintuitive, especially for in comparison to the many of the CHF correlations that have been presented and audited and developed in the past.

I think that what's also different here is the flow rates are significantly less than we'd see. So we've got a regime change in how they're approaching the critical, the crisis. And it's more in a dryout mode almost, because I think it's higher local quality in general as well. And so my sense is that's what we're seeing.

Whereas if you look at higher pressure and much higher flow rate data, then you by and large get more margin with pressure, and that's typical of most the PWR correlations we've seen in the past.

MR. DRZEWIECKI: That's right, I agree with that. And I do want to state, so, yeah, so in order to get an understanding of that, we had gone out, that was one of the drivers for us doing an audit at the KATHY facility, was for us to get an early look at that data, but to make sure we understood what was going on.

As a secondary check on that as well, there is these, there's this, it's the Groeneveld CHF look-up tables, which are like an historical, you know, database. And so we had looked for those trends in that table and found them there, at the same flow rates, at the same you know, regimes, we saw the same behavior. So that gave us more comfort.

MEMBER SKILLMAN: Thank you, that was helpful for me. Thank you very much, thanks.

MR. DRZEWIECKI: Okay, what so the staff found is that the NSP2 CHF correlation is acceptable for use of the safety analysis of NuScale power module, but new fuel NSP2 fuel, with a CHF limit of 1.17 over a range which is specified in that topical, that's table 8.2, subject to two limitations.

And similarly, the NSP4 is also acceptable for use with a higher limit over a range which is specified in a different table. And subject to the same limitations and MOC limitations are under Tong Factor or the non-uniform flux factor is always a penalty, it's never a credit. So it's always greater than one.

And that they have, and that, but the use of these correlations have to be performed in accordance with a frozen model, because all the data reduction was done using a frozen model, which is described in the subchannel analysis methodology.

MEMBER KIRCHNER: Tim, that de factor implies that they have to use the VIPRE 01.

MR. DRZEWIECKI: That's right.

MEMBER KIRCHNER: Code.

MR. DRZEWIECKI: That's right. With a certain set of correlations, yeah.

MEMBER KIRCHNER: I would just say in plain English, so to speak, as part of, expand on that. Otherwise, people go to the reference and only define that it's the VIPRE 01 code that they have to use, right. Can we put that in the SE? Well, that's your choice, but.

MR. DRZEWIECKI: You know what, it's -- you think it's we have to identify? I mean, they identify the TR, but.

MEMBER KIRCHNER: No, well, they use that same code to reduce the data. So.

MR. DRZEWIECKI: Yeah.

MEMBER KIRCHNER: But it does beg the question how good is the correlation outside of using that code. You're putting a limitation on for NuScale, but in a generic sense, this -- of course, it's tied to that new fuel, HTPTM.

MR. DRZEWIECKI: Yeah.

MEMBER KIRCHNER: But just you know, is it really dependent on the VIPRE code?

MR. DRZEWIECKI: Well, I will say --

MEMBER KIRCHNER: Or it's just a limitation you're putting on NuScale's application of this report?

MR. DRZEWIECKI: Yeah, so we're putting in on there because in terms of, you know, of all the error quantification was done using that code with these tools. So if you use, yeah, so if they use some other tool, it is possible, but they'd have to do more work.

MEMBER KIRCHNER: Do more work, okay. Thank you.

MR. BAVOL: So this concludes our public portion presentation.

MEMBER KIRCHNER: Do you want me to ask for public comment?

CHAIR CORRADINI: I think yes.

MEMBER KIRCHNER: Okay. Is there anyone in the audience who wishes to make a comment while we're waiting for the phone line to be open? Okay.

CHAIR CORRADINI: We'll wait for the characteristic crackle. Anybody on the phone line?

MEMBER KIRCHNER: Is there anyone on the phone line who wishes to make a comment?

(Off record comments.)

MEMBER KIRCHNER: Once again, anyone on the open line wishing to make a comment? Mr. Chairman, hearing none.

CHAIR CORRADINI: Okay, so Zena, please close it. And also, I think there will be NuScale folks that may be on the closed line for the closed session.

So give us a moment to change over our technology. And if you're on the closed line, please acknowledge that you're out there. I think we're expecting some subject matter experts from NuScale. Is that correct? But you guys, staff does not expect anybody on the line.

