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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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707TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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OPEN SESSION

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WEDNESDAY

JULY 12, 2023

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The Advisory Committee met via hybrid In-Person and Video-Teleconference, at 8:30 a.m. EDT, Joy L. Rempe, Chairman, presiding.

<u>COMMITTEE MEMBERS</u>:

JOY L. REMPE, Chairman

WALTER L. KIRCHNER, Vice Chairman

DAVID A. PETTI, Member-at-Large

RONALD G. BALLINGER, Member

CHARLES H. BROWN, JR., Member

VICKI M. BIER, Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member
JOSE MARCH-LEUBA, Member
ROBERT P. MARTIN, Member
THOMAS E. ROBERTS, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:
DENNIS BLEY
STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:
KENT HOWARD

ALSO PRESENT:
BUCK BARNER, Framatome
KURT CRYTZER, EPRI
LOIS M. JAMES, NRR
KENNETH GEELHOOD, NRR
KEVIN HELLER, NRR
SCOTT KREPEL, NRR
RYAN JOYCE, SNC
JOHN LAMB, NRR
JOHN LEHNING, NRR
MICHAEL MARKLEY, NRR
ALAN McGINNIS, Framatome

CHARLEY PEABODY, NRR

RADU POMIRLEANU, Westinghouse

DAVID RAHN, NRR

JIM SMITH, NMSS

GREGORY SUBER, NRR

MIKE WERNER, Westinghouse

BRANDON WISE, NRR

KENT WOOD, NRR

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

8:30 a.m.

CHAIR REMPE: Good morning. It's 8:30 here on the East Coast. This meeting will now come to order. This is the first day of the 707th Meeting of the Advisory Committee on Reactor Safeguards.

I'm Joy Rempe, Chairman of the ACRS. Other members in attendance are Ron Ballinger, Vicki Bier, Charles Brown, Vesna Dimitrijevic, Greg Halnon, Walt Kirchner, Jose March-Leuba, Bob Martin, Dave Petti, Thomas Roberts, and Matthew Sunseri.

We do have a quorum. Today the committee is meeting in-person and virtually. The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act.

The ACRS section of the US NRC public website provides information about the history of this committee and documents such as our charter, bylaws, Federal Register notices for meetings, letter reports, and transcripts of all full and subcommittee meetings, including all slides presented at the meetings. The committee provides its advice on safety matters to the Commission through its publicly available letter reports.

The Federal Register notice announcing

this meeting was published on June 21, 2023. This announcement provided a meeting agenda, as well as instructions for interested parties to submit written documents or request opportunities to address the committee. The Designated Federal Officer for today's meeting is Mr. Kent Howard.

The communications channel has been opened to allow members of the public to monitor the open portions of the meeting. The ACRS is inviting members of the public to use the MS Teams link to view slides and other discussion materials during these open sessions.

The MS Teams link information was placed in the Federal Register notice and agenda on the ACRS public website. We have received no written comments or requests to make oral statements from members of the public regarding today's session.

Periodically, the meeting will be open to accept comments from participants listening to our meetings. Written comments may be forwarded to Mr. Kent Howard, today's Designated Federal Officer.

During today's meeting, the committee will consider the following topics: EPRI Data Validation Topical Report, Vogtle License Amendment Request on Loading Lead Test Assemblies with Increased

Enrichment, and ARITA-ARTEMIS/RELAP Integrated Transient Analysis Methodology Topical Report.

Note the portions of the EPRI and ARITA-ARTEMIS sessions may be closed as stated in the agenda. I also want to note that the topic of LANCR02, Lattice Physics Model Description Licensing Topical Report, that was scheduled for tomorrow's session will be discussed during our Friday planning and procedures meeting.

A transcript of the open portion of today's meeting is being kept. It's requested that speakers identify themselves and speak with sufficient clarity and volume so they can be readily heard. Additionally, participants should mute themselves when not speaking.

This morning I do have an item of note. I want to recognize our two newest members, Bob Martin and Tom Roberts, who are joining us for their first full committee meeting open session.

Bob is a career nuclear safety specialist through employment with BWXT, Babcock & Wilcox, Framatome, and Siemens Power Corporation. Dr. Martin was responsible for the development and regulatory defense of several evaluation methodologies for the design and safety of conventional and advanced nuclear power plants.

Among these are the first applications of best-estimate plus uncertainty methods to design-basis LOCA for both fuel and containment evaluations and for demonstrating plant resiliency to beyond-design-basis severe accidents.

His experience includes employment at the Idaho National Laboratory, where he worked in the area of thermal system fluid modeling, and being the lead editor and contributor on the testing of design-basis accident analysis methods for light-water nuclear power plants.

Tom Roberts has more than 40 years' experience in the field of nuclear reactor systems and safety. He spent 36 years as an engineer and engineering manager at the Naval Nuclear Propulsion Program Headquarters, working various roles in the Instrumentation and Controls Division, and then completing his career with 12 years as the Director of Reactor Safety and Analysis.

Since his retirement from the Naval Nuclear Power Program, Mr. Roberts has served as a subject matter expert in programs for advanced reactor development, including consulting and reactor safety, and IMC for a transportable micro-reactor program and for a nuclear thermal rocket reactor concept.

I do hope that members and others will join me in welcoming Rob and Tom, and in thanking Alicia, Andrea, and Sandra for helping them through the onboarding process.

At this time, I want to ask other members if they have any opening remarks. Not hearing any, I'd like to ask Member March-Leuba to lead us through our first topic for today's meeting.

Jose?

MEMBER MARCH-LEUBA: Our first topic is the EPRI Data Validation Topical Report, which we listened to during the thermal-hydraulics subcommittee on June 7th. We are going to have only an open session. If required, we can always create a closed link and discuss.

So please try not to mention any numbers, which does tend to be proprietary. Other than that, keep it general. If necessary we can close the session, but I hope we don't.

I believe the staff is going to present. Greg or Scott, are you going to present? Greg is going to. Go for it.

MR. SUBER: Good morning and thank you. I'd like to welcome the two new members of the ACRS.

I'm glad that you got here so you can understand the term hotter than July, as you will experience over the next three days.

Once again, I'd like to thank the committee for giving us the opportunity to discuss the EPRI Technical Report for the use of data validation and reconciliation methods for measurement uncertainty recapture.

This topical report is a first-of-a-kind topical report. It proves that it has the potential to increase power for the US Nuclear Reactor fleet. It's also a first of a kind in that it is an approval of concept that is not typically the subject of a topical report.

Therefore, the staff has conducted a very thorough and in-depth technical review using contractors and support from Sandia Labs, which enhance the NRC Staff's capabilities.

EPRI, its members, and consultations do their part in submitting a detailed topical report and responded to a request for additional information in a timely manner.

As a result of everybody's dedication and efforts, the staff's safety evaluation for the topic was issued on May the 6th. The staff concludes that reasonable assurance has been provided for the DBR method.

It can be used to determine, one, core thermal power, and also core thermal power uncertainty. And thus, this report can be used by licensees for requests for measurement uncertainty power operations.

The NRC staff and EPRI are ready to discuss the safety evaluation in detail in this meeting. With that, I turn it back to the Chair.

MEMBER MARCH-LEUBA: Thanks, Greg. Lois James is going to be our presenter; is that correct?

MR. SUBER: Yes, she is.

MEMBER MARCH-LEUBA: So Lois, we're going to give you the microphone in just a moment. I'll just give you a heads up that some members were not present during the subcommittee and they have questions about a topic that Josh Kaizer presented very eloquently to the committee, which was the way he introduced the risk-informed methodology to his review.

Either you address it when you feel it's proper or we will ask you the question at the end of the presentation. I'm just giving you a heads up. So Lois, go ahead.

MS. JAMES: Thank you, Chair Rempe, Member March-Leuba, and other members of the ACRS. As stated, we are here today to discuss the staff's review of EPRI, Electric Power Research Institute's report on the use of data validation and reconciliation for DBR methods for measurement uncertainty recapture or MUR.

The staff has acknowledged that this is a first-of-a-kind report and has the potential to have a large impact on the nuclear industry.

As mentioned, my name is Lois James. I am the project manager for the staff's review. In the room or on the phone we have Scott Krepel, Nuclear Systems Performance Branch Chief, and David Rahn, one of the technical reviewers. We also have several members of EPRI and their supporting staff to answer any questions that the members may have.

The agenda for today, we will be providing a short history of the staff's review. We'll mention the purpose of the technical report. We'll discuss what DBR is. We'll also talk about how DBR impacts or is used for core thermal power. We will mention the review scope limitations and then we will discuss the staff's evaluation conclusions.

I'm not going to mention every item on the project history. As you can see from the two slides, we have provided the ML numbers. We thought this was important to provide key dates and ADAMS reference numbers for everyone, for members of the public and for the committee.

You can see that the staff's review started in early 2001. We brought Sandia Labs on to support the staff's review during 2001. During 2002 and 2003, the staff issued the RAIs and conducted its review.

In May of 2003, we issued our draft safety evaluation for proprietary review. We shared that with the committee. That's all the project history that we were going to mention today.

MEMBER MARCH-LEUBA: Lois, this is Jose. Can I ask you a question now?

MS. JAMES: Sure.

MEMBER MARCH-LEUBA: Since this is the full committee, in my opinion, this is more for the benefit of the public than for our questions. We drilled you guys during subcommittee.

I wanted to ask you, I see that the project history is two and a half years long from issuance of the topical report and issuance of the SER. Can you explain, was there any kind of problems that need to be resolved with the RAIs? Or it's simply lack of resources on the part of the staff? Because two and half years looks like a long time for a topical report.

MS. JAMES: Yes, sir. When we received the project initially, we decided that it was such a unique project and dealt very strongly on statistical methods that we chose to bring in and contract Sandia to support the staff.

That took longer than anticipated. So a good bit of 2021 was actually procuring the contract from the start. That's not an easy task to go from start to finish. That takes at least six to nine months to get a contract in place.

And then when we finally started the review, EPRI and its contractor support provided a lot of information in the topical report. The first set of RAIs we issued was to determine what exactly they wanted us to review and approve because they had provided a lot of examples. And they had provided some code information and some method information.

So the first set of RAIs that we issued in 2002 after we got the contract in place was first to determine really what we were supposed to review and what we were supposed to document in the SE. And so the actual second round of RAIs, phase B, those were kind of the technical questions that we had. So that's why the review took a little longer than anticipated.

MEMBER MARCH-LEUBA: Thank you. I want you to understand that, first, anything you hear here until we write the letter are individual members' opinions. We always say that, but we need to keep on saying it.

If I summarize what I heard you say, the reason it took two and a half years for review was resource location and the lack of proper communications at the beginning of the project, misunderstanding of the scope.

In my opinion, and it's my opinion, I found when I was doing the work on your side of the table, audits helped with the misunderstandings very much. You have an audit with the applicant and you determine the scope real well.

I hope management keeps getting lessons learned and we speed up this process because it's in the benefit of everybody. Thank you. Keep going. It was just an observation.

MS. JAMES: Understood. The staff has developed internal lessons learned. I try to do that in every one of my projects so that we can go forward, and do things more efficiently and effectively in the future.

MEMBER MARCH-LEUBA: Yes. I'm not wasting time. This is the beginning of the meeting so we still have to -- you always have a pre-submittal meeting with the applicant when you get a flavor of what's going to be submitted. But once the technical responsible staff get assigned to the project, in my opinion, having an audit speeds up everything. It really does. Okay.

MS. JAMES: What is data validation and reconciliation? It's a statistical analysis of multiple plant measurements, an aggregate to provide an accurate core thermal power.

Of interest to us in this topical report was that DBR can reduce uncertainties associated with the core thermal power and allow plants to operate closer to the approved thermal power without reducing the safety margin.

Using the data validation and reconciliation can also help plants reduce single-failure vulnerabilities, using more instrumentation to determine core thermal power. It can also improve condition monitoring and condition-based maintenance by using the data points collected by the plant equipment. This can be used by the system engineers to better monitor their plants.

We would like to acknowledge that DBR has been used in US and European nuclear power industries since 1999. It has been used to assess the target cycle thermal performance, balance of plant feedwater flow metering, and accuracy of the plant metrics.

It is also being used, it is my understanding, right now in Europe to increase power output. So that's where the US market is headed.

The purpose of the technical report was to describe the process for using the mathematical data validation and reconciliation for specifically monitoring core thermal power. And then use this DBR to proceed into the measurement uncertainty power upgrade or measurement of certain recapture power upgrades.

Anything impacting core thermal power has potential to impact safety. So the staff conducted an in-depth review that was discussed in detail with the ACRS subcommittee during the closed portion of the meeting in June.

Currently, core thermal nuclear power is calculated based on feedwater flow measurements.

There are inaccuracies in the direct measurement of feedwater flow that have resulted in lost generation and potential overpower conditions. In the past ultrasonic flow measurement devices, UFMs, have been used to gain measurement accuracy. But as technology improves, more can be done.

Data validation and reconciliation, as I stated, uses statistical analysis of multiple plant measurements to provide an accurate reading of core thermal power. The DBR can then be used to reduce uncertainties in the core thermal power and thus allow licensees to produce more power via a measurement uncertainty power upgrade.

So we expect after approval of this topical report that we will begin to see the industry's interest in measurement uncertainty recapture upgrades.

We wanted to mention that this review, the safety evaluation was not a review of any software, logic flow, or numerical method implemented by any particular vendor or licensee. It is a review of the concepts of DBR and steps needed to provide the model used by the plants to estimate feedwater flow and core thermal power need and uncertainty.

Specific evaluations of software, logic

flow, and numerical methods would be performed via a license amendment request or an application for a power upgrade.

MEMBER MARCH-LEUBA: Lois, this is Jose again. As I told you earlier, this would be a very good point at which you can introduce the risk-informed methodology or approach that the staff used to review these power upgrades. Could you give us a hint about how we did it, a summary, a high-level summary?

MS. JAMES: Yes.

MR. RAHN: If you'd like, Lois -- this is David Rahn -- I could help with that.

MS. JAMES: That would be great. Thank you.

MR. RAHN: Okay. Thank you.

Chairman Rempe and Member March-Leuba, the staff was concerned initially that there was quite a bit of new mathematical and statistics that were used in the concepts that were provided in the DBR methodology.

The process of coming up with a reconciled mean and a reconciled variance associated with the measurements from all the different parameters that are going to be used to help reduce the overall uncertainty for feedwater flow measurement was complicated.

So we wanted to look at it to say, okay, even if we're way off base and we're not understanding it properly, how far off could it possibly be? How bad could we be to come up with a computation of core thermal power?

To do that we used our regular risk-informed processes, which is to look at what we call the risk triplet. What can go wrong, how likely is it, and what are the consequences? In addition, we looked to see if any of those questions we answered incorrectly, what's the residual risk in that?

So essentially, we parsed the uncertainty measurements into what could go wrong if we made a mistake in determining the reconciled mean. And also, what could go wrong if we determined the reconciled variance with an incorrect measurement.

By looking at that, we found that there's a history of operating the plant already. And we know what the ballpark should have been for those measurements. So if we're off, we look to see how far off could they really be.

In our analysis we determined that even if we were off two percent, which is what was our initial look at it, it looked to say that it's not going to be an adverse problem on the operation of the core. We already have an allowance for that two percent.

But let's say we were even to double that. We looked to see what the possible implications are of four percent, for example, being off. And we looked to see what conservatisms we already have in either transient analysis or steady-state operations, operating with a starting point of four percent.

We found that the conservatisms that we already have built into our analysis are fairly substantial, and that we could live with the risk of as much as four percent uncertainty. So essentially, the risk analysis that was performed was a what-if type scenario. We tried to put bounds on it.

The only thing we could not do is determine the likelihood. We had no basis for establishing what would be the likelihood. So we focused on what would be the worst-case consequences.

MEMBER ROBERTS: David, this is Tom Roberts. I have a couple of questions on what you just presented.

One, you said that the analysis method is complicated. I guess my experience is most analysis methods are complicated. And you end up having to carefully go through all of the assumptions and make sure that the methods you're using are appropriate for what it is you're modeling.

