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UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

July 8, 2008

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

SUBCOMMITTEE ON POWER UPRATES

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TUESDAY, JULY 8, 2008

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ROCKVILLE, MARYLAND

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OPEN SESSION

The Subcommittee met in Open Session at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 9:00 a.m., John D. Sieber, Chairman, presiding.

SUBCOMMITTEE MEMBERS PRESENT:

JOHN D. SIEBER, Chairman
SAID ABDEL-KHALIK
J. SAM ARMIJO
SANJOY BANERJEE
MARIO V. BONACA
CHARLES BROWN
OTTO L. MAYNARD
WILLIAM SHACK
JOHN STETKAR

CONSULTANTS TO THE SUBCOMMITTEE PRESENT:

GRAHAM WALLIS

TOM KRESS

NRC STAFF PRESENT:

DAVID BESSETTE, Designated Federal Official

JOSEPH G. GIITTER, Director, NRR

JOHN G. LAMB, Senior Project Manager, NRR

BENJAMIN PARKS, NRR

SAMUEL MIRANDA, NRR

LEONARD WARD, Ph.D., NRR

AHSAN SALLMAN, NRR

SHEILA RAY, NRR

MATHHEW YODER, NRR

T-A-B-L-E O-F C-O-N-T-E-N-T-S

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

8:58 a.m.

1. OPENING REMARKS

CHAIRMAN SIEBER: The meeting will now to come to order. This is a meeting of the Advisory Committee on Reactor Safeguards, Power Uprates Subcommittee. I'm Jack Sieber, Chairman of the Subcommittee. Subcommittee members in attendance are Said Abdel-Khalik, Sam Armijo, Mario Bonaca, Otto Maynard, John Stetkar and Charles Brown. We have several other members who are in another meeting that will be with us shortly and I'd also like to welcome our consultants, Dr. Tom Kress and Dr. Graham Wallis.

Our Designated Federal Official for this meeting is David Bessette. The purpose of today's meeting is to consider the license amendment application to increase power of Millstone Unit 3 by seven percent including the safety analysis performed by Dominion Power and its contractor and a safety evaluation by the NRR staff.

The Subcommittee will gather information, analyze relevant issues and facts and formulate proposed positions and actions as appropriate for deliberation by the full committee in September. Participation in today's meeting has been announced as part of the notice of this meeting previously published in the Federal Register. Portions of today's meeting may be closed for the discussion of proprietary information.

We have received no written comments. However, we do have a request for time to make an oral statement from a representative of the public group regarding today's meeting which we will accommodate at the end of today's proceedings.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register notice. We request that participants in this meeting use one of the available microphones when addressing the Subcommittee. The speakers should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard.

The matter under consideration today is the stretch power uprate of Millstone Power Station Unit No. 3. The Applicant submitted its request as a request for an amendment to the plant's technical specifications and the staff reviewed this application under Review Standard-001, Power Uprates. This process is not a new license nor is it a renewal of a license and therefore issues discussed under consideration at the operating license stage or renewal operating license stage are not germane here. However, matters that are to be discussed under Review Standard-001 are germane here.

I would like to introduce at this time Joseph Giitter, Director of the Division of Operating Reactor Licensing in the Office of Nuclear Reactor Regulation. Joe.

MR. GIITTER: Thank you, Dr. Sieber.

MEMBER MAYNARD: I'm sorry. Mr. Chairman, a clarification. I believe we're meeting in July, the July meeting.

CHAIRMAN SIEBER: Right.

MEMBER MAYNARD: I think tomorrow we --

CHAIRMAN SIEBER: Yes. Right.

2. INTRODUCTION

MR. GIITTER: Thank you, Dr. Sieber. I worked with many of you before when I was in the Office of Nuclear Material Safety and Safeguards on the mixed oxide fuel fabrication facility and I look forward to working with you in my new capacity.

As Dr. Sieber indicated, we are in the process of -- Excuse me. Dominion Nuclear Connecticut Incorporated or DNC submitted a license amendment request for approximately seven percent stretch power uprate or SPU as we call it on June 13, 2007 for

Millstone Power Station Unit No. 3. The proposed SPU would increase the maximum authorized power level of Millstone 3 from 3411 megawatts thermal to 3650 megawatts thermal.

By memorandum from Frank Gillespie, Executive Director of ACRS, to Luis Reyes, then the Executive Director of Operations, dated April 23, 2008, the ACRS decided to review the proposed SPU for Millstone 3.

As the next slide shows we have conducted a very thorough review. Over the next several hours, I believe you will hear how we conducted that review. We had frequent communications with the Licensee. We had conference calls, letters and meetings and I believe that the frequent and effective communications between the NRC and the Licensee substantially facilitated our review.

Finally, there were several rounds of requests for additional information or RAIs issued to the Licensee. The RAIs were submitted as they were developed allowing the Licensee as much time to review and respond to the RAIs in different technical areas and that's a little different than we sometimes do it. Some of the more challenging review areas that you'll hear about in the next few hours include the fuel and core design analysis and environmental qualification.

As presented in the safety evaluation which was provided to the ACRS on June 11, 2008, there are currently no open technical issues in the NRC staff review of DNC's proposed SPU. I'm pleased with the thoroughness of the review conducted by the NRC. The staff had extensive interactions with DNC on these technical issues and was very cooperative in answering our questions which I think has led to our success in completing our review on the time frame that we did.

At this point, I would like to turn over our discussion to our NRC Project Manager to my left, John Lamb who will introduce the discussions.

MR. LAMB: Good morning. My name is John Lamb. I'm the Senior Project Manager in NRR assigned to Millstone 3 SPU. As you know, we only gave you 26 days to review the information. The staff realizes the significant burden this places on the ACRS members. On behalf of the staff, I would like to take this public opportunity to thank the ACRS for accommodating our schedule and reviewing the proposed SPU on a short turnaround. The staff greatly appreciates the ACRS members' efforts in this regard.

To quote the then ACRS member, Dr. Graham Wallis, at the NRC Commission meeting held on December

21, 2001, "one of our major activities right now and in the near future concerns applications for core power uprates and it's a very current topic. The impetus comes from the industry that sees considerable advantages to uprating the power and believes that they can do it safely. Many licensees are planning or have initiated these power uprates programs."

This statement by Dr. Wallis is as appropriate today as it was seven years ago. The staff's primary concern is safety. Our purpose is to convince you over the course of today that the staff's safety evaluation or SE for the Millstone Power Station Unit 3 SPU provides reasonable assurance that the health and safety of the public will not be endangered by operation of the proposed SPU. At the end of the day after hearing presentations from the staff and DNC, we hope that you agree with this and will recommend to the ACRS full committee on July 9, 2008 that the proposed Millstone Power Station Unit 3 SPU amendment be issued and reflect this in your letter report.

Before I go over the agenda, I would like to present some background information related to the staff's review of the proposed Millstone Power Station Unit 3 SPU. Millstone 3 is a Westinghouse 4-loop pressurized water reactor or PWR. The proposed SPU would increase the maximum authorized thermal power from the current license thermal power level of 3,411 megawatts thermal to 3,650 megawatts thermal. This represents an approximate seven percent increase from the current license thermal power.

On January 31, 1986, the NRC licensed Millstone 3 for full power operation at 3,411 megawatts thermal. Millstone 3 has a renewed license. The ACRS reviewed the Millstone license renewal at its 525th meeting and wrote a letter report dated September 22, 2005, recommending that the license renewal be approved. Millstone 3 license renewal was approved in October 2005 under NUREG 1838 titled "Safety Evaluation Report Related to the License Renewal of the Millstone Power Station Units 2 and 3." Millstone 3's renewed operating license now expires on November 25, 2045.

As far as the method of staff review, there is no specific guidance for SPUs since the staff has previously reviewed 61 SPUs and since there are no projected SPUs expected to be submitted to the NRC in the next five years, for the Millstone 3 SPU, the staff therefore used Review Standard-001, Review Standard for Extended Power Uprates as guidance along with internal document, Power Uprate Guidance, provided by memorandum from Christopher P. Jackson of the NRC to the Special Projects Branch, NRC, dated February 6, 2006 as well as experience gained from previously approved Westinghouse SPUs such as Indian Point 2 and 3 and Seabrook. The review standard includes a safety evaluation template as well as matrices that correspond to maintenance areas that are to be reviewed by the staff as well as specific guidance and the acceptance criteria that apply to those review areas.

Provided ACRS writes a letter report that states the Millstone 3 SPU should be issued, DNC has requested that the staff issue the proposed SPU amendment by August 15, 2008. DNC plans to implement the proposed approximately seven percent Millstone 3 SPU after completing the Fall 2008 refueling outage.

Basically, DNC's application followed the guidelines of Review Standard-001, Review Standard for Extended Power Uprates. DNC applied for an SPU by letter dated July 13, 2007. There were 33 supplements. A majority of these dealt with responses to the 107 requests for additional information of the staff questions. The staff spent a great deal of time reviewing the fuel and safety analysis as well as the environmental qualification.

After I conclude my remarks, DNC will provide an overview of their licensing approach as well as their modifications required and their implementation schedule. This will be followed by presentations from the Licensee and the staff on the following topics: fuel and safety analysis, containment analyses, electrical and grid reliability and lastly, flow accelerated corrosion or FAC. The bulk of the agenda is devoted to fuel and safety analysis.

So this concludes my presentation as far as the introduction. I would like to turn it over to Mr. J. Alan Price, DNC Site Vice President for Millstone Power Station. This is a position Mr. Price has held since January 2002. Mr. Price has approximately 29 years of experience in commercial nuclear power operations. Here is Mr. Price.

CHAIRMAN SIEBER: While we are changing speakers, I'd like to say that Dr. Sanjoy Banerjee and Dr. Bill Shack has joined us as members of this Subcommittee.

3. MPS3 SPU OVERVIEW

MR. PRICE: Good morning. My name is Alan Price, Site Vice President for Millstone Power Station and accompanying me this morning is Mr. Ron Thomas. Ron is the Project Manager for the Stretch Power Uprate Project at Millstone Unit No. 3.

Thank you all for the opportunity to come in and make our presentation this morning and to answer any of the questions that the Subcommittee may have for us regarding our request.

As previously stated, Millstone Unit 3 is a 4-loop Westinghouse PWR. A few of the more significant historical milestones include a license for commercial operation in January 1986, the NRC approved a transfer of the operating license for Millstone 3 to Dominion in 2001, March 2001, and then in 2005 we received a license renewal approval. With the license renewal approval and examples of other utilities before us completing the power uprates, it was natural for us to consider the power uprate for Unit 3.

I would like to mention that increasing the power for Unit No. 3 by about 80 megawatts electric it provide much needed electrical capacity to ISO New England which is the transmission authority that serves our area of the United States.

As the Millstone Site Vice President, I'm ultimately responsible for the safe and reliable operation of the units. At the very beginning of this project, we established an oversight committee, much in the similar fashion that others have done before us. We meet monthly and I generally chair the meeting.

The members of our executive oversight committee for the power uprate project include the most senior managers of our station as well as the most senior managers of our corporate organization. Typically, we review the progress of the major milestones associated with the project and any areas that may require additional focus. We review how effectively we're using operating experience from others who have completed their power uprates and we have provided special focus and attention on how effectively we have managed the margins of our power station as we've considered uprating the unit.

We set an expectation early in the project to preserve and enhance margins whenever possible. Several of the topics that we'll explore in more detail with you all today will demonstrate how we have effectively achieved this goal.

What I'd like to do now is turn it over to Ron Thomas who will go through the details of the project and then Ron will turn it back over to me at the end.

CHAIRMAN SIEBER: Thank you.

MR. PRICE: Yes sir.

MR. THOMAS: Thank you, Alan. My name is Ron Thomas. I'm been the Millstone Stretch Power Uprate Project Manager since the project kicked off in December of 2005. On the project team, when Dominion put together a project team, we took the approach that we did not want to do a turnkey operation that was solely performed by an outside company. Instead, Dominion assured that we have a significant project involvement and ownership so that we would end up with a robust margin management program and detailed in-house knowledge of all aspects of the project.

Of course, no utility has the resources or specialized subject matter experts to complete the effort by themselves. So we hired Shaw, Stone and Webster and Westinghouse to help lead us through the effort. These two companies have helped others at Seabrook, Comanche Peak, Ganay and Beaver Valley successfully complete their power uprate. They brought the operating experience of these power uprates and applied the OE to the Millstone Stretch Power Uprate.

CHAIRMAN SIEBER: Now the plants that you mentioned, some of those were at the same increase in power as Millstone. For example, Beaver Valley I think is seven percent also.

MR. THOMAS: In terms of percentage, yes. In terms of megawatts-thermal, the closest example would be Seabrook.

CHAIRMAN SIEBER: Yes. Okay.

MR. THOMAS: Let's see. A project team strength that we would like to point out is that we had a licensed senior reactor operator, SRO, as a full-time team member since the beginning of the project. The SRO was involved with the early analysis phase of the project, the licensing phase of the project and now the station modification and implementation phase of the project.

The licensed SRO was the focal point of the analysis portion to ensure that the Design Engineers did not unintentionally believe that there was sufficient margin on a system that Operations believed would cause a challenge for them to operate. In other words, the SRO made sure that the Design Engineers did not leave the Operations with components and systems that would be too difficult to operate at the new power level.

MEMBER SHACK: Now you have enough

capacity in your steam generator, right, because I know you haven't gotten a replacement? So you lose a tube you're --

MR. THOMAS: Correct. We do have excess capacity and excess margin in the steam generator.

MEMBER SIEBER: You have that now.

MR. THOMAS: And we looked at it for the full tube plugging that were analyzed for the reminder of the life of the power station.

CHAIRMAN SIEBER: Yes, that has some implication to Bill's question which is that your t hot temperature is something like 619 or 617 as I recall which is pretty hot for a t hot temperature. Have you consider that in the Alloy 600 issues in your plant and if you have, what have you done?

MR. PRICE: We have considered that as part of our application and when we get into some of the more detailed part of the discussion today, I think we'll be able to answer all of your questions.

CHAIRMAN SIEBER: Okay. But I still note that that's pretty hot for a reactor outlet temperature.

MEMBER MAYNARD: Are those thermally treated, the TT Alloy 600 or is that the conventional, the early version?

CHAIRMAN SIEBER: They model that.

MEMBER MAYNARD: I'm interested in overall what the temperatures are going to end up. The 619 is consistent with the later model, Westinghouse PWR, large PWR.

CHAIRMAN SIEBER: Right.

MEMBER MAYNARD: That's 618, 619 is what most of them are operating at now. So I do think it's a good question for later as to exactly where we are getting the power and what the temperatures end up being there.

MR. THOMAS: We'll make sure that that topic is adequately covered in the technical brief later.

MEMBER MAYNARD: Okay.

MEMBER BROWN: Jack, one observation in their general overview of their license request or their uprate request, the analysis. It was they have a table in there that talks about core outlet and vessel outlet and the core outlets are like two 628 in a couple of cases and the vessel outlet is about 622, 623. So just pointing out it's a little higher than the 618 to make the point.

CHAIRMAN SIEBER: Right. Pretty hot though.

MEMBER BROWN: Yes. Fairly toasty.

CHAIRMAN SIEBER: I guess the only reason why I bring that up is the nickel alloys are temperature sensitive and therefore the cracking and so forth that may go on accelerates this temperature acceleration. So that's a concern for me.

MR. PRICE: As part of our reviews for this project what we did pretty random analyses assuming a spectrum of t aves, and a spectrum of t colds and t hots. So what we've done is all of our analyses would be based on the worst case conditions. That does not necessarily mean that they would be the protocols that we would expect to operate.

MR. THOMAS: So we'll make sure that that question is specifically answered.

Another project team strength that I would like to point out is that we had a full-time engineer dedicated to margin management and operating experience. This engineer, Mr. Larry Salyards, helped guide the project team members as they prepared to review a system or a component by providing them with applicable operating experience from other power uprates. Then when the analysis was completed for any component or system, Larry would then ensure that the engineering documents contained a discussion of margin. In the examples we were just using with steam generator, he would look at that steam generator evaluation report, make sure that it clearly identifies what is the current margin with the full temperature swing that we're talking about that we could potentially operate within that t ave window and he would make sure it describes what is the current margin for that equipment, then what is the change in margin for that equipment at the new proposed power level of seven percent more power.

CHAIRMAN SIEBER: Do you have a margin manager all the time or just for the power uprate?

MR. THOMAS: We do have a -- Millstone does have a margin management program. It's a program that existed prior to us starting the project. What we did, that program was what I'll characterize as a small program with certain focus topics in which there was adequate industry information and adequate information at the power station.

What we've done is with a full-time margin management individual, he worked with the program owners and the individuals with that team and he met with them and brought to them every margin identifications we did and every changes in margin that we performed. So he and our team and our project is leaving behind a better, well-defined margin management program and leaves us with more knowledge of how the power station operates today in addition to how it's going to operate with the power level. So we've enhanced the knowledge of the power station.

CHAIRMAN SIEBER: I think that's important and I wish all plants, and I don't know that they don't, but I wish all plants had a margin manager because middle managers and engineers have a tendency to what to use the margin and if too different bodies are using the same margin, you have potentially some safety issues. I'm pleased that you have that.

MEMBER MAYNARD: One of the things I'll be interested in when we get into talking about some of the margins is how much of it is through the more detailed analysis versus how much of it is either through plant modifications or operating parameters that maintain the margin.

MR. THOMAS: And we will be covering some of these topics during the presentations that we have and certainly ask the individual doing the presentation related to margin specifically and we have slides that will describe what is the current value, what's the limit and then what is it going to be with the new proposed power level.

So that was one of our legacies that we did want to leave behind. If nothing else, we wanted to leave behind the best margin management program in the industry and I believe we've achieved that objective.

License core power level, today Millstone 3 is operating at its original 1986 license core power level of 3411 megawatts thermal. In the past 22 years of safe operation at Millstone 3, we have observed that the NRC has approved over 120 power uprates in the industry and 23 power uprates for our peer 4-loop Westinghouse units.

Half of those peer uprates were at measurement uncertainty recapture, MUR power uprates, and half were stretch power uprates, SPUs. Some of these units were approved for both MURs and SPUs.

More than five years ago we began to explore the concept of a power uprate. At that time, we installed an ultrasonic flow meter to more accurately measure reactor power level. We installed a system which allows up to a 1.7 measurement uncertainty recapture, MUR, power uprate.

As we studied the margins available on the primary and secondary sides of the power station, we realized that we had much more margin available than we expected. We determined that almost all components and systems had so much margin that we could easily and safely implement a seven percent power uprate while maintaining adequate operating margin without replacing any major components. The limiting component was the electrical generator and we'll hear more about that later today. It needs to be replaced or modified if we were to desire to increase the reactor power level beyond seven percent.

CHAIRMAN SIEBER: Why did you decide to replace the main feed pump turbine? The capacity or --

MR. PRICE: I'll be happy to talk about that right now if you would like. When we looked at the increased flow required for the power uprate and we started looking at the weak links in the system the steam turbine for the main feed pumps became the weak link. We had an option of doing additional welding on the first stage steam turbine blades. That would have given us the margin we were looking for.

CHAIRMAN SIEBER: I thought you --

MR. PRICE: But we decided to not pursue that option. Instead we went with the OEM and we decided to purchase new steam turbines to give us the operating margin that we were looking for.

CHAIRMAN SIEBER: So these are complete turbines and not new internals.

MR. PRICE: These are the new turbines for the main feed pumps. That's correct.

CHAIRMAN SIEBER: They were run at a higher speed.

MR. PRICE: That is correct.

CHAIRMAN SIEBER: By -- The old ones are 4700 or 4800 rpm.