MR. BAVOL: That's correct.

CHAIR CORRADINI: Okay. So the closed line is open now. So is anybody on the line, please? Okay.

MR. BRADBURY: Good morning, this is Mike Bradbury from NuScale.

CHAIR CORRADINI: Okay, anybody else?

MR. LINGENFELTER: Andy Lingenfelter from NuScale.

CHAIR CORRADINI: Okay, anybody else?

MR. GAMBLE: Robert Gamble, NuScale.

CHAIR CORRADINI: So all right, we've got three NuScale engineers online, anybody else? Okay, if you guys could mute your lines on your side, so there's no kind of behind-the-scenes chatter, and we'll keep the closed line open. If we need you, we'll inquire.

Walt. So we'll go into closed.

MEMBER KIRCHNER: Yes.

CHAIR CORRADINI: We'll go into closed session.

(Whereupon, the above-entitled matter went off the record at 8:54 a.m.)

Presentation to the ACRS Full Committee

Staff Review of NuScale Topical Reports

TR-0616-48793, REV 0,
"NUCLEAR ANALYSIS CODES AND METHODS QUALIFICATION"
&
TR-0116-21012, REV 1
"NUSCALE POWER CRITICAL HEAT FLUX CORRELATIONS"

Presenters:

Bruce Bavol - Project Manager, Office of New Reactors
Timothy Drzewiecki, Ph.D.- Reactor Systems Engineer, Office of New Reactors

June 7, 2018

(Open Session)

NRC Technical Review Areas/Contributors

- Reactor Systems NRO/DSRA/SRSB:

  Rebecca Karas (BC)

  Jeffrey Schmidt

  Timothy Drzewiecki

- Additional Support on Nuclear Analysis Methodology

  Boyce Travis, NRO/DSRA/SCVB

- Additional Support on CHF:

  Joshua Kaizer, NRR/DSS/SNPB

  David Heeszel, NRO/DSEA/RGS

Staff Review Timeline

TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification"

- NuScale submitted its Topical Report (TR) TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification", on August 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16243A517).
- Staff issued request for additional information (RAI 8807) on May 9, 2017
- NuScale responded to RAI 8807 on July 6, 2017 (ADAMS Accession No. ML17187B239)
- Staff plans to issue its final SER in late July 2018
- Staff plans to publish the "-A" (approved) version of the TR in early October 2018

Scope of the Staff Review

- Regulatory Basis
  - An approved nuclear analysis methodology is used to establish a partial basis for compliance with several general design criteria (GDC) of 10 CFR 50, Appendix A
- Applicant requested approval of:
  1. Applicability of Studsvik Scandpower, Inc. (Studsvik), Core Management Software, Version 5 suite (CMS5) to the NuScale reactor core design (RXC)
  2. Base nuclear reliability factors (NRFs)
  3. NRF update methodology
- Key areas of NRC staff review:
  - Geometric, Material, and Physics modeling capabilities of CMS5 (applicability to NuScale RXC)
  - Assessment of CMS5 suite (code-to-code and **empirical data**)
  - NRF development and update methodology
  - Uncertainty management during operation

Non-Proprietary

Scope of the Staff Review (cont)

- Areas requiring additional review (due to unique design features)
  - Geometry Modeling: Multi-module effects and modeling of heavy reflector
  - Physics Modeling: Thermal-hydraulic modeling capabilities in SIMULATE5
  - Validation analyses to address high leakage cores and heavy reflectors

- Information considered by NRC staff:
  - TR-0616-48793, “Nuclear Analysis Codes and Methods Qualification,” and supporting information obtained via RAIs and audit of supporting calculations
  - Documentation provided by Studsvik in support of generic CMS5 topical report:
    - SSP-14/P01-028-TR-NP, “Generic Application of Studsvik Scandpower Core Management System to Pressurized Water Reactors” (ML15355A285)
    - CASMO5, SIMULATE5, CMSLINK5 manuals
  - Similar topical reports/safety evaluations (Duke Power Company Nuclear Design Methodology, Arizona Public Service Company PWR Reactor Physics Methodology)