I guess I'm not seeing how this DBR is any different than any other analysis in terms of the residual uncertainty just because you choose to use that method compared to some other method.

If you used square-root accommodation of uncertainties for a typical analysis, for example, you would still have to note that those individual terms are all random, independent, and possibly undistributed, all those kinds of things that are important for any kind of analysis.

So I'm just trying to figure out why this particular analysis. That's what prompts this thought process.

MS. JAMES: David?

MR. KREPEL: This is Scott. This is Scott Krepel speaking through a sign language interpreter, if I may. I would like to take an attempt to answer this.

Josh Kaizer is one of my staff, but unfortunately he is unable to be with us today. He's on a plane somewhere, I want to say over the Atlantic. He is on his way back from Australia. So I will do my best to answer your question.

I think one thing that David left out here in the discussion when it comes to risk evaluation is that the risk evaluation itself uses risk-informed scope of review.

They did still do the review of the DBR methodology and looked at a lot of the different aspects of the methodology to make sure that they understood appropriately and had an appropriate establishment of certain criteria, so to speak, to make sure that all of the factors that you just mentioned are addressed. The risk evaluation itself is limited in scope in terms of that review as appropriate with the risks.

MEMBER ROBERTS: I'm wondering if this thought process would apply to any analysis. It seems like any analysis has the same potential pitfalls that you've identified on this DBR approach.

MR. KREPEL: This is Scott again speaking through a sign language interpreter. Sure. You could apply that same approach with any analysis or methodology. In fact, we typically do, but we don't always document it as clearly as Josh Kaizer did. And I think that's a great model for how people could do this type of thing in the future.

MEMBER ROBERTS: I think you're probably right. That leads to my next question, which is the conclusion that the four percent, I think you said, overall error is accommodated by the margins that are in other analyses.

I look at some of the analyses, like the best-estimate plus uncertainty analysis method, for example, has a 95 percent confidence requirement. I always thought that that was because it was essentially already risk-informed.

And 95 percent is good enough because of the very low likelihood of the event itself and the relatively remaining residual amount of defense-in-depth that even if you had the event fail, there is still some left. So that's already risk-informed.

This now eats into the 95 percent. And I'm wondering if you've thought about that. And the four percent, how much does that eat into 95 percent on an accident or a DNBR 95/95 criteria, something like that? Is this essentially double-spending some of the risk-informed judgement is where my question is going.

MR. KREPEL: This is Scott again. I think that's a good comment and a good point. I would say that Josh did look at the consequences in his safety evaluation. And I believe you might have already taken a look at that safety evaluation.

He discusses various things that might happen if you were to go to 40 percent, a higher power level than estimated, like increased oxidization, for example, for the fuel during normal operation. It's possible there's some loss of margin to the operating limits, but Josh also pointed out that typically you would have margin in the design limit. That is a worst-case scenario.

The staff typically, intuitively at least, thinks that is much smaller than typical for that error. Or making a mistake at 40 percent, for example. So the risk would be considered acceptable. The licensees typically do have some margin there in their design limit.

MEMBER ROBERTS: Maybe not a fair question at this point, but how high would it be before you would start to worry? Would seven percent be a problem? Would ten percent be a problem? Would 20 percent be a problem? Where does the thought process start to break down?

MR. KREPEL: No, that's not a fair question. To be honest, there is some engineering judgement there that's involved in that determination.

And so I can't really answer that question myself alone on the spot.

MS. JAMES: This Lois James. I would say one of the things that made this unique was we're essentially approving a concept. We know that there are different methods being used at different plants.

We know that vendors have different methods for calculating and they have different equipment. So we know we're going to have to go into more detail of those, of some of that when we do individual reviews that get submitted.

MEMBER ROBERTS: Thank you. I think I understood that. So if I summarize, what I think I'm hearing is, one, you are treating this as if this is a new analysis method that's expected to be done properly. And I believe your safety evaluation has some conditions in it that help you ensure that the applicant is applying it properly.

It's mathematically rigorous. I think your SE would agree with that, that it's a valid approach to use. And you don't expect it to introduce any significant error uncertainty at all. But you also do the side study to say, okay, what if we're wrong?

MS. JAMES: Yes.

MEMBER ROBERTS: And based on judgement at what's probably four percent of a very, very high estimate of how bad it could be, you conclude it's probably okay. So I think I understand. I appreciate it. Thank you.

MEMBER MARCH-LEUBA: Any more questions from members or comments from the staff?

MS. JAMES: I should apologize. David Rahn is having a little trouble with his Teams. It booted him out when he was speaking. He apologizes. He's unable to get back in at the moment.

MEMBER MARCH-LEUBA: We have a good sound on our end. Okay. Lois, you can continue with the conclusions.

MS. JAMES: Based on the staff's review, we looked at the risk assessment of the DBR results. We looked at previous treatment of similar models and simulations.

We looked at previous evaluations of nuclear power plant processes measurement uncertainty. We looked at the understanding of the DBR method and previous treatment of calculations of the feedwater flow and its uncertainty.

And based on all of this, we concluded that there is reasonable assurance that the DBR method as described in the topical report can be used to determine the core thermal power and the core thermal power uncertainty. Thus, all DBR conditions and limitations have been satisfied.

So that's all we had anticipated discussing this morning. Are there any other questions or comments? And I'm going to see if I can get David --

VICE CHAIR KIRCHNER: Lois, this is Walt Kirchner. Tom had asked already what I was going to ask at the conclusion.

What would you be expecting to see if an applicant comes in with an LAR to upgrade the power? More precision or improved instrumentation on feed order to reduce the uncertainty in that particular parameter or do you think they'll just say, well, the staff looked at four percent.

They've kind of shown their hand. And now we'll look at testing the staff to see if we can get a -- I'll pick a number -- three percent upgrade in power, How will you use this when in a practical sense an applicant comes in with a power upgrade application?

MS. JAMES: Well, the --

VICE CHAIR KIRCHNER: What are you looking for? You won't be looking for it on the reduced margin.

MS. JAMES: No.

VICE CHAIR KIRCHNER: And things like DNBR. So how in practice do you anticipate this being used with an application?

MS. JAMES: We have already rolled out this to MURs. And we understand that that's associated with the LEFMs. That's kind of the starting point. So they're going to use this in conjunction with that guidance on how to do -- well, I guess it's a reg guide on how to do the MURs.

And then we would expect them to come in with their calculation. How are they going to calculate the uncertainty? What's their computer program? What's their modeling?

Since David is not in, I don't know if any EPRI person would kind of like to step in and make the comment at this point of anything further. We expect a lot of the computational models, the uncertainties, how it's going to be used, how it's not going to be used.

VICE CHAIR KIRCHNER: Is there a need, Lois, in your opinion for further guidance from the staff for implementation?

MS. JAMES: We have not determined whether we need additional guidance, but that's definitely not out of the realm because we've done it, as I mentioned, for the LEFMs. Our focus has been to get the concept approved and out the door next month.

MR. KREPEL: This is Scott, if I could jump in there as well, again speaking through a sign language interpreter here. I just want to give a reminder.

In the safety evaluation there are ten, if I recall correctly, ten criteria. Those criteria will be looked at in more detail during the actual review because those are viewed as fundamental to whether or not the methodology is acceptable.

MEMBER MARCH-LEUBA: I believe EPRI would like to make a comment.

MR. CRYTZER: Hello.

MEMBER MARCH-LEUBA: Speak loudly.

MR. CRYTZER: Okay. This is Kurt Crytzer with EPRI. The way that we had instructed it was to capture the BDI 2048 method, which is a statistical method, and not put it into any particular software. The software we used would have to be compliant with what BDI 2048 would accept.

And so the implementation of this would be very much similar to the leading edge flow meters where a correction factor would be applied to the feedwater flow to recapture some of the uncertainty measurement.

MEMBER MARCH-LEUBA: Since we have you there, for the record, I believe the correction we're talking about is 15 megawatts electric over 1,000. So it's like 0.1 percent. Would the staff take the risk of operation to say, we were completely wrong but we would expect the correction to be 0.1, correct?

MR. CRYTZER: Yes.

PARTICIPANT: Can you say your name again clearly for the court reporter?

MR. CRYTZER: Kurt Crytzer.

PARTICIPANT: That's good for context.

MEMBER MARCH-LEUBA: Any more comments or questions from members or the staff?

Hearing none, I'm going to open it up in case there's a member of the public that wants to place a comment on the record. Please do so now.

Hearing none, I return the meeting to you, Ms. Chair.

CHAIR REMPE: Thank you, Jose.

At this time, I would note to the court reporter we're going to go off the record. We'd like you to come back at 1:00 p.m. East Coast time.

(Whereupon, the above-entitled matter went off the record at 9:06 a.m. and resumed at 1:00 p.m.)

CHAIR REMPE: Okay. It's 1:00 p.m. on the East Coast. I'd like to ask Member Ballinger to take us through our second topic for this meeting.

Ron?

MEMBER BALLINGER: Thank you, Madam Chairman. We had a meeting on this topic for the Vogtle LTA at our subcommittee meeting in June where Southern Nuclear and Westinghouse presented an exhaustive, I would say, very thorough presentation of what they claim to do and an analysis that was required.

The staff presented their analysis. Their presentations today are a subset of those presentations. And I think the staff would like to say something initially.

MR. MARKLEY: Yes. This is Mike Markley. I'm Chief of Licensing for the Division of Operating Reactor Licensing for the Vogtle site, Units 1 and 2.

I'd like to thank you all for helping and reviewing this first-of-a-kind -- it is very much a first-of-a-kind review for us in the fact that it's rated in five percent enriched uranium-235 of accident-tolerant fuel assemblies. And it's applicable only to the current burn-up limits the staff has approved for Vogtle Units 1 and 2.

You're going to hear the highlights today regarding our safety evaluation. The staff expects future submittals from the industry requesting higher enrichments and higher burn-ups for longer operating cycles.

We have been having pre-licensing of the other licensees and expecting other licensees to submit a license amendment request to transition to 24-month cycles in the next few months. We appreciate the subcommittee's questions and comments during its June 21st meeting.

For today's briefing, the subcommittee requests that the presentation discusses why NRC has confidence in technical issues to resolve properly, balancing engineering judgement and risk-informed, and look at the long game of batch loading. We are prepared to do that today.

The staff welcomes an ACRS letter. We thank you for the opportunity to present and talk with you today. We'll do our best to answer questions.

Thank you. I'll turn it over to Southern Company staff.

MEMBER BALLINGER: I think Southern Company's on.

MR. MARKLEY: Yes.

MR. JOYCE: Thank you. This is Ryan Joyce. I'm the Licensing Manager at Southern Nuclear.

First of all, I'd like to thank the ACRS for their consideration of this important initiative that benefits not just SNC but the entire industry as we move to higher enriched fuels. It ultimately will help ensure the safety, reliability, and economics of nuclear power plants, ensuring nuclear power is a viable energy source for many years into the future.

The agenda items I'll be discussing will be the LTA program review, request exemptions, summary of testing for adopting AXIOM cladding, and the various analyses that were performed that ultimately will demonstrate the due diligence performed to ensure the LTAs will operate safely and within the analyzed limits.

The goal of the program is to irradiate higher enriched fuels in a commercial reactor and generate data in support of future license applications. Although it's a limited number of fuel rods, this will allow us to exercise the regulatory process and work through various issues associated with ultimately loading higher enriched fuel assemblies beyond five weight percent, and work through logistical, regulatory, and legal challenges that will come up as we go to this higher enriched fuel.

A second objective is to obtain additional data for accident-tolerant and advanced fuel materials, fuel pellet and cladding materials.

A future goal of this, which is outside the scope of this amendment, is to ultimately increase the license burn-up limit and go to higher burn-up fuels. Again, that's outside of this specific amendment but that is the end in mind, to ultimately allow the application of higher enriched, higher burn-up fuel.

For the LTA program, we have four LTAs. Each LTA will contain four fuel rods with up to six weight percent U-235. All LTA rods will have AXIOM cladding. All but one LTA will have chromium coating. About half the rods, 136 for LTA including higher enriched rods, will have doped ADOPT pellets whereas the other half, roughly 128, will be IFBA rods.

For the reactor core itself, it will be about 16 higher enriched fuel rods out of 50,952 total rods. So a very small percentage of the reactor core will have this higher enriched fuel.

The license amendment request requested to change Tech Specs 4.2.1, 3.7.18, and 4.3 regarding these LTAs. Tech Spec 4.2.1 already allows test assemblies that have not completed representative testing to be in non-limiting core regions. In a few slides from now, I'll discuss why we do not believe this non-limiting requirement was met and why we felt the need to explicitly revise Tech Spec 4.2.1.

Tech Spec 3.7.18 and 4.3 require an assembly enrichment of less than five percent. Due to the small number of fuel rods above five percent, so only four out of 264 per assembly, this requirement is still met. In other words, the average assembly enrichment is less than five percent.

However, we felt it prudent to remove any kind of regulatory uncertainty associated with whether or not we were meeting this requirement. And as I previously mentioned, we wanted to make sure we exercised the regulatory process to lay a road map and ultimately load fuel assemblies greater than five percent enrichment.

As will be discussed in a few slides, as part of this amendment we revised our facility operating license to remove an exemption we had to

70.24 as part of our change in licensing basis, which will be discussed in a few slides.

As part of this, we requested an exemption to 50.46 and Appendix K for AXIOM cladding. The exemption request will apply to the application of 10 CFR 50.46 and 10 CFR 50 Appendix K of regulations to the LTA design, utilizing AXIOM cladding in Vogtle Unit 2.

In conjunction with the 17 percent maximum local oxidation acceptance criteria prescribed by 50.46, a more restrictive criteria was assessed consistent with the data presented in the AXIOM topical reports.

For regulatory clarity, SNC decided to adopt a newer 50.68 regulation to replace the older 70.24 regulation that was described in our facility operating license. We felt that adopting 50.68 provided a clean regulatory foundation for moving forward with high-enriched fuel assemblies.

Moving to 50.68 necessitated a requested exemption, 50.68(b)(7), to allow the LTAs to have five percent enriched fuel rods. Similar to our tech spec requirements, 50.68 refers to an assembly enrichment of less than five percent.

As previously mentioned, technically we do meet this on an assembly average, a weight average level, but we felt it prudent to once again request an exemption in lieu of any regulatory uncertainty based on the limited number of higher enriched rods.

The technical justification, the intent of 50.68 is to include inadvertent criticality. We demonstrate through our very thorough analyses and very conservative analyses that inadvertent criticality well be precluded based on the restrictions we have in place.

The NRC approved for the LTAs to include their own criticality analysis. Adherence to these analyses fulfill the 50.68 requirements. The placement is the new storage racks, which is administratively controlled.

AXIOM cladding, ADOPT pellets, and chromium-coated Optimized ZIRLO cladding have all been used in US PWR reactors, similar to Vogtle. In addition, the AXIOM cladding in the topical reports has been interviewed by the NRC and the ARCs.

The only novel feature without US operating experience associated with these four LTAs is a very limited number of fuel rods with greater than five-percent enrichment. And again, this represents a very small percentage of the overall reactor core.

There's about 51,000 rods in the reactor core, so 16 rods out of the 51,000. So a very small percentage of the overall reactor core.

MEMBER MARCH-LEUBA: And remind me again from the subcommittee, some of these rods are in limiting positions. Or they're in positions that we're not limiting until -- can you explain that?

MR. JOYCE: If I understand the question about limiting, some of these will have a highest, and as mentioned in a couple of slides, will be in --

MEMBER MARCH-LEUBA: I'll wait. Yes.

MR. JOYCE: Any other questions?

VICE CHAIR KIRCHNER: For the record, could you quickly review how many rods with those claddings and composition were tested?

MR. JOYCE: For Millstone, unit three. The licensee stated there will be up to eight re-test assemblies containing fuel rods fabricated with AXIOM cladding inserted into the core for Millstone's team. That was from May 2017.