MR. PRICE: Yes.

CHAIRMAN SIEBER: And the new ones are 5100.

MR. PRICE: That is correct and they are already on site and ready for installation this fall.

CHAIRMAN SIEBER: Right, and you're going to do all the standard things like checking alignment and measuring vibration and all that stuff when you start up.

MR. PRICE: Yes sir. That's correct.

CHAIRMAN SIEBER: Okay.

MR. THOMAS: And that's an example of where we saw a reduced margin that we could have worked around but we wanted to improve the margin as much as possible and went with new equipment.

So we began a 15-month analysis effort to confirm the seven percent potential assumption. The results of that detailed analysis concluded that seven percent was the correct selection for the new power level for Millstone 3.

CHAIRMAN SIEBER: I have another question that relates to the feedwater system. Since you're using steam turbines, the signal that tells how much feedwater demand there should be goes to the turbine throttle valves and your feedwater regulating valves essentially maintain constant differential pressure.

MR. PRICE: Yes sir.

CHAIRMAN SIEBER: Is that the case?

MR. PRICE: Yes sir.

CHAIRMAN SIEBER: Okay.

MR. PRICE: And as part of the modifications that we'll be making to the plant will be the rescaling of those components on the secondary side, the main feed as well as the main steam and the steam dump systems.

CHAIRMAN SIEBER: Do you plan to change the trim in the feedwater regulating valves or is what you have good enough?

MR. PRICE: What we have is sufficient.

CHAIRMAN SIEBER: Usually with electric pumps, you end up with problems with the feed reg valves. Steam driven pumps usually don't have that problem.

MR. PRICE: Yes sir.

MR. THOMAS: Once we understood the potential power level for Millstone 3, it became a matter of determining the licensing strategy, to reapply apply for an MUR followed by an SPU like Comanche Peak or should we apply for an SPU followed by an MUR like Seabrook? We decided on a single step approach applying for a new power level in a single stretch power uprate license amendment request.

We did retain the two percent uncertainty margin in determining reactor power level. So we are not asking for a combined approval of an MUR and an SPU. This is just an SPU. We do not intend to ask for an MUR in the near future and that is because we have a limiting component at the electrical generator that we'll hear more details about later and because of that limiting component, even for an MUR, we would have to replace that component or modify.

(Simultaneous conversations.)

CHAIRMAN SIEBER: What about the rest of your electrical system and grid system as far as stability is concerned? Is that all sized probably for --

MR. PRICE: Yes sir.

MR. THOMAS: Yes sir and --

CHAIRMAN SIEBER: I noticed that in your discussion in the SER it said that you improved your process for the uprate that really made your improvement at the voltage at that end of the system. Correct?

MR. THOMAS: We do have a topic specifically for electrical power and the electrical generator that is coming up.

CHAIRMAN SIEBER: I might save my questions for that.

MR. THOMAS: The best person to ask that are the two subject matter experts that I'm bringing up here later.

CHAIRMAN SIEBER: -- ask people who --

(Laughter.)

MR. THOMAS: Selecting the new power level was based on Operations' input. An engineering analysis showed no major modifications were necessary up to 100 percent above the current power level at 3650 megawatts thermal. Most of the station modifications that are necessary to achieve the new power level are changes to the licensing design basis document, the document design calculations, design drawings and operating procedures.

DR. WALLIS: Can I ask if you have any idea how much of an increase in power you could get by changing the generator with the existing system? Do you have any idea what that would be?

MR. PRICE: Is your question what would our next most limiting component be?

DR. WALLIS: Yes and how much power would that get you?

MR. PRICE: I don't know the answer to that question.

MR. THOMAS: I can answer that question later during the day.

DR. WALLIS: It must be tempting to go after this measurement uncertainty.

CHAIRMAN SIEBER: It's a lot of money.

MR. PRICE: It would be if we were prepared to replace the new generator at this time which we're not.

DR. WALLIS: Right. I just wonder how much that the nuclear end of it could stand in terms of an uprate.

MEMBER ARMIJO: Is the limitation to the main generator, the turbine or condenser or the generator itself.

MR. PRICE: Right now, the limiting component is the main generator.

CHAIRMAN SIEBER: So you will actually be controlling power on the basis of current flow and temperatures in the generator. That will be your summer limit.

MR. PRICE: Basically, that's one way to look at it.

CHAIRMAN SIEBER: As opposed to condenser vacuum or --

MR. PRICE: Yes sir. That's one way to look at it.

MEMBER BANERJEE: With regards to this plant, is it critical heat flux limited or LOCA limited? All levels.

MR. THOMAS: We'll leave that for our subject matter experts that are coming up later.

CHAIRMAN SIEBER: Which is after the break.

MR. THOMAS: With that, Alan, do you have any final comments?

MR. PRICE: I do. I know that the Committee will be interested in some of the changes that we'll be making for the power uprate, the proposed power uprate. We've already talked about the most significant hardware change which is the main feed pump steam turbines. In addition, we're making a number of changes in some of our safety systems. An excellent example is the logic change that we have proposed in our emergency core cooling system. We plan to implement a new permissive, P19, that will provide additional protection for the inadvertent pressurizer overfill on a spurious safety injection and we'll get to that in more detail as the technical part of the presentations continue.

Also we're making a modification to our control building emergency filtration unit. This takes out of the question operator reaction post fuel handling accident and puts an automatic system in our control building fuel ventilation system. We're making a variety of set point and scaling changes for the feed pumps' feed control, pressurizer level program, turbine generator controls, steam dump and load reject controls and the like. But as we've indicated before, right now the most major physical plant change is the main feed pump turbines.

CHAIRMAN SIEBER: So other than rescaling some instruments on your control panel and changing some set points here and there and some of the logic, you aren't making any changes to the layout of the control room instrumentation and controls.

MR. PRICE: Yes sir. That's correct. Now we are replacing our main transformers onsite. That was not part of the power uprate project. It was because the main transformers are greater than 20 years old. We're watching the OE from the industry. We wanted to stay ahead of the transformer failures that others are seeing. So we are replacing our transformers. We're going to three single phase with an installed spare and they are all sized to handle this power uprate and beyond.

CHAIRMAN SIEBER: What's your current transformer? All three phases and one can?

MR. PRICE: What we have is we have two transformers, main step-up transformers, for Unit 3. Each are about 60 percent and each of those two transformers or both of those two transformers are three phase units.

CHAIRMAN SIEBER: Elevated gas levels in the oil of either one of them?

MR. PRICE: Yes sir. That's part of the impetus for us to replace the transformers. So we will be taking ownership of the new transformers in September of this year.

CHAIRMAN SIEBER: How much of the

switchyard do you own, those transformers and the circuit breakers and your station service buses?

MR. PRICE: We own from the house side of our transformers to the switchyard.

CHAIRMAN SIEBER: Okay.

MEMBER ARMIJO: Another modification and I'm sure you'll get into this later is the rod control system logic. You're taking out the automatic rod withdrawal.

MR. THOMAS: That is correct.

CHAIRMAN SIEBER: I'm sure you'll discuss that later, too.

MR. PRICE: Yes, sir. We will.

CHAIRMAN SIEBER: Everybody has done that.

MR. PRICE: So I trust some of the examples that we've talked about provide an adequate overview of the type of modifications that we'll be proposing for our power station.

Mr. Chairman, that does conclude our overview presentation and we do recognize that since we are not going for the extended power uprate, it's not part of the normal business for you all to take a look at our power ascension testing. We do have a presentation prepared to address those questions if you would like for us to do those and throughout the day we'll be happy to accommodate that. You can request it anytime.

CHAIRMAN SIEBER: Thank you very much.

MR. PRICE: Yes, sir.

CHAIRMAN SIEBER: According to the schedule, we would take a break now. But I think it's too early to do that. On the other hand, the next topic is Fuel and Core which is a 45 minute presentation. I think we can get it in.

MEMBER SHACK: I think we'll catch up with the schedule.

CHAIRMAN SIEBER: Sooner or later. Thank you very much and let's have the other -- We'll take a break later.

4. FUEL AND CORE

MR. KAI: Good morning. My name is Mike Kai. I'm a Principal Engineer at Dominion. I've worked in Safety Analysis since 1970, a long time. I'm going to go over the fuel and safety analysis. First, I would like to introduce my partners here. Albert Gharakhanian is the Engineer who worked on the containment analysis, did most of the containment analysis and coordinated the containment analysis effort and Sandy Andre who is our contractor from Westinghouse who was responsible for the transient analysis. So hopefully between the two of us, we will be able to answer any questions.

I did try to take some notes of things that you wanted to bring up. I believe they're all covered in my presentation, but please don't hesitate to ask.

CHAIRMAN SIEBER: You can count on us.

MR. KAI: Okay. The other thing I'd like to say is how I set up my presentation. There's a lot of information in it, a lot of numbers, a lot of results. I did not intend to go over each item. I will try to highlight what I think is significant. But please feel free to ask about anything that's on the slide. Okay. Any questions?

(No verbal response.)

This slide just shows what I'm going to cover. So there's an outline of my presentation and I think I'll just skip directly to the fuel design.

The thing about the fuel I think to understand is that we have not changed the fuel design. We're actually going to go with what's our current fuel system that we're using and actually add our SPU power level, we have a full core of RFA-2 fuel. So we have no mixed core issues. That clearly simplified our analysis and we have experience with the fuel assemblies. So in general, I foresee no fuel issues.

DR. WALLIS: Did you change the burn-up or anything? What did you change?

MR. KAI: We are increasing the feed and that's how we're getting extra energy and we have a little slide that shows what we're doing. But burn-up is unchanged. We're achieving the extra energy by adding --

CHAIRMAN SIEBER: More assemblage.

MR. KAI: More assemblies. We're not really going to have what you would call transition questions. This is going to be like our normal cycle of fuel replacement.

CHAIRMAN SIEBER: This is Westinghouse 17 X 17 fuel.

MR. KAI: Right. This one is 17 X 17.

CHAIRMAN SIEBER: That's what's in the core now.

MR. KAI: That's correct.

CHAIRMAN SIEBER: That's what your transition core will be.

MR. KAI: Yes. Correct.

CHAIRMAN SIEBER: Okay. Beyond that you're not -- Right.

MEMBER BANERJEE: And you have the zero cladding.

MR. KAI: That's correct. Zero cladding.

CHAIRMAN SIEBER: One thing I would point out while we're discussing the fuel and core design is that every reactor at every refueling has to prepare a reload safety analysis which is sent into the NRC that's specific to the core that will be started up after that refueling outage. So when we discuss transition and equilibrium cores here, we're not exactly talking about the same core as the reload safety analysis core because that analysis will be done separately and probably in more detail.

What we're doing here is putting in a typical core design to determine what kind of margins there will be and where the close spots are.

MEMBER ARMIJO: Well, this core has been -- you're going to put in -- You must have ordered the fuel for the reload coming up in September.

CHAIRMAN SIEBER: Correct.

MEMBER ARMIJO: So it may be even delivered. I don't know. But now is it -- The point I want to ask is is it really identical to the fuel in there?

MR. KAI: Yes.

MEMBER ARMIJO: You know, because cladding, wall thickness, gap.

MR. KAI: Everything.

MEMBER ARMIJO: -- volume, pellet density, all of those things. You're not changing that at all.

MR. KAI: Correct. No fuel design change at all. No physical changes to fuel at all.

MEMBER ARMIJO: And you're increasing the number of burnable rods, the number of burnable assemblies.

MR. KAI: Yes, we are going to increase -- We're going to replace more assemblies. We'll get to that. I'll show you that.

MEMBER ARMIJO: Okay. When we get to that, I'll just hold off on that.

MR. KAI: We will also be adding more the integral fuel burnable poisons to control reactor activity. So really what we have is really a core that is essentially the same as what we have now.

MEMBER ARMIJO: You know, I'm getting that you're not going outside of your experience base --

MR. KAI: Correct.

MEMBER ARMIJO: -- for the existing fuel.

MEMBER BONACA: How many assemblies are you going to replace?

MR. KAI: That's on slide six but it's between 80 and 84. Right now, we do 72 to 76. So we're going to go about eight more assemblies.

MEMBER BANERJEE: So this is going to give you a flatter power distribution.

MR. KAI: Yes and that is actually the next bullet which talks about reducing the real peaking factor. So it will help a little bit.

DR. KRESS: You are increasing the linear heating rate in the fuel. How does that change the thermal gradient between midpoint and the top of the core?

MR. KAI: You mean actually? Because the rating part, yes. You would still be bounded. The average power will go up obviously because it's average. But the actual power distributions are essentially the same. They're not expected to change.

MEMBER BANERJEE: It will be simply more fuel --

MR. KAI: It will simply be more fuel. Correct.

MEMBER ARMIJO: But you do have to increase the peak in your heat generation rate and I couldn't find that number in the materials.

MR. KAI: That's in slide six.

CHAIRMAN SIEBER: The application says you're going to flatten it, too.

(Simultaneous conversations.)

MEMBER ARMIJO: The average linear will go up, that means the peak has to go up and that's what I would like to know. What does the --

MR. KAI: The shape will be the same. But you're right. The actual magnitude of the peak would be higher.

PARTICIPANT: How do you calculate the effect of borate consumption on the clad? How do you calculate the effect of the changed temperature distribution on the boric acid absorption in the top part of the core?

MR. KAI: I don't think there's going to be any change.

PARICIPANT: There must be a reason for you to think that.

MR. KAI: I don't -- I mean I know this is not my area. So I guess we'll take that note and I'll get back to you. But we have -- I was going to have our internal fuel experts go through and do a pretty comprehensive design of the fuel itself to make sure that such things as boric interactions with the fuel would be bounded.

CHAIRMAN SIEBER: If you increase power, you obviously have to -- The initial boron concentration has to change and go up because you're suppressing more.

MR. KAI: Absolutely. That's not correct.

(Simultaneous conversations.)

Boron is actually going to go down.

PARTICIPANT: It seems to me that the temperature is necessarily higher here and it seems to me that the absorption of borate on the zirconium dioxide is a function of temperature and so you're going to have more absorption of borate on the top half of the core. I want to know how they calculated that and he tells me that they did a comprehensive analysis and a bounding. I would like to see the details on that analysis.

MR. KAI: Okay.

PARTICIPANT: Because I had no --

MR. KAI: Now I understand where you're going.

CHAIRMAN SIEBER: At the risk of going out of bounds from your schedule here, I'd like to ask this question. You've made some statements about this core and you've made some statements about the neutron fluence to the vessel walls.

Some of the statements you've made about the core is that you're going to flatten the core. That's statement one. Secondly, you're going to maintain a low leakage core and I'm not sure how you flatten it and do that. Then you're going to do the calculations for neutron fluence to the vessel walls. You have taken account of the seven percent increase in power but you have not taken account of any change in the flux shape. For example, if you flatten the core, the edges will produce more neutrons and therefore you will irradiate the wall more than you would if you just grazed everything by seven percent and my question is by really flattening the core, are you really going to maintain a low leakage core and if so, can you tell me that the neutron fluence to the vessel wall is only affected by the power level and not affected by the power --

MR.: The distribution.

CHAIRMAN SIEBER: -- the flux shape in the core. Do you understand my question?

MR. KAI: Yes. I understand.

CHAIRMAN SIEBER: Okay.

MR. KAI: And actually the next bullet on -- If we go back talking about fluence basically what you said is correct. With an uprate, you would get increased fluence. But one of the advantages that we have was that we had removed a surveillance capsule. Our previous prediction --

CHAIRMAN SIEBER: I forgot that. You have three left.

MR. KAI: Right, and we benchmarked our fluence counts and actually the net result is the calculated fluence action goes down when we get to what we projected.

CHAIRMAN SIEBER: You got some margin out of the third capsule.

MR. KAI: Correct.

CHAIRMAN SIEBER: But that's not my question. Okay.

MR. KAI: Yes.

CHAIRMAN SIEBER: My question is how did you account for the change in flux shape in the fluence calculation for, what, the next 30 years of operation.

MR. KAI: We assumed that we would operate at this equilibrium power distribution that we use as our basis for fluence --

CHAIRMAN SIEBER: You're assuming the power distribution is the same as it is today.

MR. KAI: Well, no.

CHAIRMAN SIEBER: I hope not.

MR. KAI: We use the core --

CHAIRMAN SIEBER: It's not the fluence level. The fluence level is up by seven percent.

MR. KAI: Right.

MEMBER BANERJEE: I guess the answer is have you taken into account in the fluence calculations that the core is going to be flatter.

MEMBER SHACK: You have I presume.

MR. KAI: Yes. Correct. We have. And the fluence calculation is going to be redone.

DR. KRESS: I think Jack's question boils down to is the fluence in the SPU's condition going to, at the vessel wall, going to increase by seven percent or more and I think it should increase by more. It may be small.

CHAIRMAN SIEBER: That's what I thought too. But it's not in the applications. It mentions all these facts, but it doesn't reach a conclusion.

MR. KAI: And that's because on top of the FAC, yes. We didn't do a way to do apples and apples is what you're saying because we also got in the fluence calc the impact of the surveillance calc. So when you put it all together at the end, what you end up is a lower fluence which though if you had done that before would have shown -- I mean, I understand what you're saying, but we didn't do the --

CHAIRMAN SIEBER: Yes, I know you didn't.

DR. WALLIS: But you could say the real fluence is going up, but the calculated fluence isn't because of the way you calculated it.

MR. KAI: Exactly.

DR. WALLIS: The extra information you have.

CHAIRMAN SIEBER: Then they came out and said that we're okay because you got margin out of the third capsule and I don't know how much of that margin you're using up.

MR. KAI: Right. That's correct. I don't have that information.

CHAIRMAN SIEBER: Do you have graphs of the old flux shape versus the new flux shape somewhere?

MR. KAI: Yes, we do.

CHAIRMAN SIEBER: Okay. It would be a good thing for us to look at tomorrow.

MR. KAI: Yes.

CHAIRMAN SIEBER: If you can come up with that. If they are superimposed and we can see what happens to the tail ends and by how much.

MR. KAI: Okay. Well, I don't have it physically now. I will get back to you on that.

CHAIRMAN SIEBER: Okay. I consider that one of the important issues.

MR. KAI: Okay.

CHAIRMAN SIEBER: Okay.

MEMBER BANERJEE: And clearly, you know, it impacts your fourth bullet that reduction in peaking pressure leads to design limit.

MR. KAI: It's a design limit. Keep in mind that this is a design change which means that the actual core obviously is well below the design limit. We just reduced the margin on the design limit in order to gain the other margins. So what our core design is --

MEMBER BANERJEE: I understand the design. It's just that it would be nice to have a really small quantificative idea first.

CHAIRMAN SIEBER: That was one of the problems I had with the application and the SER is that there weren't a lot of numbers. So there wasn't too much for me to work on as far as seeing exactly what it is you're doing.

MEMBER BANERJEE: There's a lot of different --

CHAIRMAN SIEBER: I'm sure there's proprietary information there.

MR. KAI: Yes.

CHAIRMAN SIEBER: Sooner or later we're going to get to at least the answers that we need to have. Okay.

MR. KAI: Okay. The next slide really tries to show what I said before is that we've not changed the fuel design at all. We're saying the same enrichment. The LOCA design parameters are essentially unchanged other than like I said when we decreased the allowable rate of peaking factor.

MEMBER ABDEL-KHALIK: What is your peak P-bar?

I mean, the average bundle power.

MEMBER BANERJEE: It goes up. It has to.

CHAIRMAN SIEBER: Usually you look for the hot rod.

MR. KAI: I mean, I can round up the average kilowatts per foot.

CHAIRMAN SIEBER: No.

MEMBER ABDEL-KHALIK: What is the peak bundle power divided by the average bundle power?