Staff SER Conclusions

1. CMS5 code suite is applicable to the NuScale RXC design, subject to Limitation 1:
   - Limitation 1: Application of TR-0616-48793 is limited to the materials identified in the safety evaluation for the generic CMS5 methodology
2. The base NRFs proposed in TR-0616-48793 are acceptable
3. The NRF update methodology proposed in TR-0616-48793 is acceptable, subject to Limitation 2 and Condition 1
   - Limitation 2: Updates to any delayed neutron parameter NRF cannot reduce the magnitude of the NRF below 5 percent
   - Condition 1: Updates to the pin peaking NRFs ($F_{\Delta H}$ and F_Q) must include the pin-to-box bias and fixed in-core detector bias in accordance with the generic CMS5 topical report.

Staff Review Timeline

TR-0116-21012, Revision 1, "Critical Heat Flux Correlations"

- NuScale submitted its Topical Report (TR) TR-0116-21012, Revision 1, "Critical Heat Flux Correlations," on November 30, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17335A089), this revision replaced the submittal dated October 5, 2016 (ADAMS Accession No. ML16279A363), "NuScale Power Critical Heat Flux Correlation NSP2."
  - The revised submittal implements an additional NSP4 critical heat flux (CHF) correlation and incorporates changes associated with NRC requests for additional information (RAIs).
- Staff issued two requests for additional information (RAI 8795 on May 8, 2017 - ADAMS Accession No. ML17128A468) and (RAI 8931 on August 21, 2017 - ADAMS Accession No. ML17233A127)
- NuScale responded to RAI 8795 on (July 7, 2017 - ADAMS Accession No. ML17188A461) and RAI 8931 on (September 25, 2017 - ADAMS Accession No. ML17268A385)
- Staff plans to issue its final SER in late July 2018
- Staff plans to publish the "-A" (approved) version of the TR in early October 2018

Scope of the Staff Review

- Regulatory Basis
  - An approved critical heat flux (CHF) correlation is used to establish a partial basis for compliance with several regulations including the general design criteria (GDC) of 10 CFR 50, Appendix A:
    - GDC 10 and GDC 12 (margin to fuel design limits), 10 CFR 52.47(a)(2)(iv) and GDC 19 (radiological consequences)
- Applicant requested approval of:
  - Use of NSP2 and NPS4 CHF correlations for the safety analysis of the NuScale Power Module
    - NuFuel-HTP2™ fuel
    - Over a specified range of applicability
    - With specified CHF ratio (CHFR) limits
- Key areas of NRC staff review:
  - Collection of experimental data
  - CHF model generation
  - Quantification of CHF model error (development of CHFR limits)

Scope of the Staff Review (cont)

- Areas requiring additional review
  - Statistical Analysis – Development of the 95/95 CHFR limits
  - Correlation behavior differs from trends observed in previously reviewed CHF correlations for pressurized water reactors

- Information considered by NRC staff:
  - TR-0116-21012, Revision 1, "NuScale Power Critical Heat Flux Correlations," and supporting information obtained via RAIs and audit of supporting calculations
  - Inspection report for CHF testing at Stern Laboratories, April 18, 2013
  - NuScale CHF test program technical report, January 24, 2014
  - Audit report for CHF testing at KATHY, August 12, 2016
  - Information obtained from previously reviewed CHF correlations as documented in Appendix A of the staff's SE

Staff SER Conclusions

- NSP2 CHF correlation is acceptable for use in performing safety analyses of the NPM with NuFuel-HTP™ fuel
  - With a CHFR limit of 1.17
  - Over the range of applicability provided in Table 8-2 of TR-0116-21012, Rev. 1
  - Subject to Limitation 1 and Limitation 2
- NSP4 CHF correlation is acceptable for use in performing safety analyses of the NPM with NuFuel-HTP™ fuel
  - With a CHFR limit of 1.21
  - Over the range of applicability provided in Table 8-4 of TR-0116-21012, Rev. 1
  - Subject to Limitation 1 and Limitation 2
- Limitations
  - Limitation 1: Nonuniform flux factors must always be greater than or equal to 1
  - Limitation 2: CHF analyses using the NSP2 and NSP4 correlations must be performed in accordance with TR-0915-17564, "Subchannel Analysis Methodology"

Questions/comments from members of the public before the closed session starts?

Non-Proprietary