Byron Unit 2 from an April 2019 amendment requested to insert two LTAs designed by Westinghouse. The LTAs were based on a vintage optimized fuel assembly design. The licensee proposed to insert up to 20 test rods of three different types between the two LTAs. The rods contained a mixture of three materials: uranium, fuel pellets ADOPT, and coated optimizer of a low cladding.

VICE CHAIR KIRCHNER: So there's a good database there from that LTA set of experiments?

MR. JOYCE: I would say we believe there is a good database that supports those ADOPT, AXIOM, and Permian cladding. I don't know if Westinghouse has anything to add to that.

MR. SMITH: Ryan, this is Jim. We agree. The basis for what we were doing is the other applications.

MR. JOYCE: So to your question earlier, the LTAs will have the highest linear heat generation rate or local peaking for portions of the cycle, both at steady state and transient conditions. Nonetheless, the technical specification and limits will continue to be met.

There are no additions to the reference Tech Spec 5.6.5 needed for these LTAs as the current methods are used to evaluate them.

MEMBER MARCH-LEUBA: Will the five percent enrichment rods lean? What's the power inside the bundle for those five-percent enrichment positions?

MR. JOYCE: Radu, can you understand that question?

MR. POMIRLEANU: Could you repeat it? Could you repeat the question, please?

MEMBER MARCH-LEUBA: There are four bins in the fuel element, an enrichment at about five percent. What is the relative bin power for those bins? Is it much higher than average, about average, or lower than average?

MR. POMIRLEANU: It is not significantly different from the leading bins in any given cycle.

MEMBER MARCH-LEUBA: Okay. That's good enough. It's about the same. Thank you.

MR. POMIRLEANU: Yes. Thank you.

MR. JOYCE: You'll see in the slides. It'll be within the FDA bin factor limits that are already in the core report.

MR. POMIRLEANU: It's in the core report. This Radu Pomirleanu from Westinghouse.

MR. JOYCE: For LOCA, the existing large-brick LOCA and small-brick LOCA analysis of record for Vogtle are representative of the LTAs. The fuel is negligibly impacted by the presence of the LTAs. The 50.46 acceptance criteria continues to be met.

For non-LOCA, the non-LOCA transient

analysis that depends only on parameters are not impacted by the LTAs as these don't impact core average heat transfer characteristics, decay heat, or initial core energy. The events that depend on local effects were evaluated for potential effects to the LTAs.

It was determined that there were no impacts to the codes or methods. Any impact due to LTA features is offset by system margins. While the LTAs may lead the core in factors, they are placed in non-limiting locations with respect to analysis. Ultimately, fuel-specific criteria applicable to each accident continues to be met. There is no impact to the source or consequences. The announcement remains bounding.

For fuel rod performance, ADOPT and AXIOM were explicitly modeled to PAD5. Premium coated benefits for corrosion were conservatively not included. For fuel rod design, there is no impact to existing DNBR margin.

For core physics, the chromium coating and ADOPT fuel pellets were explicitly modeled. There is no impact to neutronics modeling for the fuel rods about five weight percent. Core monitoring with Beacon is ineffective.

VICE CHAIR KIRCHNER: Can you put that in context of looking ahead to what you want to do next, where you go to higher enrichment? Higher average fuel assembly will have a lot higher enrichment if you put more rods in, obviously, that are six percent.

What you're saying here is that for these analyses of record, fuel assembly isn't really -- the margins that you have in a conservative analysis of record are greater than the impact that you can calculate as a result of putting just four rods in?

MR. JOYCE: You're speaking regarding the core physics?

VICE CHAIR KIRCHNER: Physics and thermal hydraulics.

MR. JOYCE: Radu, do you want to take the lead on addressing that question?

MR. POMIRLEANU: Sure. This is Radu Pomirleanu from Westinghouse. First of all, I'd like to point out that there will be other licensing submittals that will address publications with an increasing number of higher enrichment fuel and higher burn-ups. But yes, we expect a wider range of impacts that will have to be addressed separately beyond the scope of this submittal.

VICE CHAIR KIRCHNER: What I was trying to infer simply on your behalf is that was only four rods in a bundle. The impact that it would calculate on your core physics parameters and your thermal-hydraulics is really minimal. When you take the next step, there will obviously be an impact on all these evaluations.

MR. POMIRLEANU: Yes. That is correct.

MR. JOYCE: Any additional comments or questions on slide 10?

For the criticality analysis, these assemblies are only requested to be used in the unit to prevent fuel pool. Unit 1 spent fuel pool storage is prohibited.

The law addresses increased enrichment, use of ADOPT pellets, and use of chromium-coated AXIOM cladding with regard to storage criticality. Current NRC-approved codes were applied to address LTA storage.

For LTA storage not requiring the new fuel storage ranks in Unit 2 2-out-of-4 checkerboard fuel pool storage, a direct analysis was performed. The new fuel storage rack analysis demonstrates a margin of limits including dry, fully flooded, maximum, moderation considerations. The Unit 2 storage racks in Unit 2 2-out-of-4 checkerboard are confirmed for the reload process.

For spent fuel pool storage, the current burn-up limit for maximum enriched fuel assemblies is about four gigawatt-days per MTU. That's the current limit for about a five percent enriched fuel assembly.

To provide significant conservativeness into this limit, a burn-up limit of 64 gigawatt-days per MTU was selected for the LTA storage. So a margin increase of 24 gigawatt-days per MTU. This is greater than the eight percent margin.

The 64 gigawatt-days per MTU was selected by additional storage options. Should the LTAs be approved for operation, the burn-up rate is 64 gigawatt-days per MTU. In other words, 64 is beyond our burn-up limits. We cannot go to 64 without a LAR. But if we do ultimately go to a higher burn-up with these four LTAs in the third cycle, we can store with the other storage option.

MEMBER BALLINGER: This is Ron Ballinger. You used the word if, but it's more like when.

MR. JOYCE. Yes, when/if.

MEMBER BALLINGER: Okay.

MEMBER HALNON: This is Greg. The Unit 1 spent fuel, is that prohibited because of lack of analysis or a failed analysis?

MR. JOYCE: I'll let Mike Werner answer this.

MR. WERNER: Mike Werner from Westinghouse. Essentially, the two pools have different racks so we only analyze those in the Unit 2.

MEMBER HALNON: So it's lack of analysis?

MR. WERNER: Correct.

MEMBER HALNON: Talk to me about physically. Can you physically get a Unit 2 assembly into the Unit 1 spent fuel pool?

PARTICIPANT: This is Matt. We're doing this for Unit 2 because that analysis is specific to Unit 2.

MEMBER HALNON: There's a gate between them?

PARTICIPANT: That's correct.

CHAIR REMPE: Excuse me. Someone on the line has their microphone open and it's hard for us in the room to hear what the conversation is. So please check your mics on your computers.

Please re-answer again, Matt. Sorry to interrupt.

PARTICIPANT: No problem.

Yes, that's correct. It's a shared pool.

They do have a physical gate between the two pools. However, they can transfer fuels between them if needed. As I stated before, this was more for an analysis convenience sake to keep it specific to the Unit 2 pool and not do the two.

MEMBER HALNON: Is it a routine or has Unit 2 fuel gone to Unit 1?

PARTICIPANT: They have. There are cases in a redesign scenario where we can pull from Unit 2 fuel to load Unit 1, but that's on a case-by-case basis.

MEMBER HALNON: How do you control that? Is that a typical type of --

PARTICIPANT: It is. It's controlled via the procedures and processes.

MEMBER HALNON: The fuel accounting process?

PARTICIPANT: That's correct.

MEMBER HALNON: And that's done by the fuel engineering?

PARTICIPANT: That's correct.

MEMBER HALNON: Okay.

MR. JOYCE: When we analyze the old Yankee fuel storage racks, the analyses are very different with regard to the filters in the two fuel racks.

MEMBER HALNON: Okay. You get rid of all the borax and all that stuff, there's nothing that you've got to worry about degradation in the Unit 2 fuel pool racks?

MR. WERNER: Mike Werner again. We're not crediting any of the older filters.

MEMBER HALNON: Okay. Crediting any boron in the pool as well?

MR. WERNER: Well, the boron is still the boron credit.

MEMBER HALNON: But you aren't crediting that?

MR. WERNER: Right.

MEMBER HALNON: Okay. Thanks.

MR. WERNER: The analysis that's currently in place credits that. We've just built upon it. We're not changing it.

MEMBER HALNON: Okay. They're credited the same amount so one can't dilute the other one when you open the gate?

MR. WERNER: I'm not sure what the --

MEMBER HALNON: I'd be interested in understanding what physically can go wrong that can cause the boron concentration to change. Typically, if it was my plan, I would credit the same amount of boron concentration in both pools so one can't dilute the other.

MR. WERNER: I believe that the spec limit is the same for both. It's 2,300, I believe.

MEMBER HALNON: Okay. That would be my guess but I didn't want to guess.

MR. JOYCE: We can confirm that, but what's listed is Unit 1 and Unit 2 -- we have combined tech specs with some different requirements for the Unit 1 and Unit 2 spent fuel pools. Where that requirement is listed is common to both units.

MEMBER HALNON: Since I'm elaborating, I'll ask the question, will the operator see anything? Will they see anything during operation or is it thermocouples, SPMDs, or anything in the core?

PARTICIPANT: I don't think we -- Radu, I'll defer to you, but my personal answer to this is they shouldn't see anything leave with four bins per assembly. Even if they're in an instrumented location, there could be some minor differences in mapping when we take those in rack engineering.

But in terms of operation, I think what we're expecting should be negligible and probably would be within the amount which we typically see in operation.

MEMBER HALNON: Okay. Thanks.

PARTICIPANT: Radu, do you have anything additional to add to that?

MR. POMIRLEANU: No.

PARTICIPANT: Thank you.

MR. JOYCE: That concludes my presentation. Are there any additional questions or discussions?

MEMBER BALLINGER: Okay. I don't know if we have our consultants on the line. I don't know who's online. If not, then thank you very much for your presentation.

Now we should transfer to the staff.

CHAIR REMPE: I don't see any consultants on the line.

MEMBER BALLINGER: Yes. I thought Steve would be online. He was this morning.

MR. SCHULTZ: No questions or comments, Ron.

MR. BLEY: Dennis is here too. No questions either.

CHAIR REMPE: I didn't see you on the list. Sorry.

MEMBER BALLINGER: Okay. Thanks again. We just need to change out. The seat is still warm.

There we go.

MR. LAMB: My name is John Lamb. I work NRC's Division of Operating Reactor Licensing at the Office of Nuclear Reactor Regulation, NRR. I'll describe the licensing actions that SNC has requested.

Charley Peabody, who is virtual and works for the Nuclear Systems Performance Branch, SFSB, in the Division of Safety Systems, DSS and NRR, will discuss the updated final safety analysis report, the FSAR, Chapter 15, accident analyses, the loss of LOCA accident analyses and the non-LOCA accident analyses.

Brandon Wise from SFNB, DSS, and NRR will discuss the code analysis and fuel rod design. Kent Wood from SFNB, DSS, and NRR will discuss the fuel handling and storage. Mike Markley, who you heard through the introductions earlier, is the Branch Chief from Doral and NRR and will provide the conclusion.

SNC has requested four licensing actions. One is a license amendment request and three are exemptions. The license amendment request is to revise the license condition 2D and four technical specifications.

The proposed change in license condition 2D is to delete a 1986 exemption to Title 10 of the Code of Federal Regulations, 10 CFR Section 72.4, which is criticality analysis requirement and criticality accident requirements, as SNC is voluntarily adopting 10 CFR 50.68.

ADOPT, AXIOM, Prime ZIRLO, and Optimized ZIRLO are trademarks or registered trademarks of Westinghouse Electric Company.

The three tech specs that SNC is proposing is to revise, one, Tech Spec 3.7.18, which is fuel assembly storage in the fuel storage pool. The second one is Tech Spec 4.2.1, fuel assemblies. And the third one is Tech Spec 4.3, fuel storage.

Technical Specification 3.7.18 refers to fuel storage in a fuel storage pool. Therefore, the tech spec note is added to the fuel storage for the accident-tolerant fuel, ATF, lead test assemblies, LTAs, to meet the Tech Spec 4.3.

Tech Spec 4.2.1 allows ZIRLO, Zircaloy, and Optimized ZIRLO only. Therefore, the tech spec change is needed for the insulation of the ATF LTAs.

In addition, Tech Spec 4.2.1 states a limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. Therefore, the tech spec change is needed to allow --

MR. PEABODY: Excuse me, John. Can you

advance the slides so they can see what you're talking about? Thank you.

MR. LAMB: There are a limited number of lead test assemblies that have not been completed. Representative testing may be placed in non-limiting regions. Therefore, a tech spec change is needed to allow SNC to place the four ATF LTAs in limiting core locations in Vogtle Unit 2.

Technical Specification 4.3 allows up to five weight percent uranium-235. Therefore, the tech spec is needed for storage of the ATF LTAs.

The first exemption is 10 CFR 50.46. The second one is 10 CFR 50 Appendix K. Those proposed exemptions are needed to allow the use of AXIOM cladding. The third exemption is 10 CFR 50.68(b)(7), which allows a greater-than-five weight percent uranium-235.

I'm going to turn it over to Charley Peabody to discuss the FSAR Chapter 15, accident analyses.

Charley, are you there?

MR. PEABODY: Yes. Thanks, John. Can you advance to the next slide? All right.

So the accident analyses, most of the accidents were addressed with one of the points that was mentioned in the previous discussion. The overall enrichment only increases negligibly by adding the four LTAs when you are addressing core-wide effects.

That didn't really change any of the dose consequences for accidents that are evaluated from a core-wide standpoint. There were a few accidents that were evaluated that do have local effects, locked rotor and RCPs. Actually, there were a couple of others.

The only one that ended up still being limiting was the rod injection analysis. So the tech specs are going to reflect that the LTA core locations will remain appropriately limiting for rod injection accidents for LTA utilization.

That's all on this. Are there any questions on the accident analysis?

VICE CHAIR KIRCHNER: Charley, this is Walt Kirchner. When you say limiting, the LTAs cannot go into a control rod position?

MR. PEABODY: That's correct.

VICE CHAIR KIRCHNER: Is that what you're saying in plain English?

MR. PEABODY: Yes.

VICE CHAIR KIRCHNER: Okay, just for the record.

MR. WISE: This is Brandon Wise for the staff. I just want to correct that they can go in rod locations but not limiting rod locations.

VICE CHAIR KIRCHNER: Okay. That is determined by previous analysis of record? How do you determine a priori which are limiting and which aren't?

MR. WISE: I believe SNC and Westinghouse have determined which locations are limiting with respect to control rod location. I believe they tend to be the same locations for most reloads, so they won't be in those locations. And I'm sure there will be some confirmatory analysis to confirm that.

MR. PEABODY: Yes. It also would depend on the individual rod worth. I know rods towards the center of the core have more rod worth than control rod locations on the periphery.

MR. WISE: All right. I am Brandon Wise with the NRC's Nuclear Methods and Fuel Analysis Branch. I did the review for the code analysis and fuel rod design.

For the most part, the codes used by SNC and Westinghouse for the analysis of the LTAs are mostly applicable to the LTAs with the exception of some enrichment limits. Given the limited number of higher enriched rods and the amount of enrichment that they have, the overall enrichment of the assembly still remains below five percent, and in a way still remains in the range of applicability for codes such as PARAGON and NEXUS.

Overall, there's very little change in the neutronic performance resulting from the higher enriched rods. I'd expect more of a change from the ADOPT fuel pellets, which are being justified by not being in limiting positions with respect to control rod injection due to the potential for more severe control rod injection accidents due to the increased density of the ADOPT fuel pellets.

As far as thermal-hydraulic codes go, there's basically no impact as a result of enrichment or ADOPT fuel pellets. There are chromium-coated rods. Although one rod is chromium-coated, we would expect there to be safety enhancement as a result of the chromium coating, but they're not crediting that enhancement.