MR. KAI: I don't have that number off the top of my head. I'll have to get that to you.

MR. GUERCI: I'm John Guerci, Manager of Fuel. Our peak P-bars are approximately 1.3.

CHAIRMAN SIEBER: Okay.

MEMBER ABDEL-KHALIK: 1.30.

MR. GUERCI: 1.30, that's correct.

MEMBER ABDEL-KHALIK: But what is your core duty index at the higher power? Core duty. They are increasing t ave. They're increasing power. Your core flow remains unchanged? So could you give us sort of a feel for how this plant at the higher power level would compare to other 4-loop Westinghouse plants in terms of core duty index?

MR. KAI: Okay. With that, I will discuss about temperature later.

MEMBER ABDEL-KHALIK: But it's more than temperature.

MR. KAI: Yes.

MEMBER ABDEL-KHALIK: But do you know what the core duty index is?

MR. GUERCI: Excuse me. Are we referring to the EPRI core duty index?

MEMBER ABDEL-KHALIK: Right.

MR. GUERCI: Yes, I don't have the number right with me. I understand the question. The core duty index could be similar to the Seabrook Power

Station with our power levels and our temperatures.

MEMBER ABDEL-KHALIK: Would your plant be considered a high duty index, a medium or low at the higher power level?

MR. GUERCI: We took it to be a high duty index plant.

MEMBER ABDEL-KHALIK: It would be a good idea to have quantitative answers to these questions.

MR. GUERCI: I understand.

MR. KAI: Understand.

MEMBER ABDEL-KHALIK: Thank you.

MEMBER BANERJEE: Said, why don't you define the core duty index?

MEMBER ABDEL-KHALIK: It's a function of bundle power, flow rate. It's also dependent on t ave, in other words, subcooling. It would give an indication of how much subcooled boiling you have in the upper half of the core and therefore it's an indication of how much boron precipitation you would have in the upper half of the core.

MR. GUERCI: Okay. Right. So yes, it would be a high duty plant. Okay. I understand what you're asking. I don't have the quantitative number.

MEMBER ABDEL-KHALIK: Would you be able to provide that sometime later today?

MR. GUERCI: Yes. We'll get that for you later today.

MEMBER ABDEL-KHALIK: Thank you.

MEMBER BONACA: Do we have an understanding? Does the Licensee have an understanding of the algorithm you're talking about?

MEMBER ABDEL-KHALIK: They should be able to. It's a fairly straightforward definition.

DR. WALLIS: Is this a moderated temperature coefficient?

CHAIRMAN SIEBER: Yes.

DR. WALLIS: Moderated temperature?

CHAIRMAN SIEBER: Yes.

MR. KAI: Okay. The next slide gives a comparison of the nuclear design parameters, the average kilowatts per foot. You can see it decreasing. It also shows that you will be replacing --

MEMBER ABDEL-KHALIK: Can we go back to the previous slide?

MR. KAI: Sure.

MEMBER ABDEL-KHALIK: Your most positive MPC greater than 70 percent power is zero. What is your peak pressure for loss of feedwater ATWS at beginning of cycle?

MR. KAI: I don't know off the top of my head, but we had -- We've done ATWS, loss of feed. She'll look it up for me and we are below the acceptance criteria for -- We're below the generic analysis that had been done for ATWS.

MEMBER ABDEL-KHALIK: Again, it would be nice to have quantitative answers.

MS. ANDRE: I'll find it here.

MR. KAI: Okay. Anything else? I'll continue.

Okay. Any other questions about the fuel?

MEMBER ARMIJO: Yes, I still want to get a number for the increase in peak power, peak linear, heat generation rate. You're increasing your average. But do you increase the peak power on these rods? And what's that number? Is it half a kilowatt a foot or what?

CHAIRMAN SIEBER: How do you want that in percent?

MEMBER ABDEL-KHALIK: Kilowatts per foot increase Delta.

CHAIRMAN SIEBER: Okay.

MR. KAI: So you want the --

CHAIRMAN SIEBER: For the peak assignment.

MR. KAI: Okay.

MEMBER BROWN: Can I ask one question back to the previous slide? You talked about the MP3. I'm not familiar. I'm new from that standpoint. Is that a moderator temperature coefficient?

MR. KAI: Yes, that's correct.

MEMBER BROWN: And it's positive as opposed to negative.

CHAIRMAN SIEBER: Yes.

MR. KAI: Correct.

MEMBER BROWN: Let's talk a little bit about that.

CHAIRMAN SIEBER: This is not a Navy point.

MEMBER BROWN: That's fine. I haven't seen this plant in about 35 years. So if they have a positive temperature coefficient of any kind, it was like a death knell. So that's why I was asking the question.

(Simultaneous conversations.)

CHAIRMAN SIEBER: They plan to launch airplanes.

MEMBER BROWN: I understand that, John. I just wanted to make sure I understood what the thing looked like based on what my reading of their request was.

CHAIRMAN SIEBER: Yes. Well, you're correct.

DR. WALLIS: You have the whole operation of the plant involved is dictated by --

CHAIRMAN SIEBER: Generally, there's an attempt not to have a positive moderator.

MEMBER BROWN: That's kind of a nice idea.

CHAIRMAN SIEBER: Coefficient on the other hand is not --

MEMBER SHACK: It's not what?

CHAIRMAN SIEBER: Forbidden.

MEMBER BROWN: Forbidden. Is that supposed to be because of the boron with the borated coolant that allows you to do that? Like you say I'm not familiar with that enabler.

CHAIRMAN SIEBER: That's sort of what causes it I think. I have no idea what causes expansion to the amount of boron in the core reduced and as you go through life the coefficient changes.

MEMBER BROWN: Okay. I didn't mean to take a tour. I'll ask questions later. Let's go ahead and get this done.

MR. GUERCI: Mr. Chairman, excuse me. John Geirsey. I can clarify. The full power moderated coefficient is a plus or minus six, minus seven, at full power conditions. We have a design limit of zero, but the actual nominals are negative --

MEMBER BROWN: Okay.

MR. GUERCI: At full power.

MEMBER BROWN: That's at full power.

MR. GUERCI: That's correct.

MEMBER BROWN: How does that translate? You said six or seven?

MR. GUERCI: Minus six or minus seven, that's correct.

MEMBER BROWN: Minus six or seven. How does that translate down as opposed to the design basis? Where you say it's plus five, does that mean it's closer to negative as you go below 70 percent?

MR. GUERCI: To sort of clarify, at full power, the limit is zero and we're approximately at minus seven. At zero power, the limit is plus five and we're approximately at zero.

MEMBER BROWN: Okay. Thank you very much. Thank you. Okay.

MR. KAI: Sandy has an answer on the ATWS question.

MS. ANDRE: For the loss of feedwater, it's 2979 psia and the loss of load was 3105 psia.

MEMBER ABDEL-KHALIK: What was the latter number please?

MS. ANDRE: That was for loss of load 3105 psia.

MR. KAI: Okay. We're going to talk about initial conditions. Currently, they are analyzed for a single nominal temperature at 100 percent power with no margin for temperature coastdown and one of the things that we wanted to do when we went to uprate was to allow us to operate over a band of temperature as well as provide margin for a temperature coastdown. So from a design standpoint, we have analyzed a much larger range of initial temperatures from the high end of our temperature band all the way down through a coastdown temperature. So it gives us additional operational flexibility in where we operate and also gives us flexibility at the end of the cycle during coastdown.

We have chosen to operate at the same t ave that we currently operate and that is specifically to limit issues that you brought up in terms of Alloy 600 corrosions.

CHAIRMAN SIEBER: Delta t increases.

MR. KAI: Correct. Delta t.

CHAIRMAN SIEBER: The average stays the same.

MR. KAI: So we have a Delta t increase obviously. So we would be operating at a slightly higher t hot than we currently do.

CHAIRMAN SIEBER: What's that? Four degrees or five degrees, something like that?

MR. KAI: It should be about two degrees higher hot rate temperature. A Delta t is about four degree increase, something in that order. So while we are analyzing, design-wise, of alloys for a full range of temperature, you know, obviously where we actually operate would be lower and we'll go over that on the next slide. We'll talk about the actual temperatures. So if you look at where we are actually going to operate, we would expect a modest impact in terms of Alloy 600 corrosion and really if you look at the cold leg side if you're operating at the same t ave the cold leg temperature actually goes down a little bit.

And the other thing I'd like to mention is that we are changing our pressurizer level initial pressure level, increasing that to take into account --

DR. WALLIS: These numbers are small. You mean, it's something like one percent or something.

MR. KAI: Yes. I have the numbers there.

Okay. This slide is a little busy and I

understand that. One thing I want to point out is other than the second column the other columns are all design numbers. They were calculated -- Temperatures are calculated based on a thermal design flow of 363,000 gpm which is almost ten percent lower than our actual flow rate.

MEMBER ABDEL-KHALIK: So can you tell me the difference between the 372,000 gpm in the first column and the 398,912 in the second column?

MR. KAI: Okay. That 372,000 is called minimum measured flow and it's used as the limit in our DNBR analysis. So this limit is an actual limit that we cannot operate below in terms of RCS flow. So it goes -- This is used in our DNBR analysis as the minimal flow for DNBR. So if you were to measure flow below that we would have to take it out.

MEMBER ABDEL-KHALIK: So are we comparing apples and apples here when we talk about these two numbers?

MR. KAI: Between design flow and --

MEMBER ABDEL-KHALIK: No, between the 372,000 at current design conditions and the 398,912?

MR. KAI: No. The line is what our expected flow is, what we got measured today. That's our best estimate. For DNB purposes, that measurement cannot be below 372,000. So we do a surveillance to assure that we meet that.

CHAIRMAN SIEBER: Yes, that's the current design and that limit changes to 379,200.

MR. KAI: But yes, we are --

CHAIRMAN SIEBER: The flow you think you're going to have is 398.

MEMBER ABDEL-KHALIK: Okay. What actions do you currently take if the measured flow drops below 372,000 gpm?

MR. KAI: That's going to be by our technical specifications. We would be forced to reduce power for shutdown. We cannot operate below that minimum measured flow.

MEMBER ABDEL-KHALIK: So would these actions also be taken if your flow drops below 398,912 at SPU conditions?

MR. KAI: No.

MEMBER ABDEL-KHALIK: So what is the meaning of the 398,000 number?

CHAIRMAN SIEBER: That's the classic flow.

MR. KAI: That's what we currently expect for our flow at uprate conditions as currently measured.

MEMBER SHACK: The feel good number.

(Simultaneous conversations.)

MEMBER ARMIJO: It shows you how much margin there is between the actual flow and what their design minimum flow rate is.

CHAIRMAN SIEBER: Yes.

MEMBER ARMIJO: And it goes up to -- The required goes up to 379,200 under the SPU conditions.

(Simultaneous conversations.)

MEMBER ABDEL-KHALIK: So does this reflect your current minimum measured flow?

MR. KAI: The number on the left.

MEMBER ABDEL-KHALIK: The 398,912, does that reflect your current flow rate measurement?

MR. KAI: Yes.

DR. KRESS: But it really isn't a minimum flow rate. This is your current flow rate.

MR. KAI: Not minimum. That's why it says best estimate. I didn't know how to put it in the table.

MEMBER ARMIJO: Okay. For the SPU, you're going to change 372,000 to 379,200 for the minimum allowable flow rate to continue normal operations.

MR. KAI: Right.

MEMBER ARMIJO: Let's try to understand

this table fully. And 398,000, just as an expression of that's where you are today.

MR. KAI: Right.

MEMBER ARMIJO: And it's enough to say, "Look, we have a lot of margin in there and then we can do this" and I would reiterate this as kind of a feel good number and have yourself -- From what I understand what you said earlier, the second column SPU, that's basically what you would expect your normal operating parameters to be. The others are all design limit, design minimums.

CHAIRMAN SIEBER: Yes.

MEMBER ARMIJO: Your second column is basically at SPU where you expect the normal operation to be.

MR. KAI: Exactly. Correct.

MEMBER BANERJEE: What is your normal operation now?

MR. KAI: Our current t ave is 571, the t hot is about two degrees lower and the t cold is --

MEMBER BANERJEE: But what's your flow rate, gpm?

MR. KAI: It's the same.

MEMBER BANERJEE: That's going to stay the same.

CHAIRMAN SIEBER: Correct.

MEMBER ABDEL-KHALIK: If you're operating at tech spec limit flow of 379,200 at the stretch power uprate, what would your t hot be?

MR. KAI: It would be about two degrees higher than what's shown here. I think that's about --

MEMBER BROWN: Say that again.

MR. KAI: It's about a five percent.

MEMBER ABDEL-KHALIK: I mean, by tech spec you're allowed to operate with a flow down to 379,200. Right? So the question is is there a column here that reflects that condition and I can't see it.

MEMBER BROWN: Wouldn't it be 622.6?

MEMBER ABDEL-KHALIK: No.

MR. KAI: No, that 622.6 is based on --

MEMBER ABDEL-KHALIK: Height, yes.

MR. KAI: -- the 363,000. So you would have to --

MEMBER: Based on what?

MR. KAI: 363,000. It's design flow. These three columns are all design flow.

DR. WALLIS: But that's based on that.

MEMBER BANERJEE: I'm a little confused here. So let's look at the design column and the SPU column.

(Simultaneous conversations.)

MEMBER ARMIJO: That's at design level also. That's not a normal condition.

MEMBER BANERJEE: Yes, I'm sort of confused because what should I look at as the cold leg temperature and the hot leg temperature and compare the current design with the new design? Which column should I look at?

MR. KAI: Okay.

MEMBER BANERJEE: So the first column is your current design. That's right?

MR. KAI: Yes.

MEMBER BANERJEE: So your cold leg temperature is 555.6 and your hot leg temperature is 618.3. Now which column is comparable to that column with the new design?

MR. KAI: Okay. Now we have --

MEMBER BANERJEE: So apples and apples.

MR. KAI: An apples and apples comparison would be that the last three columns which gives you the SPU max, the Max t ave, the Min t ave and coastdown. Those were all calculated with the same flow rate. That's the 363,200 gpm.

CHAIRMAN SIEBER: Hopefully you did the

average of columns three and four. If you're looking at the extreme boundaries of normal operations --

MEMBER ARMIJO: Fine. I don't think that there is an apples to apples comparison here because to do that you would have to have three columns for the current design to line up with the --

CHAIRMAN SIEBER: Right.

MEMBER ARMIJO: -- Last three columns. There's really not an apples to apples comparison.

MEMBER BANERJEE: Yes, there is none. Exactly. But at least it gives you a feel for it.

MR. KAI: And that's what I was trying to do here without making three or four tables. On this table, the column in the middle is where we expect to be and I think what we want and what I use that for is in terms of things like our aging management which is what Alloy 600 is. I want to see what do we really expect in terms of increases, in terms of Alloy 600.

MEMBER ABDEL-KHALIK: What I was really looking for is a column that would say what the flow rate is, 379,200 and t ave is 587.1. It gives you a nominal t ave and if that is the case, then I suspect your peak temperature would be 620.2. That would be your t hot.

MR. KAI: Yes, I think that's right. I

was just doing the math in my head, but correct.

CHAIRMAN SIEBER: Usually when you make these comparisons, you make them --

(Simultaneous conversations.)

MEMBER BANERJEE: If you had two tables you would have had much less discussion.

MR. KAI: Right.

MEMBER ABDEL-KHALIK: I guess my concern is that why show this second column rather than a column where the flow rate is at the tech spec limit of 379,200 and the t ave value is at a nominal value of 587.1, which would tell you that that nominal hot leg temperature at tech spec flow limit would be 620.1 rather than 617.1.

MR. KAI: Okay. I didn't choose to do it that way, but I see your point. Like I said, I was really trying to compare where we expect to operate compared to the design condition. So it wasn't meant to look at --

MEMBER BANERJEE: But let's make it simple. Why don't we get a table with apples and apples comparison in the process? That would be easier rather than trying to do these sums ourselves. At some point, I'd like --

MEMBER BROWN: I appreciate your attempt.

But it does make it easier. There's nothing on here that you can really compare current to the SPU conditions.

MEMBER ARMIJO: Under the same set of assumptions.

MEMBER BROWN: Under the same set of assumptions.

MEMBER POWERS: But on the other hand, there's nothing here that looks very dramatic either.

MEMBER BROWN: I agree.

MEMBER POWERS: And I can interpolate fairly quickly here the fluid that there's not of a change.

MEMBER BROWN: I don't disagree with that at all.

CHAIRMAN SIEBER: You can do it in your head.

MR. KAI: These numbers are all there like I said. I think I'll ask for Goshing Wong who is our Westinghouse thermal hydraulic engineer. He can provide additional information about the temperature.

CHAIRMAN SIEBER: Probably if you just did one. If we need to answer this at all, you just need one column which is SPU design.

MR. WONG: Okay. I would like to add real quickly some information. This table --

MEMBER ARMIJO: Identify yourself.

MR. WONG: Yes. My name is Goshing Wong. I'm a thermal hydraulic designer from Westinghouse. Good morning, everyone. This table, the last line, the last row, is additional information. Everything else are consistent with each other. Basically, the t ave, t hot and the thermal design flow are consistent. So the last row is just additional information. That's it.

MR. KAI: But what you want is this one at 587.1 with a flow rate of 379.

CHAIRMAN SIEBER: Yes.

DR. WALLIS: I just have one comment. If you can quote the normal operation flow rate to one part in 400,000 you ought to be able to capture 1.7 percent uncertainty.

(Laughter.)

MEMBER BANERJEE: How do you measure the flow rate?

MR. KAI: Pardon?

MEMBER BANERJEE: How do you measure the flow rate?

CHAIRMAN SIEBER: Thermal calorimetric.

MR. KAI: Thermal calorimetric.

MEMBER BANERJEE: That's calorimetric?

MR. KAI: Right.

CHAIRMAN SIEBER: Yes, flow meters --

MEMBER ARMIJO: Obviously you do have flow meters but then you allow them a calorimetric --

MR. KAI: Correct.

DR. WALLIS: But your accuracy is one percent or something. So this is imaginary. The last three figures are completely imaginary.

MR. KAI: We mention the errors, and we put the number up to places, but you're right. Correct.

CHAIRMAN SIEBER: Why don't we move on?

MR. HUEGEL: This is Dave Huegel from Westinghouse and I guess one question that everybody seems to be wanting an answer to is the fact that you have two different t aves yet you have two different flow rates and the question is how does that affect the t hot. If you just merely take the difference in t ave and reduce that amount from t hot and t cold, believe me, that to the nearest tenth of degree will give you what the temperature is for t hot and t cold. It's as simple as that.

DR. WALLIS: So it is as simple as that.

MR. HUEGEL: If you want to know what the t hot and t cold for the flow rate or what it would be at the 587, just take the difference in t ave which is roughly 2.5 degrees, subtract that from the 622.6 which gives you 620.1 for t hot and that's --

MEMBER ARMIJO: I understand it and the information is here.

MR. HUEGEL: Does that answer your question?

MEMBER ARMIJO: It would just make it a little easier if it just showed up that you can -- We look for things to compare when we see a table.

MR. HUEGEL: What we're trying to present here is here's what the current operating conditions are, here are the conditions that we're going to. Does that answer the question?

MEMBER ARMIJO: Yes sir.

CHAIRMAN SIEBER: Good enough.

MR. HUEGEL: Okay.

CHAIRMAN SIEBER: Why don't we move to Slide 9?

MR. KAI: Okay.

MEMBER BANERJEE: The main effect is that you now have a higher minimum flow for your DNB limit.

CHAIRMAN SIEBER: Right.

MEMBER BANERJEE: That's the main effect, I guess.