We did examine some potential detrimental effects of the chromium coating and determined that there's no significant loss of margin associated with any potential detrimental effects that were outlined in the ATF-ISG.

So overall, we expect the AXIOM cladding to perform as well as the Optimized ZIRLO cladding, which is what's in the co-resident fuel. And there's no credit for the chromium-coated cladding.

Next slide, please.

I'll go into a little bit more detail on each of the four LTA characteristics, the first being the AXIOM cladding. It's a zirconium alloy cladding that is expected to demonstrate better in reactor performance compared to Optimized ZIRLO.

For the sake of this application, we assumed it performs as well as Optimized ZIRLO. Therefore, it's on par with the rest of the co-resident fuel.

The chromium coating is a thin chromium coating. It has corrosion resistance, enhanced corrosion resistance compared to Zircaloy cladding. There's no impact to the thermal-hydraulic analysis.

The coating is extremely thin and thermal-hydraulic analysis doesn't have enough resolution to even capture the reduction in flow area that would result from the chromium coating. And of course, there's no benefit taken for the chromium coating.

For each assembly there's four enriched rods enriched to six percent. There's minimal impact to neutronic performance as a result because the overall assembly enrichment is going up by a negligible amount.

There's actually a much higher increase in fuel content associated with the ADOPT fuel pellets, which are a slightly higher density. And they are coated with chromium and alumina. Several characteristics are different from standard UO2 pellets. That enhances the performance of those ADOPT pellets.

One of the concerns the staff had with the ADOPT fuel pellets was the increased fuel content and the potential for more severe reactivity in accidents such as control rod injection. This was dispositioned by limiting the locations in which the LTAs can be stored in the core to non-limiting locations with respect to control rod injection.

Next slide, please.

Any questions for fuel rod design or coating analysis? Okay. I'll hand the presentation over to Kent Wood, who will discuss fuel handling and storage.

MR. WOOD: Good afternoon. My name is Kent Wood. I'm here to do fuel handling and storage. We've done this several times. You're getting four rods in higher enrichment to six percent, which adds a very small amount of material. The theoretical increase in density actually has much more visible material to the fuel assemblies.

For the analysis, they compared the analysis. They used applied engineering judgement to compare this analysis to the analysis of record and then added copious amounts of margin.

In particular, for the new fuel storage and for the spent fuel storage for the 2-out-4, which is the fresh fuel assembly storage in the spent fuel pool, they credited IFBA, which is not in the analysis of record for the new fuel storage or spent fuel storage. So that provides a lot of margin.

As Southern said, they really can't get to -- they also did an analysis where they can credit burn-up, but they can't get that burn-up they're crediting of 64. It will add another LAR, but I guess they won't have to talk to me that time.

So we looked at that. That provides margin. We looked at their analysis of record. They have a copious amount of margins for that 64 gigawatt-days for the burned fuel.

For the LTR they did for their accident, which was a multiple misloading where they modeled all of the LTAs together, they did an analysis. They got acceptable results, but it didn't have the copious margins that the others did. But they also didn't explicitly credit the IFBA, so I took that into account. I said they're going to have a margin there as well so we don't need to go any further.

Based on that, we have reasonable assurance that this license amendment and the LTAs will meet 10 CFR 50.68(b)(2) and (b)(3), which are the new fuel storage, fresh fuel high and dry, fuel rod moderation, and flooded conditions. And then also (b)(4), which is the spent fuel pool, which is considered a fully flooded application.

So we think that that's going to be that. It's reasonable to have an exception to 10 CFR 50.68(b)(7), which is the enrichment limit. That concludes my presentation.

MR. MARKLEY: Mike Markley again. I'm Branch Chief of Licensing for Vogtle. We appreciate the feedback that we've received from the subcommittee and each of the members. This is a first of a kind for us.

We know that we're going to be getting batch loads of higher burn-up and higher enrichment, and they're probably coming faster. So again, we want to come to the ACRS early and often, get the feedback from you, and take that into consideration as we go.

The NRC staff determined for this request that there's reasonable assurance that the health and safety of the public will not be endangered by allowing SNC to use the four ATF LTAs in limiting locations without completion of representative testing for up to two cycles, except for the locations where the LTAs may not be placed in core regions that have been shown to be limiting with respect to control rod analysis.

Again, we would appreciate a letter. We don't need one to proceed, but we really do value the ACRS' feedback. Thank you.

MEMBER BALLINGER: Thank you.

Questions from the members or consultants? Hearing no others, thank you once again, both Southern Nuclear and the staff, for a good presentation.

I think we're back to you, ma'am.

CHAIR REMPE: We need to give public comment.

MEMBER BALLINGER: Sorry about that. Are there any people out in the public that would like to make a comment? If that's true, please state your name and make your comment.

Hearing none, back to you, Madam Chair.

CHAIR REMPE: Okay. This time I will accept. Thank you. I believe you do have a letter. Do you want to read it in?

MEMBER BALLINGER: Yes.

CHAIR REMPE: We're going to go off the record.

(Whereupon, the above-entitled matter went off the record at 1:43 p.m. and resumed at 3:29 p.m.)

CHAIR REMPE: Okay. It's about 3:30 here on the East Coast. I'm going to turn the meeting back over to Member March-Leuba to lead us through our third topic for today.

MEMBER MARCH-LEUBA: So the topic right now is ARITA from Framatome, which is their transient analysis methodology for essentially everything but LOCA and logic.

We covered this topic in our subcommittee meeting on June 22nd. And without further ado, Greg is going to give us some remarks from the staff.

MR. SUBER: Thank you. Good afternoon. My name is Gregory Suber. I am the Deputy Director for the Division of Operating Reactor Licensing in the Office of Nuclear Reactor Regulations.

I'd like to thank the full committee for the opportunity to discuss the staff's review of the Framatome topical report. As previously stated, the staff did present their findings in the ACRS subcommittee meeting in June. We will be providing a high-level overview of the findings and conclusions today.

As mentioned in the opening remarks to the subcommittee, this is an effort that culminated over a four-year period. We appreciate Framatome's efforts in working with the staff and resolving some very difficult and complex issues.

The staff also appreciates the ACRS review of this topical report and the safety evaluation. The staff plans to issue the final safety evaluation either later this month or hopefully early in August. I look forward to your comments. Thank you.

MEMBER MARCH-LEUBA: Thanks, Greg.

From Framatome, Allan?

MR. McGINNIS: Hi. I'm Allan McGinnis, Licensing Manager for Framatome. I just want to take a second to thank the ACRS members for their time today and listening to the information that's going to be provided on our ARITA topical report and the NRC's draft safety evaluation.

ARITA is a methodology that is a key part of Framatome's efforts to upgrade our current PWR analysis methodologies with more sophisticated methods. As such, ARITA is going to be part of the basis and platform for further development in advanced fuel management and accident-tolerant fuel.

Framatome has a great deal of resources invested in ARITA. We are very anxious to get the final safety evaluation issued so that we can implement the methodology and its benefits for the industry. Thank you.

MEMBER MARCH-LEUBA: I wanted to remind everybody that this is an open session. So if we want to discuss questions, don't mention numbers and we'll stay safe.

MR. McGINNIS: I also want to thank the NRC for their efforts. They've put a lot of effort in, especially in these last several months to get the safety evaluation out and work with us on some last-minute issues. We appreciate that.

I'm going to go ahead and pass it on to Buck Barner, who has been the lead on this project for the majority of the development and review.

MR. BARNER: Thank you. This is Buck Barner. I'll be presenting today on ARITA.

I want to first off say thank you and

welcome. I'm excited to be here to talk about ARITA in this full ACRS meeting. I appreciate your willingness to be here and listen to us.

We're excited about ARITA. We're excited about what it brings to the industry. I'm looking forward to getting it implemented for the first time and excited to hear that we're moving forward with the SE. With that being said, we'll go ahead and move on now into the presentation.

So to start off, we'll start with an overview and go into some background and history of the topical. We'll look at the approval request and the range for applicability, describe the evaluation model, and finalize with a summary.

Next slide.

So what is ARITA? ARITA stands for ARTEMIS/RELAP Integrated Transient Analysis Methodology. As has been said, it is a non-LOCA method. It covers the Chapter 15 non-LOCA events except for the control rod injection.

It is a non-parametric approach, which is a novel approach for non-LOCA methodologies, and does employ a Monte Carlo sampling approach. It was developed using the guidance in SRP 15.0.2. It does have some elements of the in-depth process, but was generally followed using the 15.0.2 guidance.

Along with the standard non-LOCA methodology that's in there, we did address several additional topics. We'll talk about the motivation for that in future slides. That will also include methodologies for mixed-core power distribution control set points and fuel assembly reconstitution.

Next slide.

To give you some background and history, there's a little time line across the top here. As we move through the slides, we'll progress from left to right, starting up there with some advanced codes of method development.

In the 2006 time frame is when we began development of a new set of advanced PWR codes and methods, really with a focus on our neutronics solver ARCADIA, the core TH solver COBRA-FLX, and fuel performance code GALILEO. Around the same time in the industry, there was also a push to replace legacy methods, which was fortuitous and in good timing with what we were doing internally.

So with those two things in mind, we moved forward with a goal with several aspects in mind. That was to ensure that we're using our state-of-the-art modeling. We wanted to use our best codes and our best capabilities that we could. We wanted to implement the best practices we have from decades of our experience in all three of our regions -- the US, French, and German -- and really take advantage of that experience that we have.

We wanted to simplify the topical reports and interdependencies to reduce complexity. So we wanted to take what had traditionally been in several smaller topical reports and combine them into a single, consistent topical report. And we also wanted to do all this to facilitate development for future method development and continuing innovations.

After these first three codes were approved, our first methodology topical area was approved in 2020. And now today we're here talking about ARITA, which represents that final realization of our objective to bring innovation and improvement to the industry through our advanced codes and methods.

Next slide.

So a brief time line of the developments. We began working on ARITA and having pre-submittal meetings February 2015. We submitted the topical report in August 2018. After that time, we moved into the licensing and review phase.

We had the first set of RAIs on December '19. I won't go into the details. We had a lot of interaction with the NRC through RAIs and resolutions to those, audits back and forth, and ultimately over the last five years ended up with the draft SE, which was provided in April of 2023.

So we've been committed to this topical, committed to the industry to make sure that we get this thing through. We want to see it through, and we're excited to hear that the SE is at the final steps.

Next slide.

Moving forward, what does this mean and what does it bring to the industry? Now that we have advanced modeling of actual plant behavior, it helps us understand plant response and safety margins better. With this understanding, it actually helps us focus in areas where safety is really most important and ensuring compliance with the regulations.

So events where safety is really being challenged is where we're able to focus and really understand things better, have a better understanding of wherever margins are, and ensure that the plants remain safe.

Along with safety, it also helps us bring new value to the industry, things like addressing regulatory changes. Reg Guide 1.236, it actually addresses through area but any additional regulatory changes, we believe ARITA sets us up well for addressing any sort of new regulatory changes that may be coming.

Also core design and optimization, and then it also helps us move forward with advanced initiatives like advanced fuel management with increased enrichment and high burn-up. So all this allows us with our higher fidelity simulation to provide increased understanding and confidence in our plant safety.

Next slide.

So range of applicability, we're looking for a Chapter 15 methodology excluding control rod injection. As I mentioned before, we're also looking for applicability to our mixed-core methods, methods to analyze the power density limiting condition of operations and the core safety limit lines, power distribution control, and fuel assembly reconstitution.

It's generally applicable to Westinghouse pressurized water reactor designs, as well as CE pressurized water reactor designs. It does use approved critical heat force correlations that are not applicable to the correlations in the topical itself, but it will use approved critical heat force correlations.

It's mostly within the range of applicable constituent codes that are used within the methodology, which include ARTEMIS/RELAP, S-RELAP5, COBRA-FLX, and GALILEO.

Furthermore about the evaluation model, we talked about the codes. ARTEMIS is our 3D nodal simulator code, previously approved in ANP-10297. That's the ARCADIA topical report, which also includes the APOLLO2-A code. That was originally submitted and approved in 2013 and a supplement in 2018.

COBRA-FLX is the subchannel thermal-hydraulics code, which was approved in 2010. GALILEO is the fuel performance code. That was approved in 2020. And S-RELAP5 is our system thermal-hydraulics code that was previously applied in EMF-2310, originally in 2004 and again in 2011.

So using these codes, we have developed what we call a three-evaluation-model variance, which we'll refer to as the coupled system thermal-hydraulic and neutronics model, the 0D thermal-hydraulics model, and the static core evaluation model. We'll get into more detail on that in the next slides.

I'll review the coupling and the evaluation model variance. First we'll talk about the coupled EM. I have an illustration here on the left-hand side describing that, with ARTEMIS on the left and S-RELAP5 on the right.

Here we have ARTEMIS performing the 3D core neutronics modeling and simulation, with S-RELAP5 providing the system thermal-hydraulics simulation, with RELAP passing the mass flow and temperature and core outlet pressure to ARTEMIS, and ARTEMIS passing the core power back to S-RELAP5 through the sector heating rates.

That also can work with boron concentrations if they're important to the specific event. We regard it as an optional transfer, only if it's needed.

This is what's used in the coupled EM evaluation. It does this in a transient mode for the Chapter 15 transient analyses. There's a time-dependent multi-physics solution there. This model is used for cases where SAFDLs are being evaluated and non-SAFDLs.

MEMBER MARCH-LEUBA: If I understand correctly, ARTEMIS is the core simulation that has the

3D core inside the core. RELAP is used for the environmental conditions, simulators, all that.

MR. BARNER: Correct.

MEMBER MARCH-LEUBA: I don't want to call it the balance of plant, but it's kind of like the balance of plant. Let's call it the balance of the core.

MR. BARNER: Balance of the core.

MEMBER MARCH-LEUBA: Yes. Okay.

MR. BARNER: The second one is what we call the 0DEM. This is similar to what we've seen in legacy methods where data is generated in the neutronics simulator, or ARTEMIS in this case, which is provided to the simulator in S-RELAP5.

And S-RELAP5 is there to run stand-alone. Note that this is only used for non-SAFDL events, things like overpressure, secondary overpressure.

MEMBER MARCH-LEUBA: So this is when you're trying to provide figures that are mostly the primary boundary? Overpressure, temperatures, things like that, not core performance?

MR. BARNER: Correct. And then finally, we have the stack EM, which is Artemis used in stand-alone mode. That's used for events where the system thermal-hydraulics is needed.

So for static events such as a statically misaligned rod or a mis-load analysis, then the ARTEMIS core model is used. And that's also used for SAFDLs. You'll see that whenever a SAFDL is being evaluated, the ARTEMIS 3D model is what's being used to evaluate the standards.

A little bit of information on the statistical approach. Again, it's a Monte Carlo sampling approach. If you feel it better manages the complexity than the couple of non-LOCA transient analyses, there's a lot of moving parts to this. It allows us to sample that appropriately.

It uses a non-parametric approach based on Wilks method. It uses ordered statistics to make a 95/95 statement on the figures and merits.

For cases where there's events where multiple figures and merits are required for a single event, it does account for that and has an ability to handle multiple figures and merits from a statistical standpoint.

I'll also note that some parameters are still biased. This is a statistical method. There are places where we have bias and moved things to be conservative and be more consistent with the safety analysis and not so much a best estimate. So we do try to be realistic, and also provide additional margins and conservatisms through biasing some parameters.

Because we are using a non-parametric approach, we don't necessarily need to develop full distribution results. We can use a smaller number of realizations, say 59 cases, to make that actual statistical statement. And this is similar to what's been done in previous approaches I've presented to the ACRS and have been approved by the NRC before.

If you look over here on the left, this is just an illustration of a typical transient where you may see 59 realizations of a transient across time. And then you're able to use this to help with your statistical statement. This is the type of output that you might get from this.

MEMBER MARCH-LEUBA: Can you give me an example of when you use bias in parameters as opposed to sample uncertainty?

The way I understand it is when actually obtaining the certainty is too complicated -- one example you gave us during the subcommittee meeting was the position of the control rod when you have an asymmetric -- give us an example of when you bought it.

MR. BARNER: Peaking would be a place where we would use a bias. So if you're radial peaking, you'll use a bias in that case.

MEMBER MARCH-LEUBA: So you're using a limiting radial distribution?

MR. BARNER: We'll pick it up to the tech spec.

MEMBER MARCH-LEUBA: Instead of searching through your core for the words, conditions, you just place it on tech spec?

MR. BARNER: The details, I would have to have a full session. But yes, that's the general.

MEMBER MARCH-LEUBA: Okay. Thank you.

MEMBER ROBERTS: What do your outlet distributions look like? You talked about that in the opening session.

Would you say bifurcations or something kind of fishy happening past the 95 percent point in the distributions or do they look pretty much normal?

MR. BARNER: We have not seen anything like that, not something we would look at we saw something in distribution that looked strange, we would investigate it. We have not seen that in anything that we've done yet.

MEMBER ROBERTS: Is there anything in the methodology that looks for that?

MR. BARNER: There are some things we built in for certain parameters where we needed to look at certain areas, but that will probably be talked about more in closed soon. We do that as the methodology --

MEMBER MARCH-LEUBA: If you really want the answer to the question, we can sit up a clothes station in five minutes. So don't strain yourself. Just do not give any proprietary questions here. We will select the proper venue.

MR. BARNER: So I just want to summarize ARITA. It is a 3D coupled statistical, non-LOCA methodology applicable to see plants.

It represents a metered milestone in our commitment to state-of-the-art modeling and it incorporates decades of art industry experience, sets us up, sets the foundation for future development moving forward. Ultimately, the insurance confidence and plant safety compliance with all regulations and requirements is higher fidelity modeling.

I think that's all we have.

MEMBER HALNON: Just real quick, earlier in the presentation you talked about enhanced marbling, innovation, all these great things that you did. Being a layman at this kind of stuff, could you just give me one or two things that are innovative and enhanced in this?

You talked about Monte Carlo. I've heard that around forever. You talked about code coupling and I've heard that forever. So what is it that you've spent your last decade on?

MR. BARNER: Good question. So yes and yes. It's both a statistical approach and the core coupling, as well as the 3D transient.

The ARTEMIS simulator gives us the real feedback. The fact that we have that coupling feedback is what really gives us the ability to see things.

The Monte Carlo sampling is around that to take advantage of that coupling and be sure we're not breaking the coupling by doing things that the codes themselves wouldn't allow by taking unrealistic conditions and combining them with the other.

So when we were able to do that and couple them all together, that's what really took us to the next level. You said coupling is nothing new and the statistical piece is nothing new, but really it was the combining of those two pieces together --

MEMBER HALNON: So more power in smarter

brains putting it together and tying the knowledge together so that it all works together?

MR. BARNER: Yes.

MEMBER HALNON: All right. Thanks.

MEMBER MARCH-LEUBA: Any more questions from the members? Then let's switch with the staff.

I want you to introduce yourselves and speak clearly into the microphone.

MR. GEELHOOD: I'll go ahead and get started. I'm Ken Geelhood. I'm the Project Manager for Reports. I work in NRR and I'm Project Manager.

MR. HELLER: Hi. I'm Kevin Heller. I work in the Nuclear Methods and Fuel Analysis Branch. I'm obviously one of the reviewers on ARITA.

MR. LEHNING: And John Lehning. I'm also a technical reviewer in the Nuclear Methods and Fuel Analysis Branch.

MR. HELLER: All right. Let's launch into it then. Again, I'm Kevin Heller. With me here is John Lehning. It's our pleasure to be here in front of the full committee to present our review of the ARITA topical report. We certainly want to thank you for your time.

As I noted on the slide here, I just wanted to quickly point out Pacific National Northwest

Laboratory served as our consultant on this review and provided input that the staff used as a basis for the draft safety evaluation.

Next slide, please.

Okay. I think Framatome did a good job talking about the evaluation model and what that is. So really, the only thing I want to point out on this slide is that the staff's review of ARITA really did focus more so on the unique aspects of the methodology, so primarily the calculational procedure and the uncertainty treatment. I've got a couple of slides on that.

MEMBER MARCH-LEUBA: Is it the case that most methodologies have already been approved? The individual components, codes were already approved?

MR. HELLER: Yes, that would be correct. A large portion of these codes, yes.

MEMBER MARCH-LEUBA: And you leveraged that for approval? You didn't have to redo it?

MR. HELLER: Exactly. We'll see that.

Next slide, please.

Okay. To talk a little bit about the regulatory requirements and guidance, the full list of the regulations and guidance that the staff used can be found within the safety evaluation. The list here includes the key requirements and guidance.

In particular I wanted to point out Chapter 15 of the Standard Review Plan, which has guidance on how the staff ought to review evaluation models. And of course, the evaluation model development and assessment process or MDAP, which is in Regulatory Guide 1.203, that was used to structure parts of the safety evaluation in order to ensure it was a comprehensive assessment of the ARITA evaluation model.

With that, I think we can go to the next slide.

Okay. So a little bit of quick overview here on the review history. ARITA, the review can actually be broken up into four phases.

So if you actually proceed forward one more time? There we go. We can see the four review phases.

I really want to point out that ARITA was one of the most complex, challenging, and intensive reviews that I think either of us has really been a part of in our time at the NRC, not only because there were a number of first-of-a-kind issues but just given the complexity of the subject material, which in the past might have spanned several different topical reports.

We're seeing a lot of it really all in one here. A wide range of in-scope event responses, physical phenomena to be understood, limitations, and available data for doing non-LOCA statistical analyses, and the intricacies of the calculational procedure that Framatome proposed.

One of the main challenges which probably contributed to this is that to some measure, the sheer amount of information that was needed to be reviewed for this particular topical report. It's clear there was a lot of work and a lot of information that went into this.

Starting initially with beginning the review, we really didn't have enough information to begin drafting the safety evaluation or really know which way some of the identified issues would actually be resolved. So it took a bit of time to resolve a number of the RAIs. Many major issues would actually not end up getting resolved until a couple of years or maybe a little bit later within the review effort.

So to kind of put that in a little bit of perspective, the total volume of RAI responses was basically equivalent to another topical report. There's an additional 800 pages of information that was provided on top of the original submittal for about 1,600 pages in total.

What I want to point out at the last phase there, the draft SE phase, during that final phase there was not a lot of interaction that the staff had with Framatome in that particular portion because the staff was focused on completing the draft safety evaluation.

There were a handful of new limitations and conditions that were identified to close out open issues that Framatome's final RAI responses didn't resolve, but we left off the review at Framatome's request given the needed safety evaluation date.

So all parties understood that there were some issues that would probably have to get resolved through additional limitations and conditions. And that's why you can see a number of the limitations and conditions afford alternative approaches or justifications.

All right. Next slide, please.

Okay. So talking about the calculational procedures, what I'm going to try to do is high-level talk about what it is that we looked at and why we found it to be acceptable. When it comes to the calculational procedure, the key review issue that the staff identified was associated with assessing the ARITA methodology against regulatory guidance concerning suitably conservative initial conditions.

When applying this guidance, the staff considered what is the objective of the ARITA methodology with respect to its intended statistical statement and how is that going to be achieved. Ultimately, the tech specs and the safety analyses need to be in alignment.

In consideration of the staff's concern was that the as-proposed method would not provide conservative results for plant operation throughout the allowed operating domain. So during the course of the review, through interaction with the NRC staff and through RAI responses, Framatome modified the calculational procedure to establish suitably conservative event definitions.

The revised approach is really characterized -- I've got a couple of bullets here. Achieving a 95/95 or a 95 percent probability with 95 percent confidence tolerance limits over the allowed operating domain. Utilizing conservative sampling for highly influential parameters. Really, it's kind of a compromise between a traditional approach and a best-estimate sampling approach.

MEMBER MARCH-LEUBA: We talked in the subcommittee about this best-estimate plus uncertainty. We can eventually agree that this approach is best-estimate plus uncertainty plus biases, biases being conservative initial conditions, conservative power distributions. Do you agree on that?

MR. HELLER: Yes, I would agree with that. There were a number of biases that were utilized for conservatisms. I'll actually speak to that a little bit on a future slide.

MEMBER MARCH-LEUBA: If you state it in writing, we'll talk about whether calling it biases or penalty factors. What do you prefer?

To a mathematician, bias means a lot to you. But as a member of the public, I don't know what -- a bias is something psychological. We'll have a spirited discussion later.

MR. HELLER: It sounds like an interesting philosophical discussion.

MEMBER MARCH-LEUBA: It is the same thing, right?

MR. HELLER: I think effectively, as far as mechanistically at the end of the day, yes.

So when it comes to the calculational

procedure, it was through the establishment of this conservative event definition that the staff ultimately found the calculational procedure to be acceptable.

Next slide, please.

All right. So talking about the treatment of uncertainty. NRC staff utilized the guidance provided in Chapter 15.0.2 of the Standard Review Plan to review ARITA's methodology treatment of uncertainty. And per this guidance analysis, it should address all important sources of uncertainty.

When we're dealing with a statistic-based uncertainty method, one of the major challenges is developing sufficient data to support the uncertainty distributions for a wide array of phenomenon and processes. With that in mind, the NRC staff's review of the ARITA methodology, its uncertainty treatment focused on three areas.

The first area was the uncertainty and the input parameters, specifically the distributions and their technical basis. Staff assessed the uncertainty treatment for all the input parameters. As modified by RAIs and limitations and conditions, the staff found the treatment of those uncertainties to be acceptable.

The second area was the uncertainty in the calculated figures of merit, specifically the statistical basis for the method to derive the uncertainties in those figures of merit. Framatome proposed approaches to determine the uncertainties in those figures of merit on univariate and multi-varied conditions.

Staff assessed those approaches and found them acceptable because they appropriately determined a 95/95 tolerance limit. We actually discussed the details of this within the subcommittee meeting.

And then lastly, ARITA's overall statistical calculational procedure, specifically whether the procedure assures fidelity of the results. The staff reviewed the process proposed by Framatome and found it acceptable because it's a statistically rigorous approach that assures 95/95 tolerance limits are appropriately calculated.

And it appropriately avoids biasing of the procedure itself based on knowledge of the realizations, which could degrade tolerance limits. So it avoids that.

Next slide.

MEMBER ROBERTS: I guess I could ask you the same question I asked Framatome. With non-parametric statistics, it's possible to have the 95 percent, which is kind of good because you're actually getting more information about the potential worst-case scenarios that could happen. Sometimes depending on what you have, they could be very bad.

In this case, I'm guessing because you've got so many biased parameters, there aren't that many parameters left that could vary and conspire to come up with some sort of localized minimum or maximum. I was wondering if that's what you'd look for.

MR. LEHNING: This John Lehning from the staff. That was a concern that we brought up in the early part of the review.

Where I think we ended up, as you alluded to in the question, by focusing in on, as Kevin mentioned, some of the highly influential parameters and some of the biases that Framatome had input into its method, we didn't see evidence of that. And we don't believe there's a reason to think that there would be some special amount of bifurcation or spreading of this distribution beyond what's typical.

And we think, in fact, it might be less of an issue for this method because of the biases as compared to some other methods out there that may sample things even more broadly without some the biases and special focus on some of these highly influential parameters.

MEMBER ROBERTS: Okay. Thank you.

MR. HELLER: All right. Next slide.

Okay, so validation assessment. In practical terms, assessment and validation of an evaluation model involves checking the results of its computations against test facility measurements for reference data to provide insight into the credibility of the model's predictive capability for similar postulated conditions.

Framatome formulated the bulk of that assessment for the ARITA methodology in terms of the constituent codes, which we already mentioned. There was some interval validation comparison for those constituent codes, but there was also a lot of code or model-based validation provided.

It was often cited by past review efforts. That frequently made use of measured data and experimental data. So staff examined those past reviews in consideration of the intended application of the ARITA methodology and concluded they largely support the ARITA methodology.

In cases where there was limited validation for codes or models, staff concluded that some of the conservatisms that were present in the methodology were limitations and conditions acceptably compensated for this. Therefore, the staff concluded that the constituent codes have sufficient capabilities for modeling the phenomena that are relevant to the set of events within the scope of ARITA.

Talking about then the evaluation model variance themselves, for the 0D and static evaluation model variance, staff found those to be very similar to stand-alone treatments. And therefore, ultimately found those to be acceptable.

For the coupled evaluation model variant, the staff ended up assessing the information exchange between the coupled codes. And also Framatome provided integral assessments. Those results were found to be acceptable.

So the NRC staff ultimately then has reasonable assurance that the ARITA methodology has adequate capability of modeling Westinghouse and combustion engineering PWRs for the applicable events.

For supplementary evaluation model features, there's a series of what we refer to as supplemental evaluation model features for ARITA. These can serve as additional functionalities that

Framatome has incorporated into the methodology.

For the boron dilution, the mixed core treatment, and for fuel assembly reconstitution, staff found these were acceptable when consideration was given to code conservatisms in some limitations and conditions.

For the set points and the power distribution control, the staff ultimately found that these were evolutionary updates to existing methods, retaining the overall existing approaches. And therefore, the staff also found them to be acceptable.

Next slide, please.

At the conclusion of drafting the safety evaluation, staff imposed a total of 28 limitations and conditions. These can be broadly broken down as seen in the table here.

So there's about 16 of them that, really, they're there to assure appropriate treatment of uncertainties or the application of a statistical process. There's approximately ten that are there to assure appropriate application of the ARITA methodology.

And then there are two to assure that conservatisms which were cited as justification for the methodology are representative of future fuel designs in existing plants.

The staff found that these limitations and conditions were necessary in order to ensure the acceptability of the methodology based on a full review of the material that was contained in the docketed materials that Framatome submitted.

The last point then. After receiving the draft safety evaluation for proprietary review, Framatome did inform the NRC staff it intends to pursue resolution of some of the concerns underlying two of the limitations and conditions, 18 and 19, in a future regulatory review process.

We've had some dialogue as to possible avenues that could be pursued through that as far as the vehicles. For example, an update or addition to a topical report, inclusion within a plant license amendment request, or another avenue. We anticipate hearing from Framatome proposing a pathway in the near future for that.

MEMBER MARCH-LEUBA: Can you give us a high-level description of limitation conditions 18 and 19?

MR. HELLER: I was trying to do it off the top of my head, but we do actually have a back-up slide. Slide 10, I believe. It's actually after the conclusion.

MEMBER MARCH-LEUBA: I don't have it on mine.

MR. HELLER: You don't have it?

You should have it there. There we go. Okay.

MEMBER MARCH-LEUBA: We will have it if you put it on the record.

MR. HELLER: Yes.

MEMBER MARCH-LEUBA: If it's on the record, you have given us a copy.

MR. HELLER: Yes. So limitation and condition 18 deals with the axial peaking factor. Basically, what it says is licensees shall justify the approach that Framatome proposed for treating its uncertainty. Or they could conservatively estimate it via what's iterated within the limitation and condition, or they could justify another approach.

Really where this is coming from is the staff couldn't conclude from the material that was provided that Framatome's proposed approach directly followed.

MEMBER MARCH-LEUBA: Couldn't we have followed with a bias? That's typically for LOCA, right, or not?

MR. LEHNING: I think that in essence is similar to what we've found is being done here. So yes, there is a delicate balance here in terms of that.

We did try to take what we thought was somewhat of a bounding approach, especially on 19 because of the limitations and some of the data that underlies that item. So I think that's probably what I can say here in this session. We could go into --

MEMBER ROBERTS: I have a simple question. So you had 28 total limitations and conditions. I imagine some of those were legacy LNCs from the previous reviews because certainly you always have limitations and conditions.

Were some resolved from the past? Did a number of them just not apply? How many are new?

MR. LEHNING: I think they basically are new in general. And now the existing -- I think what you're referring to is there's previous safety evaluations that the staff has written on these other methods.