MR. KAI: Yes. Correct.

CHAIRMAN SIEBER: But they have margin.

MEMBER BANERJEE: And a little bit hotter outlet --

MR. KAI: Correct. And like I said, in terms of that, the analysis goes back to why I put this number, the best estimate number in this, as opposed to something to compare apples to apples which I understand now.

MEMBER BANERJEE: Referring to us that you have enough margins.

MR. KAI: Yes, that's what I'm trying to do. Correct.

CHAIRMAN SIEBER: No matter what they have margin. Okay. Slide 9.

MR. KAI: Pressurizer level, we are increasing from the 100 percent pressurizer level from 61.5 percent to 64. This represents a compromise, obviously, for accidental system. Pressurizer overfill is an issue. The higher you make it, the less margin you have to overfill and we have to weigh that versus looking at a routine reactor trip and assuring that the level collapse from the reactor trip doesn't result in things like uncovering the heaters or isolating letdown.

CHAIRMAN SIEBER: So what level would you expect to drop to during a trip from 100 percent power, reactor trip, assuming everything is tight and it actually trips instead of just drops off?

MR. KAI: Approximately the 27th or the 28th percent. We have about five percent margin.

CHAIRMAN SIEBER: Okay.

MR. KAI: That's what we try to maintain.

CHAIRMAN SIEBER: That's where the 550 is here.

MR. KAI: About.

CHAIRMAN SIEBER: And where is the range of the heaters? Where are they on this level?

MR. KAI: The set point is 22. So that's --

CHAIRMAN SIEBER: Where physically are the heaters?

MR. KAI: Physically they are below 22 and --

MS. ANDRE: Seventeen maybe.

MR. KAI: I think it's 17 or 18 percent, it's about.

CHAIRMAN SIEBER: Okay.

MR. KAI: So we are trying to maintain

under the five percent margin to the level cutoff and the heater cutoff and the letdown isolation and that currently provides --

CHAIRMAN SIEBER: But you have to run the heaters. They would deenergize.

MR. KAI: Correct.

CHAIRMAN SIEBER: On the other hand, an operator, one of the sickening feelings is to have pressurizer level disappear.

(Laughter.)

MR. KAI: Right.

CHAIRMAN SIEBER: Okay.

MR. KAI: Okay. That's all I was going to say about initial conditions. If you have any other questions?

CHAIRMAN SIEBER: Is this a good time to take a break or do you want to do Slide 10?

MR. KAI: Probably yes because I'm going to go and talk about the safety analysis next.

CHAIRMAN SIEBER: Why don't we do that and since we're ahead of schedule we can come back at 20 to 11:00. So we'll take our recess of 20 minutes now.

(Whereupon, at 10:20 a.m., the above-entitled matter recessed and reconvened at 10:38 a.m.)

CHAIRMAN SIEBER: I'd like to resume the

meeting after our short recess and what I'd like to do is call on -- One of the questions we asked had to do with the calculation of fluence to the vessel walls, how that is affected by the power increase and how it's affected by the power increase and how it's affected by the change in power shape and Ambrose Wallace of NRR can answer that question, I think.

MR. WALLACE: Yes. Thank you. My name is Ambrose Wallace. I'm with the Reactor Systems at NRR and responsible for the fluence calculations. The methodology we use and all of the licensees that have adapted so far which we have codes approved in that direction is the so-called synthesis method and what it takes into account are the following things.

One is the loadings of the outer assemblies. The second is the azimuthal radial distribution and the third is axial radial distribution. So all of these things are synthesized in a manner that accounts for the azimuthal, the radial and the axial distributions in addition to which it accounts for the average cycle loading our power condition, of each one of the assemblies in the outer two assembly rows.

So the question is as I understood it to be is the flattening of the flux taken into account.

Absolutely.

CHAIRMAN SIEBER: Okay. I guess that sort of satisfies my question. Does anybody else have any questions they would like to ask about that?

DR. KRESS: I'm still a little confused. The reason that the most -- I'm guessing, but the reason the most recent capsule fluence was lower than the previous, let's say, extrapolation was partly because of this low leakage core design. Imagine that to create that. Now with a higher power being generated by the peripheral bundles the benefits of that low leakage core are reduced.

MR. WALLACE: Yes.

DR. KRESS: So the expectation of the fluence is going to be, should be, higher and I'm just wondering if it actually was taken into account in the final calculation.

MR. WALLACE: Let me address that question. The capsules, the surveillance capsules, have really very little to do with the calculated value. The surveillance capsules are there to connect the embrittlement of the archival material that bleeds into the capsule to its exposure. That may or may not have anything to do with the maximum location of the exposure of the vessel. That is a calculated value which we ascertained, verified, if you wish, or benchmarked to the particular value of where the capsule is. But the capsule may or may not have anything to do with the calculated maximum value on which the material properties of the vessel are based.

DR. KRESS: So it's not even a comparison between the calculated and the actual measured.

MR. WALLACE: Yes, it is for this particular location where the capsule is.

DR. KRESS: I understand that.

MR. WALLACE: Yes, it is to verify the reliability of the calculation rather than to directly measure the value of the vessel. There is no way to directly measure the value of the vessel.

DR. KRESS: Okay. That explains it.

MR. WALLACE: Thank you.

CHAIRMAN SIEBER: Now Millstone has six capsules. If I read the report correctly, you may have removed three. Typically, the schedule for removal of capsules and their examination is one every ten years. So this subject gets revisited two more times in the life of the plant and each time these same calculations are made to predict what changes in vessel properties have occurred over that period of time.

MEMBER BROWN: We've talked around this a lot and I think I heard the answer. I just want to make sure that the fluence calculations for the future under SPU conditions has taken into account the flatter core --

CHAIRMAN SIEBER: Yes.

(Simultaneous conversations.)

Okay. I think we're on Slide 10 if you want to continue.

MR. KAI: Okay. We'll talk about the safety analysis now.

MR. GUERCI: I want to make a statement to the previous question. This is John Guerci. In response to the EPRI duty index, our best estimate numbers went from approximately from the mid to low 160s, 163, up to a little over 200, maybe 206. The worst case numbers are a little higher if you use minimum flow instead of nominal flow. I think we used that as a scoping tool then to get back to, I believe, the question previously with -- In fact, there's a clarification on the boron precipitation.

I assume you're referring to the corrosion of boron at the top of the core. And so we used the EPRI duty index as a scoping tool and we had looked at this a number of years ago prior to even the uprates starting and then we go through a bow analysis to look at boron deposition in the core at the new uprated conditions. So that work has all been done in support of --

MEMBER ABDEL-KHALIK: Where does the 206 place you?

MR. GUERCI: My recollection is high. I think it's above 200, approximately whatever we call the high range. Again, it's a scoping tool because we need to go back and look at the actual thermal hydraulic conditions. It's a pretty general kind of tool in terms of it doesn't have the detailed thermal hydraulics in it.

MEMBER ABDEL-KHALIK: Thank you.

MR. GUERCI: And the peak linear heat rate went from 14.2 at a 15.1 from the previous cycle to the projected uprate cycle.

MEMBER BROWN: Fourteen?

MR. GUERCI: 14.2 kilowatts per foot up to 15.1 kilowatts per foot.

MEMBER BROWN: Okay. Thank you.

CHAIRMAN SIEBER: Any others before we begin?

Go ahead.

5. SAFETY ANALYSIS

MR. KAI: Okay. I'll make some general remarks about the safety analysis first and then like I said, the slides have lots of results. I'm not going to go over them all.

The first thing I would like to state is what we did in the safety analysis is essentially redo every one of the accident analyses. Not only that, but we went back and validated every input that went into the safety analysis, not just the ones that are affected by uprate but every single parameter, flows, ECCS, performance, all of the parameters, were revalidated for uprate.

And in general, we did not need methodology changes to show acceptable results. We did, however, we have used the latest technology and methodology that Westinghouse employs primarily to go for going forward that we would be postured for the next ten years or so in terms of analysis methods.

So the scoping studies were done with the current tools and so that we accepted the margin to the uprate. In general, we have not used analysis methods to gain margins to show acceptable results. Now it does provide us some benefit of how it runs well. We have gained some benefit in terms of a new methodology. But in general, we have not used analysis as a method to gain margin.

What we wanted to do is restore margin or where we could back to our current. So we've made a number of modifications to restore DNBR margin as opposed to just using the available margin that we have.

CHAIRMAN SIEBER: Now does this discount the fact that you've used a realistic code for your Appendix K analysis.

MR. KAI: Right.

CHAIRMAN SIEBER: Rather than the deterministic one because that gives you oodles of margin that you don't get out of the deterministic method.

MR. KAI: Right, and we currently have fairly significant margins in LOCA here. We're not LOCA limited.

CHAIRMAN SIEBER: From the FAC's temperature limit.

MR. KAI: Correct. From the -- Correct.

CHAIRMAN SIEBER: And oxidation. Right?

MR. KAI: Yes. So using the current methodology, BART/BASH, we have, we already have significant margin. We didn't use ASTRUM there. We just used it as going forward and you'll see that when we compare. There are comparisons and results.

CHAIRMAN SIEBER: Let me ask you this question. Are the names that you used proprietary in any sense?

MR. KAI: No.

CHAIRMAN SIEBER: Okay.

MEMBER BANERJEE: Are these -- You'll come to it later, but I assume that ASTRUM must be a approved code of some sort.

MR. KAI: Yes, it's COBRA/TRAC.

MEMBER BANERJEE: Oh, it's COBRA/TRAC. All right.

CHAIRMAN SIEBER: There is a topic on that, I think.

MEMBER BROWN: Clarification. I think I understood you to say that you didn't need the new analysis to show acceptability of the power uprate.

MR. KAI: New codes.

MEMBER BROWN: Okay. You also tossed margins into there and I believe that if you're using your original codes and methods -- Are you saying you would have, with the modifications made you, would maintain the same margin.

MR. KAI: I believe so. Again, we did this early in the project to look at one of the ways that we could gain back margin and we did a number of sensitivity studies to show that we won't end up with zero margin coming out of this and that we have comparable margins.

MEMBER BROWN: And I understand there are a number of things that you're doing that do help you on your margins and stuff but it wasn't clear to me that under your original codes that you would have maintained exactly the same margins there.

MR. KAI: Right, and again you're right. That's not really -- I don't think you could have drawn that from our documentation. I think you're right.

MEMBER BANERJEE: You didn't do any calculations with BART/BASH or whatever it's called.

MR. KAI: At the uprate, no. We did not.

MEMBER BANERJEE: So what was the margin with -- What was the PCT margin with BART/BASH for the current design?

MR. KAI: Well, I will get to that. If you want to look at the numbers, it's on Slide 24 and you'll see that the current numbers are really low, with the current methodology.

MEMBER BANERJEE: With the current methodology, it was fairly close to 1974. With a seven percent uprate, I don't know what it would have done but it would reduce that quite a bit I would think.

CHAIRMAN SIEBER: And that's the result of using a realistic code. Right?

MEMBER BANERJEE: You're using a realistic code, but you're not using the sort of best estimate plus uncertainties, are you?

MR. KAI: This is just sort of a --

MEMBER BANERJEE: Explain to me what this code ASTRUM does. Does it have Appendix K assumptions in there or the best estimate only and if so, do we have uncertainty bands on it?

MR. KAI: Could I delay that until I get there. I will discuss that.

MEMBER BANERJEE: Okay.

MR. KAI: Okay.

MEMBER BANERJEE: And can you also explain all these other numbers in more detail?

MR. KAI: Well, I wasn't going to go over them all, but I will talk about LOCA.

MEMBER ARMIJO: Since you're talking about the codes, the new codes, that you did use, you made a comment in your documentation that you benchmarked the new codes against something. But you didn't --

You just said you benchmarked them. But was I supposed to walk away with something from that? I mean, do you validate them against some other data? Do you go back and take them or am I misinterpreting what you said in the written license request? Do you understand my question?

MR. KAI: Yes. In general, the new codes are based on a sense of proved methodology which contains a series of benchmark analyses which can demonstrate that they will produce conservative results. We followed the modeling guidelines. We have -- We did look at, for example, a full coastdown to make sure that it compares with the current analysis of record. So in general, we will rely on the current benchmarking that's done by Westinghouse for the approved methodology.

MEMBER BROWN: So you didn't literally take this code and say, "Here we did this analysis on this particular transient with our old method." You used the new method and see that you got either the same or a more conservative or less conservative result. You didn't do that as part of this. Is that correct?

MR. KAI: That's correct.

MEMBER BROWN: So benchmarking was done by

Westinghouse and whatever other process they went through to get that code to get the Betty Crocker/Good Housekeeping seal of approval or whatever.

MR. KAI: Right.

MEMBER BROWN: Okay.

MR. KAI: And then, of course, obviously we did -- If we saw something that was significantly different -- But no we did not do any plant specific benchmarking.

CHAIRMAN SIEBER: I'd like to get back to Slide 10 please. You use an alternate source term for your radiological analysis. Right?

MR. KAI: Correct.

CHAIRMAN SIEBER: That was applied for and approved prior to your application for stretch power uprate.

MR. KAI: Right. Now when I get there, I will talk more. Go ahead. Sorry.

CHAIRMAN SIEBER: And the last bullet says, "PRA results show SPU has minimal impact on risk." Could you give me the numbers for the old risk and the new risk?

MR. KAI: They are on Slide 33. I will get to that. This is kind of a summary. Slide 33, it has the new and old.

CHAIRMAN SIEBER: Okay. When you get there, my follow-up question is usually in PWRs as dominated by human factors issues which was operator response times and the only one I was able to pick out was the switch from injection to recirculation which is hours out in the sequence. So if you go through the human factors issues, that will help me.

MR. KAI: Yes.

CHAIRMAN SIEBER: You'll follow up.

MR. KAI: Yes. Don't worry.

CHAIRMAN SIEBER: Okay.

DR. KRESS: Are there any plans to do a stretch power uprate for the Millstone 2?

MR. KAI: Yes.

DR. KRESS: After this one?

MR. KAI: Yes. Of course, that's a totally different animal altogether.

To follow up on your -- One point I would like to make about the PRA model is that what we did is that we did validate using -- We did analyses to confirm that the success criteria and the operating response times that were assumed in the current analysis, for the PRA, we showed that at SPU conditions those still remained valid. In other words, we really updated all the thermal hydraulic analysis including the determination about response times and validated that what they had assumed is still valid.

CHAIRMAN SIEBER: When we get there, you can give me the times and the operator actions.

MR. KAI: Okay.

MEMBER ARMIJO: And you're going to tell us what these margins you talked about, those four items, you're going to say what they were for the current. We'll be able to get the Delta.

MR. KAI: Yes. Correct.

DR. WALLIS: We'll get to all this stuff.

MR. KAI: Any other questions before going on?

CHAIRMAN SIEBER: Go on.

MR. KAI: Number 11, what I'm doing is summarizing the methods change and as you can see that for the non-LOCA we have switched from LOFTRAN I think to RETRAN VIPRE and we have you used ASTRUM for large break LOCA.

CHAIRMAN SIEBER: And ASTRUM is --

MEMBER BANERJEE: What is the status of RETRAN? I thought we had some concerns at the ACRS about RETRAN that were raised before.

CHAIRMAN SIEBER: I think it's an improved code.

DR. WALLIS: Yes, we had some concerns and then I discovered recently that the NRC approved it anyway.

(Laughter.)

CHAIRMAN SIEBER: It's an approved code. I did check that. And ASTRUM is just an update of BART/BASH. Right?

MR. KAI: No. Okay.

CHAIRMAN SIEBER: No?

MEMBER BANERJEE: Completely different.

MR. KAI: It's completely different. Correct. It actually uses COBRA/TRAC and it is a best estimate methodology and it's approved.

MEMBER BANERJEE: And you'll explain to us how you've used it because it's approved with uncertainties calculations. Right? The code is approved, but it's typically used as a best estimate plus uncertainties.

MR. KAI: Correct. And we applied the approved ASTRUM methodology that was developed. That was approved by the NRC.

DR. WALLIS: The number you quote is a 9595 number, isn't it?

MR. KAI: It's an upper bound estimate of

9595. Correct.

MR. HARTZ: This is Josh Hartz. I work for Westinghouse Electric in a LOCA group. That is a 95^{th} percentile PCT.

DR. WALLIS: Ninety-fifth percent.

MR. HARTZ: Ninety-five ninety-five, yes, exactly. And it's advanced statistical treatment of uncertainty method. That's what ASTRUM stands for.

CHAIRMAN SIEBER: Right.

MEMBER BANERJEE: You used 59 sample drums or something. What is it?

CHAIRMAN SIEBER: That's the magic number.

MR. HARTZ: It has been increased and I believe it's gone to 124 sample cases. Don't quote me on that.

MEMBER BANERJEE: We may see them go to 500 though.

MR. HARTZ: Yes, and you may see that.

CHAIRMAN SIEBER: You ought to reduce it up to a certain limit. Okay. Go ahead.

MR. KAI: Okay. There is 124.

CHAIRMAN SIEBER: Okay.

MR. KAI: One hundred twenty-four cases.

Okay. I'm going to go over DNBR margin. The way that this is organized is based on safety limits. Start with DNBR and I'll talk about pressure and then overfill.

Obviously, we were concerned with DNBR at the uprate. Raising power is clearly a negative impact on the margin.

MEMBER SHACK: Just one. You changed your containment code, too. Right?

MR. KAI: Correct. We will talk about that in the afternoon, but yes. And there's a separate set of slides addressing containment.

One thing to understand about where we currently operate is that in this cycle and in the past we have been subject to what's called upper plenum anomaly and that results in spiking in the hot leg temperature. We've had spurious OTDT, OPDT alarms, pre trip alarms, and so that's been an issue with Millstone for a number of years. What we did in the last cycle really in order to try and reduce the likelihood of getting these alarms is that we have essentially taken all of our DNBR margin and used it to address this problem. So you'll see that in the next slide.

So obviously going forward, we could not live with that situation and one of the mods we have done to address that is to put in the modification that will reduce the severity of the --

MEMBER ABDEL-KHALIK: Are the spikes in hot leg temperatures consistently in a specific loop?

MR. KAI: No, they go from loop to loop.

MEMBER ABDEL-KHALIK: Then it's random.

MR. KAI: What happens is if you look at the core, the fluid coming out of the core from the different assemblies doesn't completely mix and depending on a somewhat random process as to whether it goes into which hot leg and also can rotate and whether the RTDCs, the temperature or not, you can get spikes in the temperature.

CHAIRMAN SIEBER: I think they call that chugging.

MR. KAI: Okay. We did do -- That was clearly something that we had to address and we needed to reestablish DNBR margin.

MEMBER POWERS: Is the spiking frequently enough that we have peak problem and there is damaged peak?

MR. KAI: Well, it's definitely a -- The big thing especially if you get a pre trip alarm.

MEMBER ABDEL-KHALIK: What is particularly the order of magnitude of these spikes?

CHAIRMAN SIEBER: They can digress.

MR. KAI: Darn. I have this in separate and not in temperature. Do you know what the order of --

MS. ANDRE: I don't.

CHAIRMAN SIEBER: Five degrees or 50 degrees?

MR. KAI: It's more five. Fifty degrees you would --

MEMBER ABDEL-KHALIK: Could you please find out and let us know later?

MR. KAI: Okay.

MEMBER BANERJEE: Do you understand why they occur?

MR. KAI: Yes. Like I said, it's because the fluid coming out of the core, of the different assemblies, does not completely mix when it goes into the hot leg. Now like I said, there's some randomness as to which outlet fuel assembly goes into which loop and whether the RTD actually sees the hot temperature and mixed temperature.

MEMBER BANERJEE: So these spikes would be what you'd expect to come out of the hotter channels. Is that it or is it larger?

MR. KAI: The hotter channel has the higher temperature. Right?

MEMBER BANERJEE: Right. But are these --

MR. KAI: The RTD is sitting in a well on the pipe. So depending on whether the fluid entering the pipe sees the water from the --

MEMBER BANERJEE: Let's say it's not mixed. Let's take an extreme case and it goes through. Now are these temperature spikes consistent with the maximum temperature you would expect of the hottest channels?

MR. KAI: Yes.

MEMBER BANERJEE: It's bounded by that.

MR. KAI: Yes.

MEMBER BANERJEE: The size of the spike is maybe five degrees or something like that.

MR. KAI: Right.

CHAIRMAN SIEBER: And you have t hot trips.

MEMBER BANERJEE: Can you reassure us that it's not something else?

MR. KAI: Correct.

MEMBER BANERJEE: It's just a lack of mixing.

MR. KAI: Yes, and like I said, a certain amount of randomness. For instance, if it actually was the same, the RTD would see the same temperature all the time. But it doesn't. It rotates a little bit.

MEMBER BANERJEE: We understand.

MEMBER BROWN: Does it always rotate in the same direction and always occurs at the same time or it just bounces around?

MR. KAI: It's bouncing around. Correct.

DR. WALLIS: So this affects the hot leg temperature.

MR. KAI: It measures hot leg temperature.

DR. WALLIS: Yes, and you have limits on that. But what does it -- It doesn't change DNBR, does it?

MR. KAI: No, physically it doesn't. But it's there.

DR. WALLIS: Why do you couple them together?

MR. KAI: In order to not get the pre trip alarms, what I did is I raised the set point.

DR. WALLIS: Oh, you raised the set point. It doesn't really change the DNBR, does it?

MR. KAI: No, but using the set point does.

MEMBER BROWN: The alarm set point or the trip set point?

MR. KAI: The trip set point.

MEMBER BROWN: When you say alarm, that's synonymous with trip.

MR. KAI: Yes.

CHAIRMAN SIEBER: Well, you get the alarm before the trip.

(Simultaneous conversations.)

We have a t hot trip. So any kind of a cycling --

MEMBER BONACA: I was asking a question, this is not unique to Millstone 3.

MR. KAI: No.

MEMBER BONACA: I'm sure you have looked at sister plants.

MR. KAI: Correct. And the more that we're implementing these, what's implemented at the sister plants to reduce the severity of the spiking.

MEMBER BONACA: Does the phenomenon exist if everybody uprates?

MR. KAI: It's somewhat random. When we did our initial studies, we just assumed it would and looked and obviously now we're predicting when we did the initial DNBR sensitivity studies, it showed that we would more frequently get pre trip alarms and actually we just arbitrarily increased it so that you could possibly get a channel to be in trip mode and for the channels to go into trip. So that would be as you said, but it would be even worse for the operator.

MEMBER BANERJEE: The spikes are -- Their magnitude is such that they're still indicative that the flow distribution to the channels is what you expect it to be.

MR. KAI: Correct.

MEMBER BANERJEE: So it's not larger. It doesn't show that some channels might periodically be getting lower flows or anything like that.

MR. KAI: No. Correct.

MEMBER BANERJEE: That's something the magnitude -- It would be interesting to know what the magnitude is and to compare them with the hottest channel of the temperatures assuming a flow distribution.

MR. KAI: Okay.

MEMBER BANERJEE: That would be a concern.

CHAIRMAN SIEBER: Is that a question you folks will find out for us?

MR. KAI: Yes.

CHAIRMAN SIEBER: Okay.

MR. KAI: Okay. Any questions?

CHAIRMAN SIEBER: Let's go to 13.

MEMBER ABDEL-KHALIK: What's the difference between WRB2M and WRB2?

MR. KAI: A slightly different correlation based on additional tests that were done to develop the correlation. So it's a slightly improved correlation that's for the RFA fuel product line.

MEMBER BANERJEE: It's been approved.

MR. KAI: It's been approved. Correct.

CHAIRMAN SIEBER: Yes.

MEMBER BANERJEE: Where were these tests done? Were these full scale bundle tests?

MR. KAI: I believe so. Let me get Westinghouse to actually talk about the tests.

MR. WONG: This is Mr. Goshing Wong again from Westinghouse. Yes, we did the DNB test in Columbia University. So the WRB2M correlation is based on those test data.

MEMBER BANERJEE: This is very old data then. Right?

MR. WONG: Yes.

MEMBER BANERJEE: So when were these tests done?

MR. WONG: Nineteen something. I forget the exact year, but I can check it out.

DR. KRESS: It's been around a long time.

MEMBER BANERJEE: So this is pretty old data. This is not new data.

MR. WONG: Yes. It's not really new data.

CHAIRMAN SIEBER: Right. Okay.

MR. WONG: Another question?

MEMBER BROWN: I presume this is you're passing the electronic filter on the hot leg. That's part of the changes you made.

MR. KAI: Yes.

MEMBER BROWN: Obviously that introduces a time response into the t h?

MR. KAI: Yes.

MEMBER BROWN: For your Delta-t function. I presume that was cranked into you other analysis.

MR. KAI: Yes.

MEMBER BROWN: You said four seconds which is fairly hefty for the most part. Do you know that this eliminates the spikes?

MR. KAI: Yes. We've done a study looking at that.

MEMBER BROWN: So you include it in as a trial to see if it --

MR. WONG: Yes.

DR. WALLIS: So the spikes have a frequency or a time of duration of four seconds or something?

MR. KAI: No, the actual time --

MEMBER BROWN: It's supposed to be slowed down for four seconds.

MR. KAI: The time constant slows down the response. I mean, if you're in the frequency of the spikes, the time constant, that's what you're asking. Right? The frequency, not the duration.

DR. WALLIS: And you're going to get a spike and its length is something like four seconds. Is that what you're doing?

MEMBER ARMIJO: No, it's less than that.

MR. KAI: It's less than that.

DR. WALLIS: Less than that?

CHAIRMAN SIEBER: Yes.

MR. KAI: And so that the filter will assure that you don't really --

DR. WALLIS: How often does this spike happen? Is it something that's very regular? I mean, it happens -- If it's very regular, I think you would have fatigue concerns.

CHAIRMAN SIEBER: Somewhere. Well, if the temperature is very small.

DR. WALLIS: Very small.

(Simultaneous conversations.)

MEMBER BROWN: It temps together. You get a little higher.

DR. WALLIS: How big is very small?

DR. KRESS: Well, they will find out.

(Simultaneous conversations.)

MEMBER BROWN: That's why I asked whether it's five or 50 and he said roughly or whatever. We would like to see that that's confirmed.

MR. KAI: Okay. And like I said, I have this in set point and not in degrees unfortunately. When we benchmarked this, typically the spikes are of very short duration, a couple of seconds, and the frequency is on the order of about -- And it varies. It could be as much as 30 seconds apart or --

MEMBER BROWN: So you have to come in like that.

DR. WALLIS: What's the amplitude?

MR. KAI: Unfortunately, what I have is in terms of the set point, not in terms of degrees and that's --

DR. WALLIS: So what do you call a t hot if it's oscillating like this? I mean, when you say t hot max in your table do you mean the maximum of the spike or the maximum of the average?

MR. KAI: The maximum of the average.

DR. WALLIS: So what's a fair measure of t hot if you have these spikes in there in terms of its effect on materials and so on?

MR. KAI: Well, I can tell you that we used the average temperature for the impact on materials. I mean, it does again --

DR. WALLIS: Doesn't the effect on materials go up rather rapidly with temperature?

MEMBER BANERJEE: It should be exponential.

CHAIRMAN SIEBER: Or you just average.

DR. WALLIS: You think that's okay?

MR. KAI: Not bad.

DR. WALLIS: Just fine. It will get washed out in the transit of the --

CHAIRMAN SIEBER: It's random in the thermal dynamics.

MEMBER BANERJEE: But why does it affect the --

(Simultaneous conversations.)

MEMBER SHACK: What you need to do is establish that to the degrees.

DR. WALLIS: That's right.

(Simultaneous conversations.)

MEMBER ABDEL-KHALIK: Based on your P-bar limit of 1.3, the potential maximum value of this spike is about 19 degrees. So what we need to find out is what is the actual magnitude of these spikes.

MR. KAI: Yes, and maybe we can have Bob Branum, our INC expert.

MR. BRANUM: We're gathering data for the spikes from studies we did that preceded the modification we're proposing for the plant. We'll have that available for you when we get it. Okay.

MR. KAI: Like I said, I have the studies in set points rather than temperature.

MR. RUSSELL: Can I add something to the discussion with regards to these spikes?

MR. KAI: This is Paul Russell, our Operations representative on our --

MR. RUSSELL: My name is Paul Russell. I'm a licensed operator at the Unit 3. From the operational standpoint and you did mention that they are just an annoyance to the operators, this is actually a very infrequent annoyance to us. We very rarely get the alarms that come in. It's not something that we'll see a drastic jump-up in our temperature indications. So it is a spike, not necessarily electronic-wise. It's a true indication of the temperature, but it is a very infrequent occurrence. It's not something that we see on a very regular basis.

DR. KRESS: But it's infrequent at the location you're measuring. Other parts of the system where you're not measuring, it could be happening all the time. I think that's really if you get into a materials concern that's where it would happen. But this sounds like a generic kind of issue for PWRs --

MR. KAI: Yes, it is.

DR. KRESS: It's not unique to the SPU.

MEMBER BANERJEE: But going back to what you are showing on the slide, you are saying you're installing an electronic filter and the purpose of this is to smooth out these spikes I take it.

MR. KAI: Yes.

MEMBER BANERJEE: But how does it affect the DNBR margin? On the top of the slide, you say DNBR margin and that's what I don't understand.

MR. KAI: Okay. Let me explain what we did because like I said, what we did in this cycle was we raised the set point. The alarm follows with the set point. It's like a couple of degrees different from the set point. So in this analysis because of a potential when we started up from the last cycle to get the spikes and cause pre trip alarms, they usually occur at the beginning of the cycle, we raised the set point, the OPDT/OTDT set points. So therefore they will trip later for those events, DNBR events, where it's credited and as a result you lose DNBR margin.

It's not the effect. It's the solution that we implement to try to overcome these --

DR. WALLIS: So you lose some margin because of this.

MR. KAI: What we did for this cycle is we actually physically raised the set point so it's closer to break point in order to.--

DR. WALLIS: How much do you raise it by? Three degrees? Four degrees?

MR. KAI: It's a complicated thing because OPDT/OTDT is an equation with coefficients with model t ave and Delta-t but what we ended up doing is raising the set point to as high as we could possibly make it and still show acceptable results for this current cycle.

MEMBER BROWN: Typically what you do is move your margin around. You don't just reduce it in one place. You're usually taking it from someplace else. So your overall margin may not be changing but you're using it from someplace else.

MR. KAI: Right. We're getting an

operational margin and reducing safety analysis margin.

CHAIRMAN SIEBER: That's why you need a margin manager.

MEMBER BANERJEE: But now with more channels and these hot conditions and things, you would expect these events to become more frequent -- Right?

MR. KAI: No, I mean, not the frequency. We did assume that the peaks would get bigger, in other words, because of the fluid.

MEMBER BANERJEE: Why would the peaks get bigger? Your powers are not higher.

DR. WALLIS: It's a flatter --

MR. KAI: When we did the initial studies we assumed that --

MEMBER BANERJEE: I'm trying to understand what's happening. So let me sort of give it back to you the way I see it. What's happening is that there are some channels where the outlet temperature is hotter than other channels. Because we have incomplete mixing before it gets to the hot leg, some of this hot fluid is going directly into the hot leg without having mixed with the colder fluid. Right?

MR. KAI: Correct.

MEMBER BANERJEE: So that gives you a temperature dispersion because of the lack of mixing. That has to be bounded obviously by your 19 degrees which is what Said calculated. It has to be less than that. If it is more than that, there's some other effect happening.

MR. KAI: Right.

MEMBER BANERJEE: But let's say it's five degrees or ten degrees or something due to this. Now as you are going to have more channels now which are going to produce hot fluid, you are going to have higher frequency of these spikes. The amount may not change because your power per channel is not increasing very much. Is that the correct idea?

MR. KAI: Yes.

MEMBER BANERJEE: The magnitude should not increase, but the frequency should.

MR. KAI: It's possible. Correct. But again, I would expect --

MEMBER BANERJEE: Or is there something else happening? I don't understand.

MR. KAI: No, I mean, how often it happens is more of a function of how the flow rotates or it goes from loop to loop. Okay. If it stays constant, the fact that you have higher temperatures of assemblies it doesn't make any difference. You're still going to see the same. You'll see a constant temperature.

MEMBER BANERJEE: And then either you're going to have a higher frequency or a longer duration, one or the other, because what you're saying is that there's incomplete mixing plus some sort of overall rotational behavior to the flow. I'm not clear why it happens, but let's say that's what happens. So you visit different hot legs with some frequency more or less. Without looking at the data, I have no idea what's going on. But assuming that this is some sort of flow pattern --

DR. WALLIS: It might be less evident. They have a flatter power distribution --

MEMBER BANERJEE: I know.

DR. KRESS: That may be the other way to go.

(Simultaneous conversations.)

MR. KAI: And that's exactly -- I mean, that's why the model -- We have to put in a solution for uprate and that's why we've done this, putting in a hot leg filter much like our sister plants.

DR. WALLIS: If you take the filter, then you're going to miss diagnosing what's really happening. So you ought to have at least a measurement of the real signal. You use the filter for your set points and stuff.

MR. KAI: Yes.

DR. WALLIS: But you ought to have a real indication because as you increase the power you want to know how does this anomaly change with power level. So you really have to keep recording the real signal.

MEMBER BANERJEE: Yes. It's an electronic filter for the purposes of a set point or whatever.

MR. KAI: Yes, but that's only for the set point.

MEMBER BANERJEE: For the set point. Right. You should be actually logging the data and trying to understand this thing. I mean, in some way, is it changing as you're raising power or what's happening to it?

MEMBER MAYNARD: I'm assuming that you also have thermal couples in the core. But we're talking about the RTDs used for plant trips and alarms.

CHAIRMAN SIEBER: In the loops.

MEMBER MAYNARD: But typically that's a measuring issue. You really have to use thermal couples in the head to see or in that area --

CHAIRMAN SIEBER: In that area.

MEMBER MAYNARD: -- To see what the temperatures and variations are really doing.

(Simultaneous conversations.)

CHAIRMAN SIEBER: Those thermal couples don't alarm.

MEMBER MAYNARD: Right.

CHAIRMAN SIEBER: So the operator is actually going to look at that in order to be able to see what's going on.

MEMBER BANERJEE: TheY are usually very slow anyway, they are filtering out temperature spikes to --

DR. WALLIS: What will concern me is what you would measure would depend on where you put the thermal couple. I don't know. This hot fluid, does it go to the top of the hot leg? Does it go to the bottom? They presumably tend to go to the top of the hot leg. So if you put your thermal couple lower down, you won't see it.

MR. KAI: And actually what you see, normally you would have -- These are RTDs. They're not thermal couple -- We will have them at various locations around the pipe and so you see them on one of them and not the other. It varies and it changes.

MEMBER BANERJEE: So you are putting the filter only so that you don't have to do things to your set point. It's to remove an operational --

MEMBER ARMIJO: That's not what their license is. The question is they put their filter at the input to the t hot signals. The t hot input is where the filter comes. In other words, you take -- That's the way I'm reading the document and I may be wrong. So the data, the output of the RTDs is fed through a filter before it gets to the measurement stuff where you do all the gains and zeros and all that kind of stuff and then before it goes and does t ave and the delta-T. That's the way I read Page 2.4-10.

MR. KAI: Correct.

MEMBER BROWN: So it's not just a filter on the alarm. It's a filter on the whole t hot and the whole --

(Simultaneous conversations.)

DR. WALLIS: You don't want to do that. You really want to know what the oscillation is, don't you? The real thing? You have to record an unfiltered signal somewhere.

MR. BRANUM: Excuse me. My name is Robert Branum and I'm the INC Engineer on Project. The t hot signals will be available in our plant process computer unfiltered so we can monitor those.

DR. WALLIS: Okay.

MR. BRANUM: Where we are installing the filter is downstream of where the three t hots per loop are combined and that's the point at which they will be filtered. The filter signal will be used as an input to the t hot or the t ave and the delta-t circuits which are computing the set points of the trips for the OP and OT Delta-t safety functions.

MEMBER BROWN: So you take the raw data and you feed it off to another acquisition system.

MR. BRANUM: That's correct.

MEMBER BROWN: To go along with the normal operational instrumentation. Is that the way?

MR. BRANUM: That's correct. Now a couple of points that have been talked about here. Michael mentioned that in this present cycle I want to clarify the present operating cycle, now that's not post SPU but the present operating cycle, we implemented an optimization program where we actually raised the set points of our OP/OT Delta-t functions to get them above these spikes that we've been seeing. Okay. And as a result, it ate into some of our DNBR margin.

Now implementing the SPU, that will not be an acceptable result. So by installing the t hot filter, we'll be able to reduce the set points back down and it actually gained the margin back because the t hot filters will be filtering out the momentary spikes that we're seeing in the individual combined t. hot channels.

MEMBER BROWN: Another thing. Does a time response increase, if that reduction in set point is a far greater effect than the time response, the four second time response, which you refer to these as four second time response filters or seconds filters? Excuse me.

MR. BRANUM: Yes, the filters are four seconds. The duration of the spikes are much shorter than the four seconds that we've seen to date.

MEMBER BROWN: Like what? A half a second? A quarter of a second?

MR. BRANUM: A second or so.

MEMBER BROWN: Okay.

MR. BRANUM: The result of the four second filter time delay has been considered in the dynamic safety analyses and in response to the plant to the OP and OT Delta-t trips.

MEMBER BROWN: So the tradeoff is positive relative to the reduction in set points does more than delay, the additional delay, in the time response of the incident.

MR. BRANUM: That's correct.

MEMBER BROWN: To help you recover that margin.

MR. BRANUM: That's correct.

MEMBER BONACA: Now you made a statement before that this has been implemented before in other plants.

MR. BRANUM: Yes. Correct.

MR. KAI: There are a number of Westinghouse plants that have implemented it. I don't --

MEMBER BROWN: Do you want to make that statement before you know?

MR. KAI: I don't know the names of --

MR. HUEGEL: This is Dave Huegel with Westinghouse and if you look at any of the Westinghouse plants out there you are going to see fluctuations in the t hot signal and depending upon if you have fast response or slow response RTDs installed in the thermal wells and they are at 120 degree angles about the hot leg and we take the average of those signals, you want to make sure that the spiking that you're seeing and we see this in all plants that you're filtering that so that it doesn't cause an unwanted trip signal.

And as part of the evaluation for EPU or for any plant for that matter, we do take into consideration the set point, how high we set it and all the associated dynamic compensation. It's a very complicated process that we go through to ensure that the plant has sufficient operating margin. But again, you'll see this fluctuation in t hot signal on any plant that's out there operating and we get an average signal again from each of the four loops and we've installed this optimized set point as we've called probably on, I'd say, close to eight to ten plants where we looked at the entire package of the set point, the OTDT and OPDT where they are set and also the filters we have in the RTDs and then also the dynamic compensation and the lead lag functions on t ave and also on Delta-t to ensure that you have (1) met your DNB criterion and (2) ensure that you have sufficient operational margins so that the plant can safely operate without getting spurious trips. Does that answer your question?

MEMBER BANERJEE: This whole methodology has come before the Committee before and sort of exposed in detail and been approved in some ways.