We have words in our safety evaluation that basically, the previous safety evaluations also have to be respected. So we didn't incorporate any of those existing limitations and conditions into our safety evaluation. The understanding is that those methods would be used.

MEMBER ROBERTS: So these are kind of in addition to the legacy applications. Okay. So really, in a sense there's more than 28. There's 28 plus the old ones too. Okay. Those ones don't get captured again in your SE?

MR. LEHNING: That's true. The way that the methods work, there probably are some -- for instance, if we were to look back at -- I think in particular, the ones that are the clearest are the ARTEMIS and GALILEO because those codes would be used in the same manner basically that they're being used in those code specifics or the topical reports that we reviewed.

The S-RELAP5 might be a little bit different in the sense of the realistic LOCA topical report. There are some aspects of that that may or may not necessarily -- some of the limitations might trivially be satisfied because they're not really applicable to non-LOCA analysis.

I'd say again, for the 28 items that are in the safety evaluation for ARITA, a lot of those deal with things that are specific to ARITA in terms of the uncertainty distributions and the calculation procedure for that evaluation model. So there is not a repeat of existing safety evaluation limitations and conditions into this safety evaluation we've written for ARITA.

MEMBER ROBERTS: Okay.

MR. HELLER: So for limitation and condition 19, this is somewhat a similar situation in that there was a proposed approach for treating the uncertainty with the axial peaking factor for a specific event. The staff could not conclude from the information that was provided before the review ended that the approach was appropriate.

MEMBER MARCH-LEUBA: The solution was the staff proposed what they considered to be a conservative treatment of the uncertainties, which Framatome considers to be too conservative; is that correct?

MR. HELLER: I think that would be a fair assessment.

MEMBER MARCH-LEUBA: Probably the solution would be the first LAR to be followed by a supplement. That seems like a path forward, but that's my opinion.

MR. HELLER: So really, the last thing I wanted to point out with this is -- you can see it there at the bottom of the slide -- the introduction of these limitations and conditions, they reflect only where we ended the review based on the docketed material. There was a cut-off date and it was understood that there might be some things that end up resulting. And these were two of them.

MEMBER MARCH-LEUBA: Okay. Thank you. Go back to your conclusions.

VICE CHAIR KIRCHNER: It begs a different question, a more generic question on limitations and conditions. How many of them are based on physics and how many of them are based on events? In other words, the methods of the first order don't know it's a streamline break.

You put those boundary conditions on the system in the analysis but is it -- do you see where I'm going with this? It sounds like a lot of limitations and conditions on top of those that already exist. So are they mainly because of inadequacies and treating physical events or just the range that the code is being applied to?

MR. LEHNING: Maybe I'll take a first crack at that. I don't think a lot of them have to do with basically physics or perceived inadequacies in the way that the physics of the code works.

I think we feel pretty comfortable with

those codes and the models. And any of the issues with those would have been identified in these previous reviews that we just discussed in response to a previous question.

So I'd say the majority, if not all, of these limitations and conditions really go to how the calculation ought to be done and whether we agreed with the specific way that Framatome said we're going to -- maybe assumptions, inputs, or other things like that much more so.

MR. HELLER: I would simply follow that up with -- I guess I would ultimately end up reiterating to a certain extent but using different words. Alluding back to what Framatome said, the novel aspects to this, these constituent codes are being applied in a particular manner.

So you can kind of think of it as there's a wrapper around those codes. It's the nature of that wrapper that -- those limitations and conditions exist to ensure that it's being applied appropriately.

VICE CHAIR KIRCHNER: Okay. I'm testing you because I didn't want to hear that it was something fundamental in the neutronics or whatever. It's more how it's tied together and how it's applied. Okay.

MR. HELLER: Yes. That would be the case.

Okay, conclusions then. Staff found that the ARITA methodology is acceptable for modeling in-scope Chapter 15 events from the Standard Review Plan. This includes all three of the ARITA evaluation model variants, the associated calculational process, and the statistical uncertainty methodology. The staff also found that the supplementary evaluation model features are acceptable.

MEMBER MARCH-LEUBA: Can you explain what those are?

MR. HELLER: Let me actually go back up in my notes here. You can stay on this slide.

MEMBER MARCH-LEUBA: Is it boron?

MR. HELLER: Yes. Boron dilution, the setpoints approach, the fuel assembly reconstitution, the mixed core treatments.

MEMBER MARCH-LEUBA: That's good, a good example. We have memorized the Section 3.8.

MR. HELLER: So the NRC staff found it was acceptable. And the staff's conclusions then are predicated upon two things.

First, the ARITA methodology being used within its proposed range of applicability. And that licensees acceptably address the limitations and conditions within the safety evaluation. That's it.

MEMBER MARCH-LEUBA: Thank you.

Any more questions from the members?

MEMBER HALNON: I just wanted to explore back on your schedule slide. You had an acceptance review. Did you take it back or just not accept it?

You said the first acceptance review was completed and then you had to supplement it. I know this doesn't affect the fact that everything is acceptable and you guys have done a good and thorough job of getting through this, but that's a long time and a lot of RAIs.

MR. LEHNING: Basically, what happened there was that we found information during the acceptance review that staff felt was not fully complete, and so there was a supplement. It wasn't --

MEMBER HALNON: So this was if you can provide this, then we can -- it wasn't extended beyond the 45 days?

MR. LEHNING: It wasn't rejected.

MEMBER HALNON: Okay.

MR. LEHNING: There's an option to reject it, accept it, or to accept it with a supplement. This went down the path of accepting it with a supplement.

(Simultaneous speaking.)

MR. LEHNING: Correct. That's true. It's at this very coarse level of detail. Is all the information we need there to do the full detailed review? And we found that there was not. This yellow bar represents basically the time to complete that.

MEMBER HALNON: As you ramped up your understanding of it, you can see it getting more and more detailed and more involved. Did you predict it was going to take that long? Or did it kind of just occur because of all of the technical complexities?

MR. LEHNING: We did. I think we had a pretty clear idea that the review schedule was going to be substantially longer than the standard two-year review period that we assume for topical reports, but there are a lot of uncertainties.

For example, there were some issues that even in -- I think there were three pre-submittal meetings for ARITA. There were some issues that we identified there very early on that continued to persist through the review and maybe didn't even get resolved until June 2022.

That was because, I think, the vendor in this case had a principal idea that they were in the right, that their approach was justified, and it took some time before we could get to that point where there was this agreement on going forward with that.

So yes, the exact length of time was not completely clear, but we had a clear idea from the very beginning. Framatome was aware of that when they got into the review.

MEMBER HALNON: I mean, the 19 meetings and all that, we've approved entire reactor plants in much less time. I can only imagine the complexity and how deep you guys got. It's good. I'm glad you presented that because it gives us the context of just how complex it was.

MEMBER MARCH-LEUBA: Any more questions from members, even the member on the phone line? I assume if you don't raise your hand, you don't want to talk.

MEMBER DIMITRIJEVIC: No questions, Jose.

MEMBER MARCH-LEUBA: Thanks, Vesna.

So at this point, I'd like to open the floor for members of the public. If somebody wants to place a comment on the open transcript, please do so now. I hear none.

Madam Chair?

CHAIR REMPE: Thank you.

At this point, I'd like to tell the court reporter we're going off the record for the rest of this meeting. Thanks for your support.

(Whereupon, the above-entitled matter went off the record at 4:20 p.m.)

Southern Nuclear

Lead Test Assembly (LTA) License Amendment Request (LAR) – 707th ACRS Meeting – July 12-14, 2023

Ryan Joyce – SNC
July 12th, 2023

Agenda

- Vogtle LTA Program Overview
- Requested Exemptions
- Summary of In-Pile Testing (ADOPT, AXIOM, Cr-coated cladding)
- Analysis

Vogtle LTA Program Overview

- Initial goals of program (within scope of proposed LAR):
  - Irradiate higher enriched fuel in a commercial reactor to generate data in support of future licensing applications
  - Obtain additional data for accident tolerant fuel (ATF) materials

- Future goal of program (outside scope of proposed LAR):
  - Support licensing applications for higher burnup fuels

- Four Westinghouse ATF LTAs with higher enrichment capable of higher burnup
  - Four rods in each LTA with enrichment up to 6 wt.% ^{235}U
  - **AXIOM**® high performance fuel rod cladding (WCAP-18546-P/NP-A)
  - **EnCore**® chromium coated cladding
  - **ADOPT**™ doped fuel material for non-IFBA (Integral Fuel Burnable Absorbers) rods (WCAP-18482-P/NP-A)
  - Standard (undoped) fuel material for IFBA rods

Vogtle LTA LAR– As Supplemented on Sept. 13, 2022 and May 5, 2023

- The license amendment requested the following changes:

  - TS 4.2.1, “Fuel Assemblies,“ will be changed to reflect the LTAs:
    - Placement in limiting core regions (except for control rod ejection transients)
    - Inclusion of advanced coated cladding with doped or standard fuel material, and
    - Having a maximum nominal ^{235}U enrichment of 6.0 wt.%

  - TS 3.7.18 “Fuel Assembly Storage in the Fuel Storage Pool,” and TS 4.3 “Fuel Storage” will be changed to reflect LTA:
    - Spent and new fuel storage restrictions
    - Allowance for maximum nominal ^{235}U enrichment of 6.0 wt.% in the New Fuel storage racks

Exemption Requests

- An exemption to §50.46 and §50 Appendix K was requested for use of **AXIOM** cladding
- A more restrictive embrittlement criterion was used in conjunction with the 17% maximum local oxidation criterion
  - **AXIOM** cladding topical presents the data in support of this application

Exemption Requests (cont'd)

- Licensing basis to change from §70.24 to §50.68
- Exemption is needed to §50.68(b)(7) to allow these LTAs to have greater than 5 wt.% ^{235}U rods
- Technical Justification:
  - Intent of rule (to preclude inadvertent criticality) being maintained
  - Administrative controls will be in place for temporary storage of LTAs in Traveller-B containers prior to LTA placement in their designated storage locations
  - Remaining §50.68(b) criticality requirements are unaffected and continue to be implemented
    - New Fuel Storage Vault
    - Spent Fuel Pool

Summary of In-Pile Testing (ADOPT, AXIOM, Cr-coated cladding)

- Millstone Unit 3 LTAs
  - AXIOM cladding
  - Completed 3 cycles of irradiation
- Byron Unit 2 LTAs
  - ADOPT pellets and Cr-coated Optimized ZIRLO cladding
  - Completed 2 cycles of irradiation
  - Hot Cell PIE after 1st irradiation cycle
- Significant BWR irradiation experience for ADOPT pellets (WCAP-18482-P-A) and PWR irradiation experience for AXIOM cladding (WCAP-18546-P-A) was used to establish corresponding irradiation properties databases

The primary novel feature that does not have prior commercial industry operating experience is the 4 fuel rods per LTA (out of 50,952 total fuel rods in the reactor core) with enrichment greater than 5 wt. % ^{235}U.

Analysis

- LTAs will lead the core during portions of steady-state operation and during some transient conditions
  - Leading the core = highest linear heat generation rate
  - LTAs do not establish core operating limits

- The LTAs and co-resident fuel will continue to meet all Technical Specifications (TS) 2.1 Safety Limits, 3.1 Reactivity Control, and TS 3.2 Power Distribution Limit requirements

- The analytical methods used to determine the core operating limits will be those previously reviewed and approved by the NRC (per TS 5.6.5)
  - No new methods were employed in the evaluation of the LTAs
  - Where pertinent, newer approved methods were used to confirm efficacy of the current licensing basis methods

- A separate LAR will be required to go above the licensed fuel rod burnup limit (not expected until 3rd cycle of operation)

Analysis

- **Small and Large Break LOCAs (SBLOCA, LBLOCA)**
  - BASH evaluation model (EM) for LBLOCAs and NOTRUMP EM for SBLOCAs are acceptable for evaluating the LTAs
    - LBLOCA meets all acceptance criteria per §50.46
      - **No impact** to peak clad temperature (PCT)
      - Maximum local oxidation (MLO), and core-wide oxidation (maximum hydrogen generation) meet acceptance criteria
      - Demonstrated inconsequential impact associated with radiative heat transfer of Cr-coated rods

  - SBLOCA analysis of record (AOR) is representative of LTAs
  - LTAs have a negligible impact for co-resident fuel for SBLOCA and LBLOCA

- **Non-LOCA/Transient Analysis**
  - No impact on AOR for transients dependent on core-average effects
  - Negligible impact of 4 LTAs on core-average heat transfer characteristics, decay heat, initial core stored energy
  - For events dependent on local effects (SLB, Locked Rotor, Loss of Flow, RWFS, Rod Ejection)
    - **No impact** due to LTA on approved non-LOCA codes, methods, or relevant acceptance criteria
    - The LTAs will not be placed in limiting core locations for Rod Ejection

Analysis

- **Source Term and Dose Consequences**
  - Utilized ORIGEN-ARP to generate core inventories for the LTAs
    - Performed parametric runs across a range of inputs (power, enrichment, burnup) that is broader than expected in operation
    - **No impact** to source term or dose consequences (AOR source term remains bounding)
    - Bounding source term will be confirmed per Reload Analysis on a cycle specific basis

- **Fuel Rod Performance and T/H Design**
  - The latest fuel performance models, PAD5 (WCAP-17642-P-A, Revision 1), ADOPT Fuel (WCAP-18482-P-A), AXIOM cladding (WCAP-18546-P-A), are used to explicitly model LTA features
  - Improved corrosion resistance of Cr coated rods are **conservatively** neglected
  - PAD5 approval up to 5 wt. % ^{235}U, but was validated to 13 wt. %
  - T/H – **No impact** to existing DNB margin (AXIOM cladding, ADOPT pellets, 6 wt. % ^{235}U, Cr coated cladding)

- **Core Physics**
  - Explicit modeling of Cr - cladding coating and ADOPT fuel pellets
  - Negligible neutronic impact from AXIOM cladding
  - No change to reload analysis methods, or the currently approved neutronic methods
  - **No impact** to neutronic modeling for fuel rods above 5 wt.% ^{235}U
    - Few rods per fuel assembly → neutron flux spectrum similar to currently operating core
  - Core monitoring with **BEACON** Core Monitoring System is unaffected

Analysis

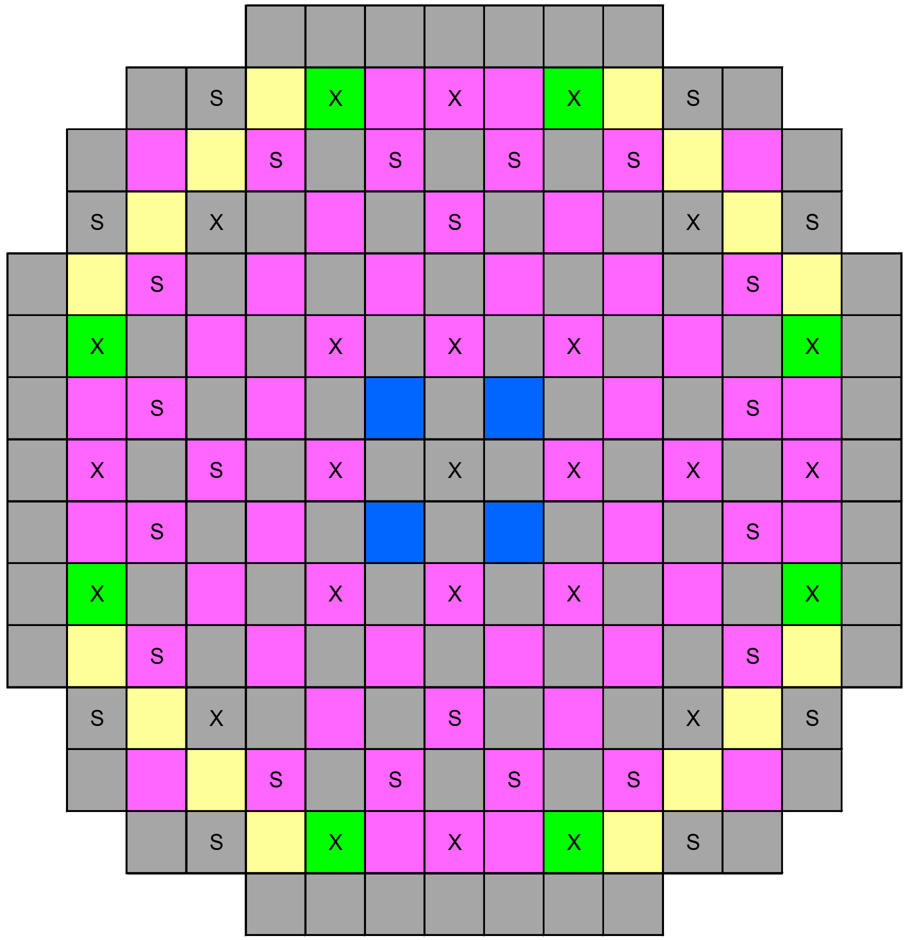
- **Criticality Analysis**
  - Unit 1 SFP Storage prohibited
  - Paragon was used for depletion calculations, while SCALE 6.2.3 was used for the rack criticality analysis.
  - Storage analysis performed via direct reactivity analysis
  - Storage not requiring burnup credit:
    - New Fuel Storage Racks
      - Demonstrated significant margin to storage limit including Dry, Fully Flooded, and Optimum moderation conditions
    - Unit 2 SFP two-out-of-four storage pattern
      - Demonstrated significant margin available
    - Both storage analyses credit IFBA, which will be confirmed on a cycle specific basis using the reload process
  - Storage requiring burnup credit:
    - “All-cell" storage limit of 64 GWd/MTU selected for LTAs (> 8% K_{eff} margin)
    - Unit 2 AOR “all-cell” storage pattern burnup limit is approximately 40 GWd/MTU
  - Multiple full pool misload event also analyzed using Technical Specification soluble boron limit of 2000 ppm showed acceptable results