MR. HUEGEL: The methodology for calculating the set points is contained in 8745 which was approved in 1986.

CHAIRMAN SIEBER: W-cap.

MR. HUEGEL: Right. I'm sorry. It's a Westinghouse W-cap.

CHAIRMAN SIEBER: We don't approve W-caps. But the staff does.

MR. HUEGEL: But it's a methodology that we've been using in Westinghouse plants for years. Does that help?

MEMBER BANERJEE: Yes. It helps. I'm not -- I still don't understand what --

DR. KRESS: Why this is happening?

MEMBER BANERJEE: Yes, why this happens. But more than that, it seems by doing a little filtering you're getting some DNB margin. DNB is a real thing. How does it change the margin? That's what I don't understand.

MR. HUEGEL: As it was explained, as Mike had stated, what you're doing is you're changing not the initial condition but you're changing where you're tripping and that is how it's affecting your DNBR. If you delay because you've added a filter, when you trip on the OPDT/OTDT trip function you are affecting the

DNBR and that's how it plays out, not as an initial condition per se, but you're changing where you trip the plant and that's how it affects the DNBR. You run these different transients. The protection function kicks in and trips you so that you meet the applicable acceptance criterion. Typically for OPDT/OTDT we're trying to demonstrate the DNB design basis is satisfied.

MEMBER BONACA: Have you noticed a dependency between the frequency of the spiking and the amount of new fuel that you put in the reactor?

MR. HUEGEL: No, I don't think that they're related. I'm sorry. The question again.

MEMBER BONACA: I was wondering was there a dependency in the frequency to the spiking occurs in the number of new fresh fuel assemblies you put in there.

MR. HUEGEL: No, there's not any relation.

MEMBER BONACA: It seems to be a signature of the plant.

MR. HUEGEL: I think you would see -- You're going to see that fluctuation in the t hot signal no matter what plant you look at and actually it's a problem because in the safety analysis we don't predict spiking in the safety analysis. If you would look at the t hot signal for any of the safety analysis that we do (1) it's a nice signal. Yet (2) at the plant they are incurring these spikes which when you amplify it with lead lag function gives you trips much sooner than we would predict in a safety analysis.

I wish I could take credit in the safety analysis for the spiking that occurs at the plant. I could give you a much better safety analysis answer. But again, you do have to make sure that you have filtered the t hot signal so that you aren't getting spurious trips and that's what we were trying to do. As Mike had explained earlier is when we evaluated the EPU we wanted to make sure that based upon historically what we've seen for a t hot signal that the filter that we'd be installing in addition to the exact set point was such that (1) you met your DNB design basis and as I stated earlier (2) also you would have sufficient operational margin. It's a fairly rigorous and detailed process that we go through to ensure that that all fits together.

MEMBER MAYNARD: I have a general question on this. I don't know that much about this system, but does Westinghouse or does the staff understand the mechanisms that are causing these temperature fluctuations and have they dispositioned those? The temperature changes are small. At least, that's what we're told. So maybe there is no problem. But it just seems like nature is trying to tell you something and you're filtering it out. And that's a little bit of a worry.

MR. HUEGEL: Again, what we're seeing in this t hot spiking is applicable to any plant. You're going to see noise in the RTD signal for any plant that you're looking at when you're measuring the t hot signal. We do try to account for it in a number of ways. Again, we have three RTDs in the hot leg to make sure that we have an average signal.

We also have in the uncertainty calculations a PMA term to make sure that we've accounted for any streaming that you would see in the hot leg so that that is accounted for not only in your t ave uncertainty that we've accounted for in the safety analysis but it also factors in the uncertainty calcs that go into the OTDT and OPDT. So we've accounted for them in two different places in addition to filtering the signal. But this is something you would see at any plant.

DR. KRESS: It's generic. It's not unique.

MR. HUEGEL: Yes. And again, the spiking is relatively small in terms of magnitude. But it can be depending upon if your plant has a fast response RTD or a slow response RTD as we call them. Your spiking can vary. But it can be tough to deal with. Again, I wish in the safety analysis space, we could take credit for it because they give a much earlier trip than the safety analysis would predict. Do you want to --

MR. KAI: Anything else that I can answer about that?

MEMBER BROWN: I had just a real quick question. One of your mods is the elimination of this automatic rod withdrawal feature.

MR. KAI: I'll get to that next.

MEMBER BROWN: Okay, Mike. What do you lose when you eliminate that feature from a safety side?

MR. HUEGEL: A nasty accident.

DR. WALLIS: What do these spikes look like? You have a hot spike. Do you have a cold spike as well?

MR. KAI: No.

DR. WALLIS: Just an average and then a hot spike? Is it like that?

MEMBER BROWN: You wouldn't expect a cold spike.

DR. WALLIS: No, a cold spike in the hot leg. And if you get hot spikes, you might have cold spikes, too, because cold eddies can come in as well as hot eddies.

MR. KAI: But the cold -- Remember you have the RCP and the --

DR. WALLIS: In the hot leg. You get cold spikes, the temperature goes down.

MEMBER MAYNARD: But you only get trip or alarm based on --

DR. WALLIS: I'm interested in what it looks like and if the cold spikes are bigger, then I might be more worried about fatigue. It would be nice if you could show the spikes. Give us a picture.

Can you sometime later in the day give us a trace of these spikes? That would help a great deal.

MEMBER BANERJEE: It would be very nice to see what really turns up.

MR. HUEGEL: They are looking into getting that information.

DR. WALLIS: Thank you. That would help.

CHAIRMAN SIEBER: What I would like to do is we have some outside things that some folks have to do and I'd like to end with Slide 15 and then break for lunch at that time.

MR. KAI: Okay. Now we're going to talk about the outer rod withdrawal.

CHAIRMAN SIEBER: We're on 14.

MR. KAI: We're still on 13. Okay. What we're going over is the modification section. On Slide 13 what we show is what we've done to assure that coming out of SPU we will have DNBR margin for our plant. We've talked about the other three items already, WRB2, WRB2M, the measured flow rate and the radial peaking factor and we've talked about the second bullet in terms of -- We talked about the first bullet which is the outer rod withdrawal.

The limiting DNBR event from Millstone 3 is the steam line break with coincident rod withdrawal and we have to assume that because DNRs are not qualified for a steam line break inside containment we postulate that the rod control system will be -- We will be able to withdraw the control rods in response to the steam line break and exacerbate the power excursion. So what we're doing like a number of other plants that have had this problem, we are essentially eliminating the capability for the rod control system to automatically withdraw the control rods. So in a steam line break, we would not have the control rods withdraw.

Now obviously the rod control system is for the operators. I mean that helps them manage the plant. Typically, for start ups where we're going to be withdrawing control rods, that's all done manually. We do not generally -- We never really have the operators increase power using an automatic rod withdrawal system. The operators must maintain positive control of reactivity. So it is very rarely used in terms of the operators.

DR. KRESS: So that was designed in as a nice feature to have, but it turns out it's not a very good thing to have.

MR. KAI: Yes, and you could probably go back and look --

CHAIRMAN SIEBER: They are between the one that's where an operator could look at it and smooth it out.

MR. KAI: Most of these systems, I'm talking about my old history here, they were designed in the early days for load foul where the plant would automatically increase power. We don't do that. So this feature, we'll obviously check with our operations staff to make sure that this will have no impact on the capability to increase power and start-ups, etc.

MEMBER ARMIJO: How do you normally operate when you're 100 percent power? The rods on auto or rods in manual?

MR. KAI: Rods on auto.

MEMBER ARMIJO: There are some advantages. I think the overall advantage of deleting the automatic rod withdrawal outweighs the other benefit. The other is that there is a change. The operators hear the rods clicking immediately and alerts them that there is a change.

(Simultaneous conversations.)

MR. KAI: Now, remember, normally you want 100 percent power.

MEMBER ARMIJO: Right.

MEMBER BROWN: But occasionally you'll get one clicking out of sync.

MR. KAI: Okay. And so that's another thing that we have done to restore DNBR margin. It essentially eliminates that as a DNBR limiting event.

We've also decreased the power range high flux neutron set point from 118 percent to 116.5 percent to assure that for rod withdrawals of power that we will maintain DNBR margin. So we made part of our modifications primarily to try and restore DNBR margin and if you look at the next page you can see how this is done and I tried to make this nonproprietary. So that's why there's no numbers in some of these columns.

But if you look, I explain how DNBR margin is calculated. We start with a correlation limit and that is based upon how well the correlation matches the experiment and the 9595 limit for the WRB2M is 1.14.

The next step that we do is we define a design limit. Again, this is proprietary, but we statistically combine the uncertainties and the initial condition parameters. Let's say temperature, pressure, peaking, etc. to statistically combine the uncertainties and get a design limit so that when we do the DNBR analysis, it's done at nominal conditions. The statistical uncertainties for the parameters have been statistically combined.

DR. WALLIS: What sort of number do you come up with typically?

MR. KAI: It's usually only 1.2.

DR. WALLIS: One point two. Okay.

MR. KAI: Proprietary numbers is my

problem.

Now if you look at our current cycle, we have a safety analysis. That's a DNBR configuration management tool that we use with Westinghouse's methodology. We establish a conservative limit which we call the safety analysis limit. I want to make sure it's not safety limit. It's called safety analysis limit. We do all the Chapter 15 analyses to show that we meet the more conservative safety analysis limit.

DR. WALLIS: The number that you simply arbitrarily define, isn't it?

MR. KAI: That's correct. It's a number that we arbitrarily defined to ensure that --

DR. WALLIS: When you changed it from 1.39 to 1.6, does that mean that you're now getting more safety margin?

MR. KAI: Correct.

DR. WALLIS: Okay.

MR. KAI: And the benefit there like I said before is the filter because previously we had moved the safety analysis limit all the way up to get as much margin as we could so that the set point would be these little spikes that we get --

DR. WALLIS: That's interesting because I would have thought with the power uprate that you could actually reduce this. But you increased it.

MR. KAI: Yes. Obviously, that's a big concern.

MEMBER BANERJEE: They can use the filter.

MR. KAI: Right. That was exactly -- You're exactly right, Graham. That's what we're concerned about initially when I did this because I knew we had no -- We do not have margin coming out of this if we continued our same practice in terms of operational margin for this effect.

MEMBER BROWN: What's a relationship between DNBR design limit and your safety analysis. I haven't seen that differentiation before.

MR. KAI: Okay. That --

MEMBER BROWN: Should I be educated in some other point?

MR. KAI: Remember the 1.0 obviously, is the DNBR, a DNB at 1.0, 1.14 is the 95th/95th percentile that bounds the data used for the calculation of the uncertainty. Design limit statistically combines this 9595 limit with uncertainties --

MEMBER BROWN: You use the probability as opposed to the deterministic approach.

MR. KAI: Yes.

MEMBER BROWN: That's what that is, isn't it?

MR. KAI: Okay. Now we're talking about --

MEMBER BROWN: Forget that statement. Just go on.

MR. KAI: Okay. Not only obviously are there uncertainty in the correlation but there's uncertainty in the your instrumentation and initial conditions. For example, for t ave and pressure -- .

MEMBER BROWN: Got it.

MR. KAI: So those uncertainties in instrumentation are then calculated and then statistically combined with the DNBR uncertainty. Then the DNBR analysis at nominal conditions because the uncertainty has been factored into design limit.

MEMBER BROWN: Okay.

DR. KRESS: How does that relate to the safety analysis one?

MR. KAI: Okay. So in theory, you can go right up to the design limit and assure yourself you would not be in DNB. But in practice that's not a good idea. I mean obviously issues will come up over the other times you're operating. So what we do is we as a utility establish --

(Simultaneous conversations.)

MEMBER BROWN: The size of a design basis -- One is a design basis consider where you do your design design and the other part is how you do the safety analysis relative to that. You just made it higher.

MR. KAI: Right.

DR. KRESS: This also allows the margin for fuel design if you had any issues with that.

MR. KAI: Right.

DR. KRESS: This allows --

MR. KAI: And that is shown below in fuel issues and instrument biases which cannot be readily combined because some of them have biases.

MEMBER BANERJEE: I guess it really -- What you call the generic margin, it would be an interesting number to know how's that affected for --

CHAIRMAN SIEBER: Proprietary.

MR. KAI: It's proprietary.

MEMBER BANERJEE: Well, you can -- I guess after the meeting. Right?

CHAIRMAN SIEBER: Well, okay. We'll do it after lunch. Do you want to --

MR. KAI: Yes. I think this is probably

a good stopping point unless you want me to go on. Would you like me to go onto 15?

CHAIRMAN SIEBER: Is this a good stopping point?

MR. KAI: This is the generic --

CHAIRMAN SIEBER: Or is the end of this slide a good stopping point?

MR. KAI: The end of slide 14.

CHAIRMAN SIEBER: Okay.

MR. KAI: So like we pointed out, we do have a penalty for -- When we did the initial -- When we initially set the safety analysis limit, we underestimated what we needed for -- from power so instead of redoing all the calcs that we'd done we assessed that as an additional penalty against the generic margin.

MEMBER BROWN: So this is -- I was kind of trying to figure that out. This is a bank account you're keeping.

MR. KAI: Yes. Correct. It's a bank account.

MEMBER BROWN: You can only go in and you can only withdraw so much as you go through and find -- So you allow yourself to not meet your requirement.

CHAIRMAN SIEBER: Right.

MR. KAI: Yes, and typically like I said, that was not the ideal solution.

CHAIRMAN SIEBER: You want the margin to keep --

MEMBER BROWN: I'm seeing that. That's why I had to ask.

MR. KAI: So the net result is if you subtract these penalties from the generic margin you get really truly what's available margin for issues that may come up or future problems.

DR. WALLIS: So what are all these blanks in the table? Why didn't you just put numbers there?

MR. KAI: Proprietary.

DR. WALLIS: Proprietary?

CHAIRMAN SIEBER: Trying to keep it such that --

DR. WALLIS: You mean, I have to read those.

MEMBER BROWN: No, there's another table.

DR. WALLIS: There's another table.

MEMBER BROWN: It has some of them for those who are able to see it, I guess.

CHAIRMAN SIEBER: But we have to close the session.

MEMBER BROWN: And we don't want to do

that.

CHAIRMAN SIEBER: Right.

MEMBER ABDEL-KHALIK: Are you going to continue with Slide no. 15?

MR. KAI: Yes. Okay. Now let me just go on and go past 14. What this shows are the DNBR results for all the different transients that --

MEMBER ABDEL-KHALIK: What do you use W3 for?

MR. KAI: W3 is used where the WRB2M does not apply. That's used for low pressure transients, for example, steam line break where the pressure and which you're calculating DNBR is outside the range of the correlation. It's also used for rod withdrawal subcritical where DNBR is occurring below the first grid because the correlation takes into account the mixing grids. So for those two transients, we would use either WRB2 or W3 depending on what is appropriate for that transient.

DR. WALLIS: The only place here you seem to be close to some limit is this rod withdrawal from subcritical.

MS. YOUNG: Yes, and that's a little misleading. Okay. And the big thing there is to look at the pumps.

DR. WALLIS: Yes, why the two instead of three?

MR. KAI: This has been a limiting transient for Millstone 3. I don't know. Maybe it's Cycle 4. So what we did, the standard methodology for Westinghouse is two RCPs are running when you're subcritical. We changed our specification to require three RCPs whenever we are capable of withdrawing control rods when they are coupled. So therefore our current analysis is based on withdrawal of the three RCPs.

When we went to SPU, we actually showed it was okay at two RCPs. We had no intention to run three.

DR. WALLIS: If you would run three, it would look much better.

MR. KAI: Exactly right. We had no intention of changing the spec.

DR. WALLIS: So apples to apples, you'd actually come out better.

MR. KAI: Yes. Correct.

DR. WALLIS: Why don't you show us that as well?

MR. KAI: We did not calculate with three RCPs because we made it with two RCPs. But like I said, we had no intention of changing the specification. So yes, I --

CHAIRMAN SIEBER: You don't withdraw rods until you're at temperature with four pumps running.

MR. KAI: Correct.

DR. WALLIS: Right.

MEMBER BROWN: This assumes you do. Right?

CHAIRMAN SIEBER: Pardon?

MEMBER BROWN: This assumes you do.

CHAIRMAN SIEBER: Well, this is --

(Simultaneous conversations.)

MR. KAI: This is a failure. Okay. We're looking at some kind of either a failure in the control system -- But now that we've eliminated the --

CHAIRMAN SIEBER: You eliminated the automatic rod withdrawal. So it has to be a failure by the operator.

MR. KAI: The operator. Correct.

DR. WALLIS: Yes.

MR. KAI: So yes.

MEMBER BROWN: But your tech spec still requires three.

MR. KAI: It will require three and we did not change that.

DR. WALLIS: This is from subcritical though. This is with a steam line break maybe.

MR. KAI: No.

CHAIRMAN SIEBER: Okay. Go ahead.

MR. KAI: Anyway, what this shows is current and SPU and as you can see in some cases we're getting margin and that is really what the net result of all the changes that we made. Even at the higher power we're maintaining DNBR margin and that was our goal and not to use --

CHAIRMAN SIEBER: Some of that is changes in set points. Some of it is changes in analysis methods. Right?

MR. KAI: Yes. Correct. And to me the big part is the change in the hardware changes that we made.

MEMBER ABDEL-KHALIK: Can you explain to me physically how you would gain that much margin for the inadvertent opening of a PORV?

(Simultaneous conversations.)

MR. KAI: No, this is a short transient. Remember DNBR is going to be a problem primarily for power increases and temperature increases. In this case, the pressure decreased. So this is primarily a function of the DNBR correlation and those types of changes.

MEMBER ABDEL-KHALIK: There is that much difference between the two DNBR correlations, the WRB2M and WRB2. Is that what you're saying?

MR. KAI: No. I can look at this, but this is -- We'll get back to it.

MEMBER ABDEL-KHALIK: Okay.

CHAIRMAN SIEBER: Anything else in this arena? Any questions from anyone?

MEMBER BROWN: We're coming back to the other parts after lunch.

CHAIRMAN SIEBER: Yes. We just want to continue on.

DR. WALLIS: So the next slide would wrap this one.

CHAIRMAN SIEBER: This is a natural time for us to break. Why don't we return at 1:00 p.m.

DR. WALLIS: Can't start until 1:30 p.m.

(Simultaneous conversations.)

CHAIRMAN SIEBER: Okay, 1:30 p.m.

(Whereupon, at 11:54 a.m., the above-entitled matter recessed and reconvene at 1:30 p.m.)

CHAIRMAN SIEBER: Let's take our places so we can resume the Subcommittee on Power Uprates. Are we ready?

MR. KAI: Yes. One thing, Mr. Chairman, I'd like to know is, what do I still owe you in terms of answers? What do I still owe you in terms of answers?

MEMBER ABDEL-KHALIK: I have two items on my list.

MR. KAI: Okay.

MEMBER ABDEL-KHALIK: Delta-t for the hot leg temperature spikes, both positive and negative, and why is the DNBR for PORV opening significantly different? Is that a result of the difference in the DNBR correlation?

MR. KAI: Okay.

DR. WALLIS: You were going to give us a trace of these spikes, you were going to give us actual trace of temperature in the hot leg so we could look at it. I think your Westinghouse friend was going to go after that.

MR. KAI: Yes. Understand.

CHAIRMAN SIEBER: I have nothing. Does anybody else have anything, any questions that are still open?

PARTICIPANT: Excuse me, Mike. I can give an answer in regards to the question about the traces. We'll get to data from the plant that will show the traces for the t h indications, and we'll use that. We'll provide that to you tomorrow.

MR. KAI: I actually have the traces in terms of the set points. And really, the Delta-t is just a scaling from the set point, so I could show you that.