Questions/Discussion

Acronyms and Terms

- ACRS: Advisory Committee on Reactor Safeguards
- AOR: Analysis of Record
- ATF: Accident Tolerant Fuel
- BWR: Boiling Water Reactor
- DNB: Departure from Nucleate Boiling
- GWd: Gigawatt Days
- IFBA: Integral Fuel Burnable Absorber
- LAR: License Amendment Request
- LBLOCA: Large LOCA
- LOCA: Loss of Coolant Accident
- LTA: Lead Test Assembly
- MLO: Maximum Local Oxidation
- MTU: Metric Ton Uranium
- NRC: Nuclear Regulatory Commission
- PCT: Peak Clad Temperature
- PIE: Post-Irradiation Examination
- PWR: Pressurized Water Reactor
- SBLOCA: Small LOCA
- SLB: Steam Line Break
- RWSC: Rod Withdrawal from Subcritical
- SFP: Spent Fuel Pool
- SNC: Southern Nuclear Company
- TS: Technical Specification
- wt.%: weight percent

BACKUP SLIDE – Example Loading Pattern



LTA Assemblies

Fresh Fuel lowest enrichment

Fresh Fuel mid-enrichment

Fresh Fuel highest enrichment (non-LTA)

Burned fuel

X Control Bank Location

S Shutdown Bank Location

NRC Staff Presentation
Accident Tolerant Fuel (ATF) Lead Test Assemblies (LTAs) Vogtle, Units 1 and 2

July 12, 2023

U.S.NRC
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Protecting People and the Environment

OPENING REMARKS

Michael T. Markley, Branch Chief

Division of Operating Reactor Licensing

Office of Nuclear Reactor Regulation

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LICENSING ACTIONS

- SNC requested:
  - Amendments to License Condition 2.D and TSs: (1) TS 3.7.18, “Fuel Assembly Storage in the Fuel Storage Pool,” (2) TS 4.2.1, “Fuel Assemblies,” and (3) TS 4.3, “Fuel Storage.”
  - Three Exemptions
    - 10 CFR 50.46 and 10 CFR part 50, Appendix K to allow the use of coated AXIOM cladding.
    - 10 CFR 50.68(b)(7) to allow greater than 5 weight-percent U-235

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UFSAR CHAPTER 15 ACCIDENT ANALYSIS

- Avg enrichment of the core is only increased 0.0003% by 4 LTAs resulting in negligible changes to source term or dose consequence for core wide accidents.
- Accidents with local core effects were re-evaluated using LTA data by the vendor and remained bounding with one exception…
- Rod Ejection Accident remained limiting for LTAs and Tech Specs will reflect that LTA core locations will be appropriately limited during LTA utilization.

CODES AND METHODS

- Neutronics
  - PARAGON and NEXUS
  - Average Assembly enrichment is still within the range of applicability for PARAGON.
  - Very little change in neutronic performance resulting from four enriched rods per assembly.
- Thermal Hydraulics
  - VIPRE-W, DNB Correlations, Rod Bow Evaluation Methodology
  - No impact to T-H performance or methodologies.
  - AXIOM is expected to perform at least as well as Optimized ZIRLO.
- Control Rod Ejection
  - The LTAs will not be placed in positions that have been shown to be limiting with respect to CRE.

FUEL ROD DESIGN

- AXIOM Cladding
  - Zirconium alloy that is expected to demonstrate better in-reactor performance compared to Optimized ZIRLO.
- Chromium Coating
  - Thin chromium coating for corrosion resistance with no impact to the thermal-hydraulic analyses.
- Increased Enrichment
  - Four rods per assembly enriched up to 6 wt% U-235.
  - Minimal impact to neutronic performance
- ADOPT Fuel Pellets
  - Higher density pellets containing chromia and alumina

FUEL HANDLING & STORAGE

- Enrichment exemption: four rods per LTA.
  - ADOPT increased TD actually adds more U235.
- Analysis
  - LTAs compared to Vogtle New Fuel Storage Rack & Spent Fuel Pool analyses of record.
  - LTA LAR credits IFBA whereas AORs do not.
  - LTA LAR credits extra BU
  - LTA LAR did not credit IFBA for SFP accident. NRC review did.
- Reasonable assurance 10 CFR 50.68(b)(2), (b)(3), and (b)(4) are met.

CONCLUSION

- The NRC staff determined that there is reasonable assurance that the health and safety of the public will not be endangered by allowing SNC the use of four ATF LTAs for up to two cycles operation in Vogtle, Unit 2.
- The NRC staff looks forward to the ACRS Letter Report.

ACRONYMS

- ATF – Accident Tolerant Fuel
- LTAs – Lead Test Assemblies
- NRC – U.S. Nuclear Regulatory Commission
- ACRS – Advisory Committee on Reactor Safeguards
- SNC – Southern Nuclear Operating Company
- NRR – Office of Nuclear Reactor Regulation
- DSS – Division of Safety Systems
- DORL – Division of Operating Reactor Licensing
- SNSB – Nuclear Systems Performance Branch
- SFNB – Nuclear Methods & Fuel Analysis Branch

ACRONYMS - Continued

- TS – Technical Specification
- CFR – Code of Federal Regulations
- U-235 – Uranium 235
- UO2 – Uranium Dioxide
- MWd – Megawatt-Day
- MTU – Metric Ton Uranium
- ZrB_2 – Zirconium Diboride
- IFBA – Integral Fuel Burnable Absorber
- LOCA – Loss-of-Coolant Accident
- EM – Evaluation Model
- RCS – Reactor Coolant System
- DNBR – Departure from Nucleate Boiling Ratio

ACRONYMS - Continued

- PCT – Peak Cladding Temperature
- REA – Rod Ejection Accident
- TD – Theoretical Density
- NFSR – New Fuel Storage Rack
- SFP – Spent Fuel Pool
- AOR – Analysis of Record
- 2oo4 – Two out of Four Configuration
- 4oo4 – All-Cell Configuration
- B10 – Boron 10
- ppm – Parts Per Million

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EPRI Report 3002018337, "Use of Data Validation and Reconciliation Methods for Measurement Uncertainty Recapture: Topical Report" (open session)

ACRS Full Committee Meeting

Lois M. James, Senior Project Manager
Office of Nuclear Reactor Regulation

July 12, 2023

Agenda

- Project History
- Purpose of the Technical Report
- Data Validation and Reconciliation (DVR)
- Data Validation and Reconciliation for CTP – MUR Relevance
- Review Scope Limitations
- Safety Evaluation Conclusion

Project History

Date	Activity	ADAMS Accession No.
1/27/2021	Electric Power Research Institute (EPRI) Transmitted "Use of Data Validation [DVR] and Reconciliation Methods for Measurement Uncertainty Recapture [MUR]: Topical Report“ for U.S. Nuclear Regulatory Commission (NRC) review and approval	ML21053A027
3/16/2021	NRC issued its completeness and withholding Determination for EPRI DVR for MUR Topical Report	ML21110A049
5/2/2022	NRC issued request for additional information regarding regulatory review scoping issues (RAIs)	ML22118A052
8/8/2022	EPRI submitted responses to Phase A RAIs	ML22223A052

Project History (cont.)

Date	Activity	ADAMS Accession No.
12/7/2022	NRC issued follow-up technical (Phase B) RAIs	ML22341A075
3/9/2023	EPRI submitted Phase B RAI responses	ML23066A242
4/13/2023	EPRI submitted supplemental information to the RAI 14 Response	ML23103A150
5/6/2023	NRC issued the draft safety evaluation for EPRI proprietary review	ML23088A184

Purpose of Technical Report

- EPRI Report 3002018337 describes a process for using a mathematical data validation and reconciliation (DVR) method for monitoring core thermal power and use of the methods for measurement uncertainty recapture uprates.

Data Validation and Reconciliation (DVR)

- What it is
  - A statistical analysis of multiple plant measurements, in aggregate, to provide accurate core thermal power
- What it can do
  - Reduce uncertainties associated with the core thermal power and allow plants to operate closer to the approved core thermal power
  - Reduce single failure vulnerabilities by using more instrumentation to determine core thermal power
  - Improve condition monitoring and condition-based maintenance by using the data points collected by plant equipment and defined physical relationships between measurements monitor equipment performance
- History
  - Used by the US and European nuclear power industry since 1999 to assess turbine cycle thermal performance, balance of plant feedwater flow metering, and accuracy of the plant calorimetric
  - Used DVR to increase power output in Europe

Data Validation and Reconciliation for CTP – MUR Relevance

Current Practice of Determining Core Thermal Power (CTP)

- Determination of CTP relies on feedwater flow measurements
- Inaccuracies in the direct measurement of feedwater flow have resulted in lost generation and potential overpower conditions
- In the past, ultrasonic flow measurement devices (UFM) have been used to gain measurement accuracy, but as technology improves, more can be done

Data Validation and Reconciliation (DVR)

- DVR uses statistical analysis of multiple plant measurements, to provide accurate CTP readings
- DVR can be used to reduce the uncertainties in the calculation of CTP and thus allow a licensee to produce more power via a measurement uncertainty uprate

Review Scope Limitations

- The safety evaluation
  - Is not a review of any software, logic flow, or numerical methods implemented in any particular vendor's packaging of the DVR methodology.
  - Is a review of the concepts of DVR and the steps needed to model the plant for the purpose of estimating feedwater flow and core thermal power mean and uncertainty.
- Specific evaluation of software, logic flow, or numerical methods implemented would be performed via a license amendment request or application.

Safety Evaluation Conclusion

Based on the

- NRC staff's risk assessment of the DVR results
- NRC staff's previous treatment of similar models and simulations
- NRC staff's previous evaluation of nuclear power plant process measurement uncertainty
- NRC staff's understanding of the DVR methodology
- NRC staff's previous treatment of the calculation of the feedwater flow rate and its uncertainty

NRC staff concludes that there is reasonable assurance that the DVR method as described in EPRI TR 3002018337 can be used to determine the core thermal power and the core thermal power uncertainty, provided all DVR conditions and limitations have been satisfied.

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ARITA ARTEMIS/RELAP Integrated Transient Analysis Methodology

Buck Barner

ACRS Full Committee, July 12th, 2023

Content

1. Overview
2. Background and History
3. Approval Request and Range of Applicability
4. Evaluation Model Description
5. Summary

Overview

- **ARITA - ARTEMIS/RELAP Integrated Transient Analysis Methodology**
  - Defines a methodology to analyze **non-Loss-of-Coolant (non-LOCA) events**
  - Uses a **non-parametric** statistical approach to make a 95/95 statistical statement for each figure of merit (FOM) using a **Monte Carlo** approach
  - Standard Review Plan (SRP) Chapter 15.0.2 was used as guidance in development of the method
  - Addresses additional topics, such as
    - mixed core
    - power distribution control
    - setpoints and
    - fuel assembly reconstitution
  - **Excludes Control Rod Ejection** (CRE) which is analyzed using AREA – ARCADIA Rod Ejection Accident Topical Report

Background and History

Advanced C&M Development → ARITA Development → ARITA Licensing → ARITA Future

- In 2006, Framatome began the development of a new set of advanced PWR codes
  - ARCADIA (ANP-10297PA, Revision 0 and Supplement 1, Revision 1)
    - Includes the 2D cross section code APOLLO2-A and the 3D nodal code ARTEMIS
  - COBRA-FLX (ANP-10311PA, Revision 1)
  - GALILEO (ANP-10323PA, Revision 1)
- Around the same time (2010) there was a push in the industry to replace legacy methods
- The goal was to develop new methodologies that:
  - Use **state-of-the-art** modeling
  - Implement best practices from decades of US, French and German **industry experience**
  - **Simplify topical report** interdependences to **reduce licensing complexity**
  - Facilitate future **method development and innovation**
- AREA (ANP-10338PA, Revision 1) was the first methodology topical approved.

➢ **ARITA represents the realization of Framatome's objective of bringing innovation and improved performance to the industry through advanced methods.**

Background and History

Advanced C&M Development → ARITA Development → ARITA Licensing → ARITA Future

- **Pre-submittal meetings held February 2015, June 2016, and July 2017**
- **The ARITA Topical Report was submitted August 2018**
- **The first set of RAIs (1-13) were transmitted to Framatome December 2019**
- **Responses to RAIs 1-13 were transmitted to the NRC April 2020**
- **All RAIs transmitted to Framatome April 2020**
- **Responses to RAIs 14-92 were transmitted to the NRC June 2021**
- **Audits and discussions between NRC and Framatome continued through April 2022**
- **Final updated responses to all RAIs to address reviewer comments were transmitted to the NRC June 2022**
- **The Draft SER April 2023**

Background and History

Advanced C&M Development → ARITA Development → ARITA Licensing → ARITA Future

- What advantage does this provide to the industry?
  - Enhanced modeling of the actual plant behavior leads to better understanding of plant response and the actual safety margins.
  - Increased understanding allows us to focus on the areas that are most important to safety and demonstrate compliance with all safety regulations and requirements
  - Bringing value and new opportunities to the industry
    - Address Regulatory Changes (e.g., RG 1.236)
    - Core Design Optimization
    - Advanced Fuel Management (AFM) - Increased Enrichment and High Burnup

➢ **Higher fidelity simulation and modeling provide increased understanding and confidence in plant safety**

Approval Request and Range of Applicability

- **Non-LOCA "Chapter 15" methodology, excluding CRE**
  - Mixed Core Method
  - Local Power Density Limiting Condition of Operation (LPD LCO) and Core Safety Limit Lines (CSLL)
  - Power Distribution Control (PDC)
  - Fuel Assembly Reconstitution

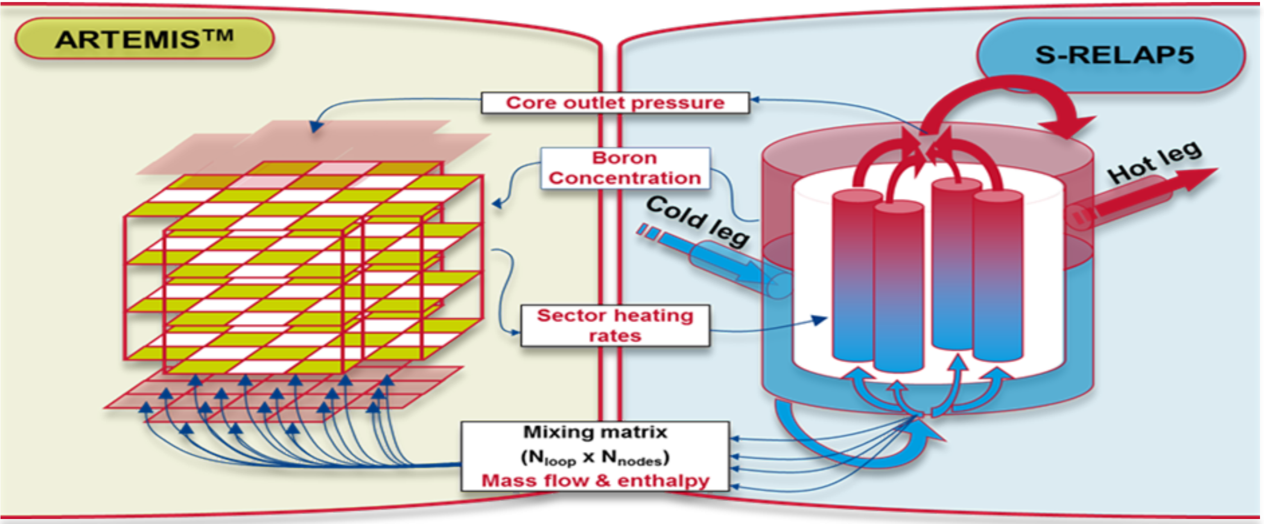
- Applicable to **Westinghouse** (2-, 3-, and 4-loop) Pressurized Water Reactor (PWR) designs and **Combustion Engineering** (CE) PWR designs
- Use of **approved Critical Heat Flux** (CHF) correlations
- Within the range of applicability of the constituent codes (**ARTEMIS, S-RELAP5, COBRA-FLX, GALILEO**)

Evaluation Model Description

- **Constituent codes**
  - **ARTEMIS** – 3D nodal simulator code previously approved in ANP-10297 (2013 & 2018)
  - **COBRA-FLX** – Subchannel core thermal-hydraulics code previously approved in ANP-10311 (2010)
  - **GALILEO** – Fuel performance code previously approved in ANP-10323 (2020)
  - **S-RELAP5** – System thermal-hydraulics code previously applied in EMF-2310 (2004 & 2011)

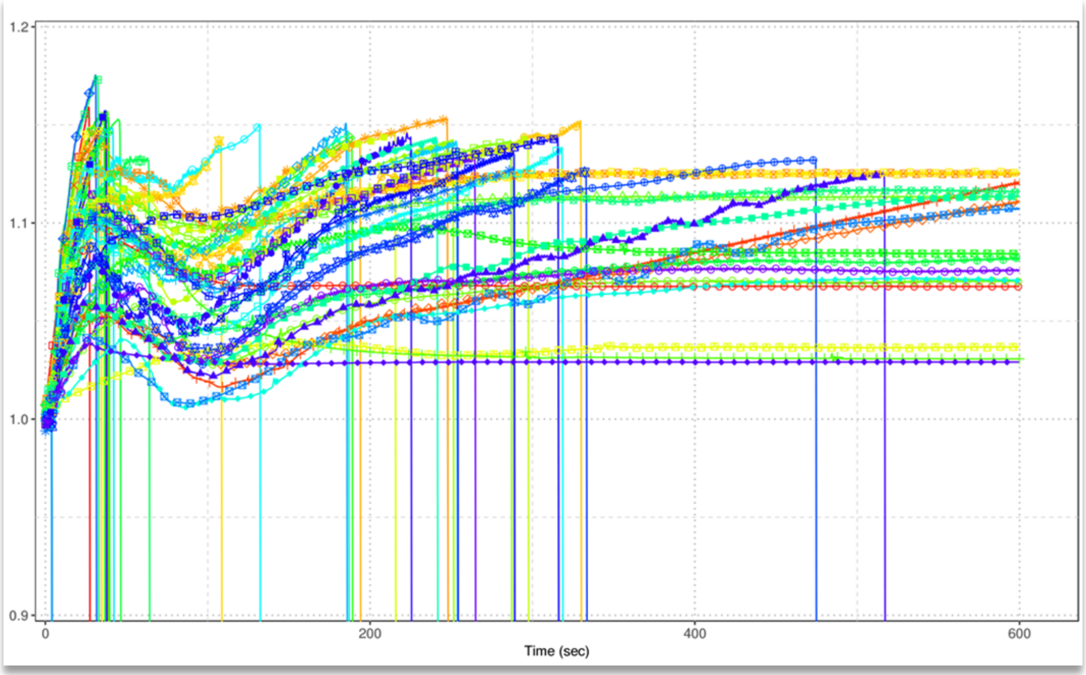
- **Evaluation Model (EM) Variants**
  - There are 3 EMs described in the ARITA topical:
    1) **Coupled** system-thermal hydraulic and neutronics model,
    2) **0D** system thermal-hydraulic model, and
    3) **Static** core evaluation model.

Evaluation Model Description



- **Code Coupling**
  - In the **Coupled EM**, ARTEMIS and S-RELAP5 are coupled together to solve time-dependent multi-physics problems (Specified Acceptable Fuel Design Limits (SAFDLs) and non-SAFDL FOM)
  - In the **0D EM**, point kinetics data generated in ARTEMIS is provided to S-RELAP5 (Non-SAFDL)
  - In the **Static EM**, ARTEMIS is used for events that do not require a system thermal-hydraulic solution (SAFDL)

Evaluation Model Description



- **Statistical Approach**
  - **Monte Carlo** approach better manages the complexity of the coupled Non-LOCA transient analysis
  - **Non-parametric** approach based on the Wilks method is used to make a statistical statement on the FOM.
  - Account for **multiple FOM**.
  - Some parameters may still be **biased**
  - No necessity to develop a full distribution of results
  - The statistical approach is used for all 3 EM variants described above.
  - **Similar non-parametric approaches have been presented before**

Summary – ARITA

- Is a **3D coupled statistical Non-LOCA** method applicable to CE and Westinghouse plants
- Represents a major milestone in Framatome's commitment to **state-of-the-art modeling** and decades of industry experience.
- Provides the foundation for **future development and innovation**

Ensures confidence in plant safety and compliance with all regulations and requirements through high fidelity modeling

Acronyms

AFM – Advanced Fuel Management

AREA – ARCADIA Rod Ejection Accident

ARITA – ARTEMIS/RELAP Integrated Transient Analysis

CE – Combustion Engineering

CHF – Critical Heat Flux

CRE – Control Rod Ejection

CSLL – Core Safety Limit Lines

EM – Evaluation Model

FOM – Figure of Merit

LOCA – Loss of Coolant Accident

LPD LCO – Local Power Density Limiting Condition of Operation

Non-LOCA – non-Loss of Coolant Accident

NRC – U.S. Nuclear Regulatory Commission

PDC – Power Distribution Control

PWR – Pressurized Water Reactor

SAFDL – Specified Acceptable Fuel Design Limits

SRP – Standard Review Plan

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Thank you

NRC Staff's Review of Framatome Topical Report ANP-10339P, *ARITA – ARTEMIS/RELAP Integrated Transient Analysis Methodology*

Open Presentation to the
Advisory Committee on Reactor Safeguards
July 12, 2023

K. Heller, U.S. NRC
J. Lehning, U.S. NRC
K. Geelhood, D. Richmond, T. Zipperer,
B. Schmitt, D. Engel, PNNL

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Introduction

- The ARITA methodology is a statistical approach for performing most Standard Review Plan (SRP) Chapter 15 reactor safety analyses
  - Not including LOCA and rod ejection
  - Applicable to conventional Westinghouse and Combustion Engineering PWRs
- ARITA involves three distinct evaluation model variants
- The codes used in the ARITA methodology have been previously reviewed by the NRC staff
- NRC staff's review focused mainly on the calculational procedure and uncertainty treatments

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Key Regulatory Requirements and Guidance

- 10 CFR 50, Appendix A, General Design Criteria, e.g.,
  - GDC 10, Reactor Design
  - GDC 15, Reactor Coolant System Design
- 10 CFR 50.36, Technical Specifications
- 10 CFR 50.67 or 10 CFR Part 100 Dose Limits
- Standard Review Plan, Chapter 15
- Regulatory Guide 1.203, Evaluation Model Development and Assessment Process

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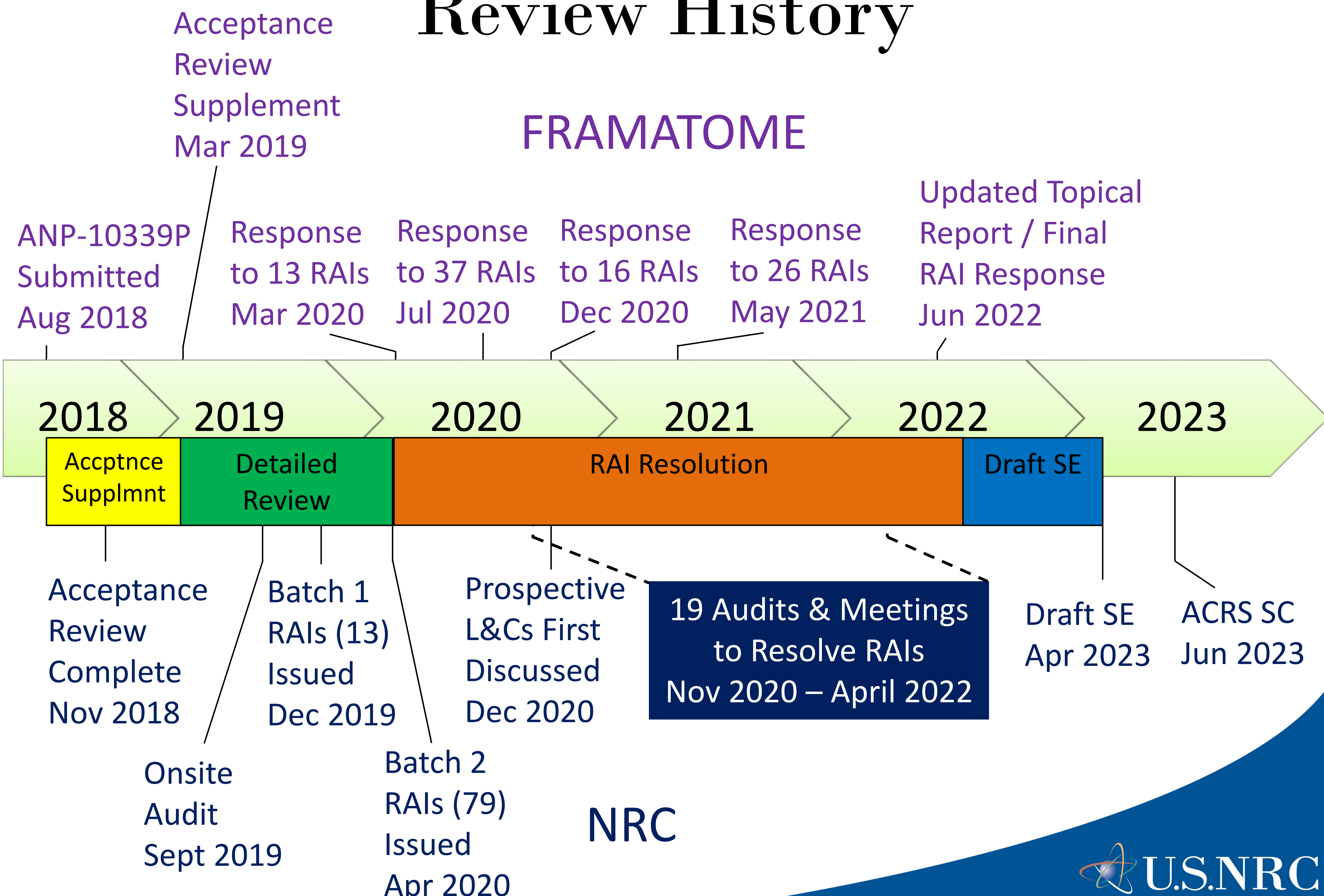
Review History



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Calculational Procedure

- Key review issue: Assessing ARITA methodology against regulatory guidance concerning initial conditions, for example SRP 15.0.I.6.C.ii:
  - *"The reviewer verifies that… initial conditions used in the analyses are suitably conservative…"*

- What is objective with respect to ARITA's intended statistical statement and how to achieve it?
  - Technical Specifications and safety analyses need to align

- Staff's concern: as-submitted method may not provide conservative results for plant operation throughout the allowed operating domain

- Establish a conservative event definition
  - Achieve 95/95 for postulated event over the allowed operating domain
  - Conservative sampling of highly influential parameters
  - Compromise between traditional and best-estimate sampling approaches

Treatment of Uncertainty

- The ARITA methodology uses a statistical sampling approach for uncertainty quantification
  - Wilks method: non-parametric estimate of 95/95
  - Challenge lies in developing sufficient data to support uncertainty distributions

- NRC staff's review focused on treatment of:
  - Uncertainty in input parameters
    - As modified, treatments found acceptable
  - Uncertainty in calculated figures of merit
    - Univariate and multivariate approaches acceptable
  - Overall statistical calculation procedure
    - Rigorous determination of tolerance limits
    - Appropriately avoids biasing that could degrade tolerance limits

Validation Assessment

- Constituent Codes
  - S-RELAP5, ARTEMIS, COBRA-FLX, GALILEO
  - Most validations cited past reviews
  - Conservatisms and limitations and conditions account for cases/areas of limited validation
- Evaluation Model Variants
  - 0D and Static
    - Very similar to prior stand-alone approvals
  - Coupled
    - Assessed information exchange and the provided integral assessments
- Supplementary Evaluation Model Features
  - Acceptable with appropriate conservatisms in place
  - Largely evolutionary updates

Limitations and Conditions

- A total of 28 Limitations and Conditions
  - Approximate breakdown:

Statistical Process and/or Uncertainty Treatment	16
Methodology Application	10
Conservatism Assurance	2

  - Framatome intends to address L&Cs 18 and 19 in a future review

Conclusions

- The NRC staff found the ARITA methodology acceptable for modeling in-scope SRP Chapter 15 events, including
  - all three ARITA evaluation model variants
  - the associated calculational process
  - the statistical uncertainty methodology

- The NRC staff found the supplementary evaluation model features described in Section 3.8 of its safety evaluation acceptable

- The staff's conclusions are predicated upon
  - the ARITA methodology being used within its proposed range of applicability in Section 13.0 of ANP-10339P
  - licensees acceptably addressing limitations and conditions in Section 5.2 of the staff's safety evaluation

Limitations and Conditions 18 and 19

- Limitation and Condition 18
  - Licensees shall justify Framatome's proposed approach for treating the uncertainty in the axial peaking factor (F_Z), conservatively estimate it, or justify another approach
  - Impetus: Staff could not conclude Framatome's proposed approach directly follows from the information presented during the review
- Limitation and Condition 19
  - For the stated event, licensees shall treat the uncertainty for the axial peaking factor (F_Z) for the post-scram main steam line break event via the specified conservative approach or justify an alternative
  - Impetus: NRC staff concluded Framatome did not adequately justify the proposed uncertainty treatment would well-characterize the event conditions
- Introduction of L&Cs 18 and 19 reflects only where we ended the review, based on docketed material

Future Framatome Plans to Address Limitations and Conditions 18 and 19

- NRC staff and Framatome have discussed potential pathways to address Limitations and Conditions 18 and 19 in a future regulatory review
  - Additional technical justification may reduce or eliminate impact of these limitation and conditions
  - Additional technical dialogue would facilitate timely resolution

- NRC staff will complete current review of ANP-10339P and issue final SE as scheduled

- Framatome to propose pathway to address technical issues in Limitations and Conditions 18 and 19 in a separate review process
  - e.g., expedited review of update/revision to the topical report, inclusion in a plant license amendment request, etc.