To go back to your question on DNBR, the question was based on the stuck open --

MEMBER BROWN: I'm sorry, 15 in your slides.

MR. KAI: Fifteen, right. Okay. There are two reasons why this has increased, one of which I explained, which is the DNBR correlation. The 100 percent power, which is what the DNBR correlation starts, it's significantly higher at the start under the normal conditions.

This DNBR is actually calculated with the correlation that's in RETRAN, rather than a VIPRE. So what we have is a crude correlation, I shouldn't say crude, but we have a correlation of DNBR in RETRAN, and we use that for events like this where there's expected to be very large margins to DNBR. And the way that RETRAN works is that it looks at the delta between where you start and where your safety limit is, and so since in both cases you're starting higher and the safety limit is higher, when it calculates the delta in DNBR for the current condition, it's calculating a much higher value. So it's a simplified method for estimating DNBR for those events where DNBR is really not a concern.

MEMBER MAYNARD: Was it calculated the same way for the current and for the --

MR. KAI: Yes. Right. But remember it's a delta. Okay? Because --

MEMBER MAYNARD: Well, I understand that, but it's --

MR. KAI: It's the same way. But the current one reflects both the fact that the initial DNBR starts off lower because of the WRB2 correlation, and also that our safety limit is lower. So what it's doing is calculating the percentage change, and so it results in the current one being a lot lower.

MEMBER ABDEL-KHALIK: I'm sorry.

DR. WALLIS: I'm thoroughly confused.

MR. KAI: Well --

DR. WALLIS: The safety limit has changed. Why does that change this?

MR. KAI: Well, I guess, maybe we'll ask Dave Huegel to try and take a second shot at explaining this.

MR. HUEGEL: Yes. This is Dave Huegel with Westinghouse, and as Mike had explained, what we're doing with RETRAN is really a simplified calculation for those transients where we don't expect to see any peaking occurring in the core, like you would see for a dropped rod event. And what we do, as Mike had explained, is we take the nominal conditions, and with the case of your current analysis, calculate a nominal DNBR based upon the nominal conditions, your current conditions. And say that gives you a value for DNBR of 2.2, then you look at the SPU conditions. You have a higher power level, you have a lower f-Delta-h, you have a higher flow rate, plus you're using WRB2M correlation, which gives you some amount of margin. And you end up with a higher initial DNBR.

And what RETRAN is then doing is it takes the core thermal limits, which are based upon the VIPRE calculations, which are doing a detailed calculation of what your limits are in terms of changes in temperature, pressure and power, and then you're seeing, during the transient, as my power, temperature and pressure are changing, how quickly do I approach the DNBR limit?

Now, remember with the current condition you have a DNBR of 1.39, so you can come down and you're going to be at a minimum DNBR of 1.584, so you're significantly above your DNB limit, which for your current plant condition is 1.39. Now, you go to the SPU condition, and as Mike had noted, you have a higher initial DNBR condition. You then go through the transient looking at your temperature, pressure, power changes, but instead of being compared to a DNBR limit of 1.39, you are now doing the calculation compared to a 1.60 limit. So again, you are having margin to the limit. You can't compare the two, because you have a different safety analysis limit. You have a different nominal initial DNBR limit and you're using a different correlation. So you can't compare these one-for-one.

But what this is showing you is that you do have, in both cases, significant margin to limit, whether it's for the current design, which is a limit of 1.39 for the safety analysis limit, or whether it's for the SPU condition, which has a limit of 1.60. But in both cases --

DR. WALLIS: Both are rod withdrawal at power, which is less than 1.60.

MR. HUEGEL: Well, that gets into what Mike had talked about earlier, with the concerns with the addition of the filter because of the hot leg spiking, we took some of that margin, and when we ran all the different cases for the rod withdrawal at power what we found is we went below the limit of 1.60, so we had to assign some generic DNB margin, because again, your true limit is the design limit, which is down around 1.22.

DR. WALLIS: So what's the function of 1.6, if the real limit is 1.2? I don't understand this at all.

MR. HUEGEL: The 1.60 is to provide you with a DNBR limit that again, as Mike had explained, provides you with margin to assess unknown things that come up on the plant that you may need to address.

DR. WALLIS: But then when you get below it, you do something else then?

MR. HUEGEL: Typically, the process that we would follow is we would try and demonstrate that you meet the safety analysis limit across the board. However, knowing that we have the question about the spiking in the hot leg temperature, we added a filter, knowing that yes, you might drop a little bit below the safety analysis limit, but that's an okay condition because again, the true limit is the design limit. And we allocated some amount of generic DNB margin for that penalty. So in the end, we are still meeting the design-basis limit of the plant.

DR. WALLIS: So what does the NRC do then? They check that you're above the design basis, or above something else?

MR. HUEGEL: The true licensing limit is the design limit.

DR. WALLIS: Well, the safety analysis limit is a somewhat ephemeral thing that you can vary around to suit yourselves?

MR. HUEGEL: That's probably one term you could use, yes.

MEMBER SHACK: It's the utility's decision.

MR. HUEGEL: Yes. It's really up to the utility to decide what that limit needs to be.

MEMBER BROWN: Under the current design, isn't the rod withdrawal less than your safety analysis limit? It's 1.39, it's 1.38. Am I reading your chart wrong?

MR. KAI: No, you're correct.

MEMBER BROWN: Before, they had the same exact generic margin. They had to go borrow. They were still below the limit.

MR. KAI: Okay. Now, let me explain this slide, which is different.

MEMBER BROWN: He's really digging this hole.

MR. KAI: Okay. We have a different limit depending on whatever the assembly has, a thimble plug, or it does not have a thimble plug. And unfortunately, I put in on Slide 14 the result with the thimble plug, and I put on Slide 15 without the thimble plug. So there actually are two limits here, which I took out to simplify it, but obviously, caused more confusion.

If you look at Slide 14, the 1.29, there are really two limits, there's a 1.39 and a 1.37, depending on whether you have the thimble plugs in or not. So I apologize for that, I really should have put the two numbers on Slide 14.

MEMBER SHACK: So you don't meet it.

MR. KAI: No, we do meet the 1.37.

MEMBER BROWN: Okay. So the 1.38 goes with the thimble plugs.

MR. KAI: Correct.

DR. WALLIS: So what you're trying to convince us about is that the margins are unchanged.

MR. KAI: We've actually got --

DR. WALLIS: All your arguments lead to this conclusion, that the margins are essentially unchanged.

MR. KAI: Yes. Right.

DR. WALLIS: And it's a little bit fuzzy what that really means in terms of these numbers.

MEMBER SHACK: It's probably safer to say that he has margin. Whether he's got more margin because he changed his analysis method and everything, physically --

CHAIRMAN SIEBER: The margins are the same. The numbers are different.

MEMBER MAYNARD: Yes, I'm with you. I think the important thing is whether or not there's adequate margin. I think you have to be real careful when you go to try to compare margins, when you change analyses, and you change other things.

CHAIRMAN SIEBER: Right.

MEMBER MAYNARD: And there is a lot of this, you use it to do set points, and you can move it around. At the end of the day, though, the bottom line is do you meet your regulatory requirements, and do you have margins?

DR. WALLIS: And how much margin is adequate? I don't know. I mean, that seems to be a very iffy thing.

MEMBER SHACK: Regulatory limits.

CHAIRMAN SIEBER: Yes. I think one of the confusing things is that currently SPU analysis in some cases used different assumptions.

MR. KAI: Right. And that --

CHAIRMAN SIEBER: So you get different answers, which are sometimes more difficult to understand than meets the eye.

MEMBER BROWN: Well, they don't scale directly either. They don't scale directly.

CHAIRMAN SIEBER: Right.

MR. KAI: Right. You're exactly right.

DR. WALLIS: You're trying to explain it to us, but I think if you tried to explain it to a judge and a jury, you'd have a tough time, because they'd want you to show that A is bigger than B, and which A are you talking about, and which B? It's not clear.

MR. HUEGEL: I mean, at the end of the day, what we're trying to demonstrate is that the design limit is satisfied, and that's what we've done.

DR. WALLIS: The design limit is satisfied.

MR. HUEGEL: Yes. And that's the true regulatory limit.

DR. WALLIS: Is that really what you're

trying to show? That should be what your conclusion is then. That should be the conclusion then, shouldn't it?

MR. HUEGEL: Yes. That's correct.

MR. KAI: Okay. Go on to where we left off, 16. I'll skip that and go to 17. There's nothing much on 16, you can look at the results. But in terms of RCS overpressure and some general pressure, the results are essentially identical, it's really unchanged.

CHAIRMAN SIEBER: Okay.

DR. WALLIS: So let's see now. Let me see this. Turbine trip is exactly at the limit currently, or is that a misprint?

MR. KAI: No, it's about 20 --

DR. WALLIS: Thirty and 20, 30 and 20, the same.

MR. KAI: Right.

MEMBER BROWN: All right.

MR. KAI: I'm sorry. That's a typo.

DR. WALLIS: That's a typo?

MR. KAI: Correct.

DR. WALLIS: What should it be, should it be 1340?

MR. KAI: I apologize for that.

DR. WALLIS: What should it be? You'll find it.

Well, the message is it's less than 1320. Is that it?

MR. KAI: Yes.

DR. WALLIS: And 1302 isn't somehow a reversal of the two and the zero?

(Laughter.)

MR. KAI: No.

DR. WALLIS: So this is psia assuming an atmospheric pressure which is average?

MR. KAI: Correct.

DR. WALLIS: No hurricane, or high pressure region or something which would change it by one psi?

MR. KAI: Correct. This is assuming atmospheric.

MEMBER ABDEL-KHALIK: So how can -- I know the difference is relatively small, but how can the value for SPU for the turbine trip on the primary side be less than the current value?

MR. KAI: Well, the steam --

MEMBER ABDEL-KHALIK: No, no. I'm not talking about the secondary side. I'm talking about the primary side.

MR. KAI: Right. And you've got to keep in mind we've also gone from LOFTRAN to RETRAN, and these small differences, I believe, are due to the fact that we're using RETRAN versus LOFTRAN. The current one is LOFTRAN, and the SPU one is done with RETRAN.

In terms of overpressure limit, remember that we've got pressurizer safety valve that will assure that the pressure remains at the pressurizer safety set point, so these small differences are due to the switch in methodology from --

MEMBER ABDEL-KHALIK: But if you had not changed methodology, shouldn't this number be higher?

MR. KAI: No. Again, the pressure is controlled primarily by safety valve set point, which is unchanged.

MEMBER ABDEL-KHALIK: What is the safety valve set point? What's the pressurizer safety valve set point?

MR. KAI: Well, we assume 2,500 psi plus 3 percent uncertainty.

MEMBER BROWN: To answer your question on 1320, your license says it is 1320, 1319.6.

(Simultaneous conversations.)

MEMBER BROWN: A .4 psi margin.

DR. WALLIS: Point four psi margin unless there's a very low pressure region going by.

MEMBER BROWN: I'm not going to argue about what the current might be, it's just that's what --

DR. WALLIS: I'm always bothered by these psias, because it's presumably psig that bursts things.

PARTICIPANT: I'm surprised that for the current the turbine trip was still so high, 2731. You're very close to the limit.

MR. KAI: Correct. That's -

MEMBER ABDEL-KHALIK: Now, you don't get an automatic reactor trip on a turbine trip?

MR. KAI: We don't credit it.

MEMBER ABDEL-KHALIK: You don't credit it.

MR. KAI: Correct.

(Simultaneous conversations.)

DR. WALLIS: Now, let's go back to that. This is inside containment, isn't it? So the actual pressure bursting this thing is this pressure minus the containment pressure? Why is the criterion an absolute pressure? Isn't there a difference in pressure between what's inside and what's outside that bursts these things?

CHAIRMAN SIEBER: The containment is sub-atmospheric.

DR. WALLIS: Sub-atmospheric.

CHAIRMAN SIEBER: Yes.

DR. WALLIS: So I have to take away from this whatever, 10 psi or something to get the bursting pressure? Presumably, the ASME gives you the pressure difference, doesn't it?

MR. KAI: Right.

DR. WALLIS: So are you quoting an ASME number or an ASME number plus the containment pressure here? What are you doing?

MR. KAI: First of all, the peak pressure is at the reactor vessel plant, RCS discharge. And it's calculated, so the containment impact is not on this pressure, but on the safety valve. Correct? The safety --

PARTICIPANT: Isn't this measured in the pressurizer?

MR. KAI: Pardon?

PARTICIPANT: Is this in the pressurizer?

MR. KAI: No. This is at the highest pressure point in the system.

PARTICIPANT: In the RCS.

MR. KAI: It's in the RCS. So it would be at the pump discharge, with the largest hydrostatic head at --

MEMBER MAYNARD: We have to be careful. We're talking sometimes RCS, sometimes about secondary side. You were talking primary side.

(Simultaneous conversations.)

DR. WALLIS: The secondary side is somewhat different.

MR. HUEGEL: Well, I think if you looked at the ASME requirements, I think it's 110 percent of the design pressure, which is given in psig.

CHAIRMAN SIEBER: Right.

MR. HUEGEL: So you take that times 110 percent, and then if you want to present the result in psia, you can add the 14.7.

DR. WALLIS: Is 2750 an ASME number, or is it ASME and --

MR. HUEGEL: I've seen in the ASME requirements, the psig. However, I've seen in the licensing documentation from the NRC, psia. I've seen results presented typically in a licensing document in psia.

DR. WALLIS: The reason to worry is that you're so close to the limit that it makes a difference.

MR. HUEGEL: Understood. But we are meeting 110 percent of the design pressure, as required by the ASME requirements.

CHAIRMAN SIEBER: That's 2500.

DR. WALLIS: That's 2500?

MR. HUEGEL: Twenty-two thirty-five is your design pressure, psig, plus 110 percent.

DR. WALLIS: Is how much?

CHAIRMAN SIEBER: Twenty-seven fifty.

DR. WALLIS: Twenty-seven fifty psig?

MR. HUEGEL: I'm sorry, 2500. Right.

CHAIRMAN SIEBER: The g.

MR. HUEGEL: Right.

CHAIRMAN SIEBER: ASME.

MR. HUEGEL: Right.

DR. WALLIS: So sump pressure in containment, you're okay.

MR. HUEGEL: Right.

MR. KAI: Okay. We'll now talk about the pressurizer overflow. As we just discussed before, one of the key factors here was the setting of the initial pressurizer level, so we took into account the results of these two events that we've analyzed here, the results of which are shown on Slide 20. We looked at loss of feed, and we looked at the inadvertent ECCS actuation.

One of the things that we found was that, number one, is that we did need a design change to provide us marginal inadvertent ECCS operation. In terms of operator action, this is one of the more challenging events that we analyzed. Currently, the way the current analysis shows is that, as you can see, with the PORVs available that we will reach a water solid condition in about 18.7 minutes. With the PORVs not available, it's 10-1/2 minutes, so we have, in terms of operator action, about 10 minutes.

DR. WALLIS: Oh, these are minutes here?

MR. KAI: These are minutes, correct.

MEMBER BANERJEE: So what was the hardware modification?

MR. KAI: Okay. That's what I'm going to go over. What we've done is -

MEMBER BROWN: Can you explain one -- I didn't understand a number. The maximum pressurized volume was 1,800 cubic feet.

MR. KAI: Correct.

MEMBER BROWN: In the loss of feedwater case, before you reach some value of what, 1,061 cubic feet of water volume in the pressurizer?

MR. KAI: Correct.

MEMBER BROWN: Now when you run that transient, you're at 1731.

MR. KAI: Correct.

MEMBER BROWN: So you've got 69 cubic feet left to go solid. And I don't know how many cubic feet there are per inch. Normally, you should be able to spit that out. It seems like you're almost solid for this transient. There's no times associated, or whatever it is, for the water relief associated with the loss of feedwater. And you can talk about the --

MR. KAI: Right.

PARTICIPANT: It just goes to the 1731 minute drop, so that's the peak it's going to reach.

MEMBER BROWN: It used to go to 1,061. And this is just a swell due to the temperature, the expansion of everything.

MR. KAI: Correct.

MEMBER BROWN: And the swell in the pressurizer.

PARTICIPANT: We're starting at a higher level, too.

MEMBER BROWN: But that's not a whole lot.

MR. KAI: No. You're exactly right. This is probably the biggest impact of the uprate --

MEMBER BROWN: Yes, that's a big number,

and you're not far from being solid. There's nothing talking about why that little bitty -- recognizing you meet the limit, but from the other part of the license review, that is the volume.

MR. KAI: Correct.

MEMBER BROWN: That is solid volume. Am I correct?

MR. HUEGEL: This is Dave Huegel. Just a point I wanted to make, is that keep in mind for the loss of normal feedwater, what we're truly trying to demonstrate is that you have adequate cooling post-reactor trip condition. This is not a true limit that we're meeting. We're just trying to avoid a condition where we go water solid during the time period that we've analyzed this event, because it's a situation that we don't want to start sending water out of the pores, but it's not a true limit.

The true limit is trying to demonstrate that you have adequate heat removal capability via your aux feed pumps for post-trip reactor trip condition. So I just want to point that out.

MEMBER BROWN: But isn't the concern, you don't want that power operated relief valve.

MR. HUEGEL: Sure.

MEMBER BROWN: You don't want it to open, because you don't want it to not --

MR. HUEGEL: You don't want it to pass water. We don't want it to pass water, it clearly passes steam.

DR. WALLIS: If the level goes all the way up there, it will open, will it not?

MR. HUEGEL: Yes. Another very good point is they are qualified for water relief.

DR. WALLIS: It does open, though, in this transient.

MR. HUEGEL: Yes, it does. Yes.

MEMBER BROWN: The water relief opens in this circumstance, or the steam?

MR. HUEGEL: No, the steam relief.

MEMBER BROWN: Oh, okay.

MR. HUEGEL: And the Millstone valves are qualified for water relief.

MEMBER BROWN: The steam relief valves are qualified for water relief.

MR. HUEGEL: Yes.

MR. KAI: What this transient is, if you can remember, number one, it's very conservatively analyzed. The RCS will continue to expand until your aux feed can match decay heat. And what you have in this situation is the power increase has pushed that time out when you match decay heat much further. If you look at decay heat in this time period, it's pretty flat, so the time that it takes for decay to drop such that you match decay heat, and then consequently start to pre-shrink the RCS is significantly extended. Like I said, this is probably the biggest impact on uprate.

Another thing to keep in mind here is that what we've done, the limiting case is actual off-site power available, and we're modeling an extremely conservative RC pump key on top of decay heat. So yes, I think you're right that this is a significant impact, and is a subject that we ought to monitor in our --

(Simultaneous conversations.)

MR. KAI: So I agree with that. The other thing is that we look at initially, and felt that we would show acceptable results with some loss of margin to overfill. And you're right, I think this is an area that we need to recognize going forward. It is very conservatively modeled. Like I said, it's very conservative that we add in, but we were able to show acceptable results.

MEMBER BANERJEE: What's the hardware change?

MR. KAI: Okay. The issue that we struggled with is the fact that we have high charging pumps. So what will happen is you'll get both trains to start up, and you'll inject ECCS, charging for 2-D ECCS valves at a significant rate. Remember, the charging pumps really can pump against 2750, so it can provide a significant flow.

MEMBER BROWN: SI and ECCS are synonymous?

MR. KAI: Right.

MEMBER BANERJEE: High pressure.

MEMBER BROWN: Safety injection or emergency core cooling, whatever. I just want to make sure they were --

MR. KAI: Right. No, they are normal charging pumps --

PARTICIPANT: They have five pumps at all high head. Right?

MR. KAI: Yes, we have two charging pumps running, two SI pumps, and two pumps that would start up in VNSI. And we actually have an event like this, so one of the things that we wanted to come out of this is to find a solution for this. I'll get to the mod that we did.

What we are adding is a permissive, a new permissive. The permissive uses an independent signal from the low RCS pressure RPS, so what it does is, that permissive has to sense low RCS pressure before it will allow the charging injection valves to open. Okay? So if you have an ECCS where RCS pressure is 2200 or 2500 or higher, the charging injection valves will not open. The rest of the ECCS will actuate. But obviously, because your pressure is high, they won't be injecting. So that permissive we've added. In this event of an inadvertent SI, what would happen is the charge injection valves won't open. The only injection will come through the RCP seals.

MEMBER STETKAR: Let me stop you right there. I noticed in your analysis that you consistently said those exact words, that the only injection will occur through the RCP seals. What about the normal charging line? How do you model the normal charging flow?

MR. KAI: Okay. If you have an inadvertent SI signal, the normal charging flow is isolated by redundant valves. The path is switched to the ECCS.

MEMBER STETKAR: Is that actually true for Millstone, because it's not typically true for most Westinghouse plants that I'm familiar with.

MR. KAI: Yes. It's definitely true for

Millstone.

MEMBER STETKAR: You isolate all charging.

MR. KAI: We isolate the normal charging --

MEMBER STETKAR: The normal charging line.

MR. KAI: Other than the seals, other than RCP seals. Josh can --

MR. HARTZ: My name is Josh Hartz. I work for Westinghouse Electric. Millstone Unit 3 has what we refer to as a high pressure emergency core cooling system.

MEMBER STETKAR: Right. So did Zion, where I worked.

MR. HARTZ: Okay. Many Westinghouse plants have that same design feature, to answer your question.

MEMBER STETKAR: Zion's charging line was not isolated on a safety injection.

MR. HARTZ: Ours definitely are.

MEMBER STETKAR: So hence my question.

MR. HARTZ: Well, the charging pumps bail out of the chemical and volume control system, and they take suction from --

MEMBER STETKAR: Zion's charging line was not isolated on a safety injection. Hence my question. I wanted to confirm that your charging line was isolated.

MEMBER BROWN: What's the initiating signal for your ECCS?

MR. KAI: It's an inadvertent signal. It is some kind of short on the --

(Simultaneous conversations.)

MEMBER BROWN: Now what about non-inadvertent, what triggers it?

MR. KAI: Well, there are a number of signals, including low pressurizer pressure is the one that normally is credited, and that's obviously for LOCAs. And in that situation --

MEMBER BROWN: But it's independent of a low pressure reactor trip.

MR. KAI: Correct. It's a totally different, totally independent.

MEMBER BROWN: And you're using the low pressure reactor trip to isolate --

MR. KAI: Correct.

MEMBER BROWN: To provide the --

MR. KAI: The permissive.

MEMBER BROWN: Drive the permissive.

MR. KAI: Correct.

MEMBER BROWN: How many plants use a

permissive to block a safety injection system?

MR. KAI: I don't know of anyone that does --

MEMBER BROWN: Are you the first?

MR. KAI: Yes.

MEMBER BROWN: Okay. That was what I got out of reading this stuff. That's an interesting point. I mean, how many places do we put a permissive in, as opposed to the actual plant sensed signal evidence of a LOCA, and then we put something else in, that says well, we don't really need it if we haven't got a low pressurizer reactor trip.

MR. KAI: And part of our advantage is like Josh said, we are a high pressure ECCS plant, which is an advantage because we have what we call HPSI, or in one sense we'll call it an intermediate HPSI plant, pumps, and we have low pressure --

MEMBER BANERJEE: With high pressure do the HPSI pumps inject -

MR. KAI: Mike O'Connor can explain more of the background. This has been an industry issue that we've been trying to resolve for a number of years.

The inadvertent SI causing a water solid situation. It's not, by any means, a desirable situation for our operators, or for our equipment.

MEMBER BROWN: So if you didn't do this fuel change, you'd be back in eight and a half --

PARTICIPANT: It would be shorter.

(Simultaneous conversations.)

MEMBER BROWN: So the only thing you're trying to do with this permissive to keep a longer dead time, not dead, wrong word, excuse me. Longer response time for the operator of the plant --

MR. KAI: Correct.

MEMBER BROWN: -- to take action in this circumstance. So as opposed to 10 minutes, you're trying to get to 30.

MR. KAI: And actually, there's a high likelihood that the operator will prevent the overflow altogether. I'll turn it over to Mike O'Connor.

MR. O'CONNOR: My name is Mike O'Connor. I'm the Manager of Systems and Component Engineering. When the project started, I was an on-shift Shift Manager working for Millstone Unit 3, and had a Senior Reactor Operator's license.

The design change, we have not categorized it as a block. Rather, we've let the actuation logic circuitry make a decision for operations and the plant on what's needed. Much like we don't get a automatic containment spray actuation for every safety injection, or every plant trip, we let containment pressure make that decision for us. That's what we're really doing with this modification to the plant at this time.

And as Mike was trying to describe, it takes that pressurizer fill time, or as we've been referring to it, the operation action time, that moves it from the -- currently we have it at 10 minutes, where an operator is required to make sure that there is a pressurizer relief valve path available at 10 minutes. That's the current action. Really extends that time frame out to near 70 minutes, because the only water going into the plant for that event would be the seal injection that we were discussing earlier.

MR. BUCHEIT: Mr. Chairman, if I might say a word. I'm Dave Bucheit, the Manager of Safety Engineering for Dominion. And we did look at this from a risk assessment perspective before we made the change, and we have some slides to address this later on. But essentially, from a risk tradeoff point of view, you're trading off the increased risk of plant transients from an inadvertent SI against the slight increase in risk of this permissive which now has to actuate. It was a risk --

(Simultaneous conversations.)

MEMBER STETKAR: Since you brought up the risk assessment, I was going to save it until then, but I didn't notice anything in your discussion of changes to the risk assessment, except the only discussion in the risk assessment regarding this modification is the assertion that the unreliability of the new logic is small compared to the unreliability of the injection valves. Okay. That's an assertion. Nobody did an analysis that I could see.

Are there any scenarios, either initiating events or developing scenarios in the risk assessment that either explicitly or implicitly include credit for safety injection where you would not have a low pressurizer pressure signal? Are there any scenarios where you include credit for safety injection under any conditions, and I don't know what your models are, and I don't know what scenarios you've looked at, but where you take credit in the PRA for a successful safety injection actuation where you would not have a low pressurizer pressure signal? I saw no discussion of that in your evaluation of plant design change impacts on the PRA. I would hope that the answer to that question is there are none, but I saw no discussion of that. So it's not clear to me that that question was even considered.

MR. BUCHEIT: The primary risk tradeoff was the risk of --

MEMBER STETKAR: That's a small system-level, not a functional level of a plant response evaluation. It'll come up in the main Committee meeting tomorrow, so I hope someone has an answer.

There are -- I've been in some studies where you don't get the low pressurizer pressure SI. You have to take credit for some other SI signal to get it started.

(Simultaneous conversations.)

MEMBER ABDEL-KHALIK: -- scenarios. Wouldn't this permissive make them a lot worse?

MR. KAI: No, it would not. You would not even get -- you wouldn't get an ATWS, a ECCS signal.

MEMBER ABDEL-KHALIK: Well, but the fact that you had PD pumps injecting borated water at fairly high pressure.

MR. KAI: We don't have PD pumps.

MEMBER ABDEL-KHALIK: They're centrifugal pumps with a very high shutoff head.

MR. KAI: Correct. But these scenarios do not generate low pressurizer pressure signal to generate an injection.

MEMBER ABDEL-KHALIK: That is not my question. So during those scenarios, you don't take credit of the fact that you can actually inject boron at high pressure?

MR. KAI: Yes, we do. But this permissive, what it affects is not the normal charging pathway, not the ECCS pathway. The normal charging pathway is still available.

CHAIRMAN SIEBER: Do you have the boron --

MR. KAI: No. We did. That's been removed.

CHAIRMAN SIEBER: You took it out. Okay.

MEMBER MAYNARD: Now, if I understand, I think the only open question is your question.

MEMBER STETKAR: Yes. Take a look at it, because I'm curious.

MEMBER BROWN: We've answered the one about nobody else does it. So that one is -- I think we know the answer to this. The other part that I didn't get to was there was a rough diagram showing the reactor trip signal, two out of four that you wanted to use going to something which then triggered this stuff, initiated, or uninitiated the permissive, or withdrew the permission. It appeared to be that was a single channel path, as opposed to a dual channel path where both paths would have to fail in order to initiate this unpermissive that you didn't want to occur. That's from the diagram. It's very simple -

MR. O'CONNOR: Perhaps I could clarify the simplified diagram. You have four completely different pressure sensing signals.

MEMBER BROWN: I got that part.

MR. O'CONNOR: Providing the two out of four.

MEMBER BROWN: Yes.

MR. O'CONNOR: That goes to two trains of components. The safety injection path that we're talking about is two completely fully qualified paths in parallel with each other, controlled by two separate valves that are powered from two separate electrical buses, so those signals, and the safety injection signal all go to the valve, two different trains.

We only need one of the two trains to function, to provide full safety injection from charging. I think that might be the piece that wasn't shown clearly for the diagram you're looking at.

MEMBER BROWN: So you're saying there's

two different -- one channel feeds one valve, and another -- you call that a train?

MR. KAI: Yes.

MEMBER BROWN: Okay. The other channel feeds the other valve?

MR. KAI: All four -

MEMBER BROWN: I know all four pressure is coming in, but something's got to say two out of four.

MR. KAI: Right.

MEMBER BROWN: And it's got to tell one valve two out of four you can do it, or be isolated, and the other's got another two out of four, or be isolated. Is it one or the other?

MR. KAI: Bob Burnham will clear up the logic circuitry for you.

MEMBER BROWN: Well, a diagram would be real nice, as opposed to words.

MR. BURNHAM: My name is Robert Burnham. I'm the I&C Design Engineer on the project. As Mike stated, we take four discrete pressurizer pressure signals. They have their own bi-stables. Those bi-stables feed into two parallel trains of solid-state protection. Inside the parallel trains of solid-state protection, two out of four voting is where it occurs, so this occurs completely separate -

MEMBER BROWN: Two sets of two out of four voting?

MR. BURNHAM: That's correct.

MEMBER BROWN: That's what I was asking.

MR. BURNHAM: Two sets of two out of four voting in each train of solid-state protection. That is where the signal is generated for the P-19 permissive. From each of those trains of solid-state protection, the P-19 signal hits independent auxiliary relays at the output of the solid-state protection cabinet. Those relays that are P-19 are placed in series for the existing safety injection signals, so that you need both an SI, and a P-19 permissive to open up the ECCS injection valve. And as Mike stated --

MEMBER BROWN: So there are two sets, one works on one valve, and one works on the other valve.

MR. BURNHAM: That's right.

MEMBER BROWN: A failure on one won't prevent the other one from opening, so you would at least get partial ECCS operation.

MR. BURNHAM: But it isn't partial. It's two 100 percent -

MEMBER BROWN: I understand that. By partial I mean part of both loads.

MR. KAI: Capability.

MR. BURNHAM: There is no single failure that can -

MEMBER BROWN: It's fully redundant all the way down.

MR. BURNHAM: That's correct.

MEMBER BROWN: That was not stated in the write-up. Okay. Thank you. A thousand words always substitutes for a nice simplified one-page picture.

MEMBER BANERJEE: I just want to follow-up on Charles' question regarding the -- is this the only plant that has these conditions?

MR. KAI: Has this mod? Yes.

MEMBER BANERJEE: You said the mod, but you get the pressurizer almost going solid. Are there other plants that do this?

MR. KAI: Yes. And, actually, like I said, this is being -- I'll let Mike O'Connor elaborate, but this has really been an issue that we have been trying to champion to find some way to provide some relief to the operator action time frames, but I'll let -

MEMBER BANERJEE: But other plants have the same conditions that arise. Right?

MR. KAI: Yes.

MEMBER BANERJEE: They have to live with short operator action, so why do you have to do all this stuff?

MR. BURNHAM: If I might answer that question for you, we were looking to improve the operation of the plant, and limit the need for an operator to take short credited operator action times to prevent the condition from happening in the plant when the logic circuitry was able to determine that it was not needed. So you're correct in that similarly constructed plants do have this, and they've resolved these issues in different manners over the years. And, previously, we've ended up with that short 10-minute operator action time at our facility, as a result of making sure that the PORVs on the pressurizer were qualified for water relief, as well as the downstream piping. And so that was -- and, in addition to that, we made some modifications about 10 years ago to ensure that there was a safety-grade actuation system of the PORVs.

PARTICIPANT: Just to follow-up on that if I could real quick. Part of the culture that we're trying to develop at Dominion is a culture of risk management. When we update our PRA models now, as a matter of course, we look for opportunities to reduce risk, and there are several examples of where we've done it.

This is a proactive example, proactive instance on our part where we've tried to reduce risk. The one question withstanding that we need to answer, we believe that this was a net risk-benefit to do this.

MEMBER BROWN: And I understand the 30 minutes. In our stuff that I used to work on for the Navy for the operator actions, we hated -- we fought to try to allow the operators to have about 30 minutes to respond to things. When we got down in the 5-10 minute range, we would take fairly heroic actions trying to work our way -- convince ourselves that they had to analyze-wise or put a hardware change in. I understand the desire to go away from it, depending on the operator who's in a 5-10 minute, what can be a chaotic time, is not as desirable as you would like, so I understand the need. I just was concerned about the redundancy of the whole setup, not excluding the other question relative, which is a very good question, also.

MR. KAI: Okay. Any other questions about that? We can move on.

MEMBER STETKAR: There's still pretty

short times for your limiting charging system malfunction.

MR. KAI: Yes. And that's always been there.

MEMBER STETKAR: Yes. I mean, that hasn't changed.

MR. KAI: Correct. Get to the design-basis results. And I've got a lot of the results here. I'm not going to go over them all in any detail. I will discuss a little bit about LOCA, to answer the questions about -

MEMBER ABDEL-KHALIK: Can we go back to the previous slide, please? Are the entries for the two rows when the PORVs are or are not available reversed?

MR. KAI: No.

MEMBER ABDEL-KHALIK: So if the PORVs are not available, it would take longer for the safeties to lift?

MR. KAI: What this -- okay. Yes, because you'll be at a higher pressure. What this is meant to show is really how much time the operators have. Okay? The first case where the PORVs are available, the PORVs are controlling pressure at about 2300 psi. Okay? So you will -- so to get water solid, you're going to be using more water out of the pressurizer faster, so you will get to a water solid condition, which you were designed for. Okay? So we are designed for that scenario. If the PORV is available and they're working controlling pressure at 2300, the pressurizer will go water solid -

MEMBER ABDEL-KHALIK: But the table is giving time for pressurizer safeties to open.

MR. KAI: Okay. Yes. Okay. Now, go to the next row. Okay? Do you understand what we did there? What we have there is, with the PORVs in operation controlling pressure, the time to currently fill the pressurizer solid is 8.7 minutes, and that's because the pressurizer is at 2300 psi. Okay? Get higher charging flows, going to discharge more water faster, and you will hit a water solid condition. Okay? If the PORV is inoperable, or not available, it will take longer to reach that water solid condition to open the safeties, and that's because you're going to be at a higher pressure. The safety valves will be cycling. The charging flow will be less, so you would get water discharge off the safeties at 10.5 minutes.

What we have the operators do is, in this time period of 10 minutes, they are to make sure that the PORVs are available. Their job is to make sure that the block valve, if it's closed, is open so that the PORVs will function. So the difference here is that the first number is done at RCS pressure of 2300 where the PORVs are controlling pressure. And the second line is with the RCS pressure at 2500 psi, where it's higher and -

MEMBER ABDEL-KHALIK: I still don't believe it.

PARTICIPANT: The key is on the first one where the PORVs are available, the safeties don't open. I guess water solid, but it's going out the PORV, not the safety.

MEMBER ABDEL-KHALIK: What do these numbers give? This is time for pressurizer safety valves to open.

PARTICIPANT: But that's not really where this one line -- the time is to become water solid, but the safeties don't open on that.

CHAIRMAN SIEBER: With water available, the safety valves won't work.

MR. KAI: Right.

MR. WALLIS: It's really time for the water to get up to the PORV.

MR. KAI: Well, it's hard to -

(Simultaneous speaking)

MEMBER MAYNARD: Mr. Chairman, it's 2:30. We haven't even heard anything from the staff. We have a number of subjects to go.

CHAIRMAN SIEBER: Why don't we try to go through this a lot quicker.

MR. KAI: I've given you all the design-basis results. One thing that I would use, we are using ASTRUM for large-break LOCA. We do have the results that compare the current, on page 24, they did the small break results, which actually will give you a much better idea of the LOCA impact, because in small break, we're using the same -

MEMBER POWERS: Let's go back to the rod ejection results.

MIKE: Okay.

MEMBER POWERS: You don't indicate what fraction of the core has suffered damaged rods with these kinds of energies.

MR. KAI: I'll ask -- I'll make sure our radiological guy -

MR. BARTON: Right here. Could you repeat the question, please?

MEMBER POWERS: Well, the kinds of maximum strut energies did you have in the rods here are sufficient to break them, and they're going to go rod ejection, so I assume there would be a lot of fission gas release?

MR. BARTON: I'm sorry. I'm having a hard time hearing.

(Off mic comments.)

MR. AIKEN: This is Bill Aiken. We assumed 7 percent of the fission gases.

MEMBER POWERS: And that assumption was tutored by what experimental data?

MR. AIKEN: It's our design-basis, the previous was at 6 percent, and we increased it up to 7 percent.

MEMBER POWERS: Why didn't you increase it to 20 percent like observed in the experiments?

MR. KAI: I assume that you're referring to the Cabri test.

MEMBER POWERS: I was thinking more about NSRR, but okay.

MR. KAI: Right. Okay. Now, what we've done here, we have not -- as I said before, there are some industry issues out there that we have to decide, is that significant for SPU or not. They're not resolved yet in terms of industry setting limits, especially for high burn-up fuel where those types of failures can occur at really low calories-per-gram limit. So at the high rate of this, and determined that we believe that this is not -- that SPU really has no impact on resolution of that issue, and so we are using the current methodology for fuel failure, with the recognition that industry -

MEMBER POWERS: I mean it's just flat wrong. Right?

MR. KAI: Well -

MEMBER POWERS: That's inconsistent with reality.

MR. KAI: Well, if we're going to talk about reality, I think one thing you've got to keep in mind is that when we operate our plant at full power, the rods are essentially 100 percent withdrawn. There is no -- there's practically no rod -

MEMBER POWERS: That's not when the hazard comes about.

MR. KAI: Correct. The hazards are at low power, which, in terms of SPU is not really -- I mean, SPU itself then doesn't make any impact on the low power head. So that is how we evaluated whether we were to do something in terms of logic, and we have thought about it, but the real question is, is what kind of analyses do we do, what limits do we apply? None of which have been set, so when we convince ourselves that this was not an issue at the uprate power level versus the current power level, we decided to continue with the current methodology. And we would, obviously -

MEMBER POWERS: Couldn't you increase your inventory, your release fraction from 6 to 7 percent?

MEMBER ARMIJO: You mean, percent failed fuel.

MEMBER POWERS: No.

MR. KAI: Percent failed fuel, and that was stated -

MEMBER POWERS: Gas release fractions went from 6 to 7 percent. I mean, you're neither fish nor fowl here.

MR. AIKEN: Yes, sir. The bottom line is this is a mystery issue. There's rule making coming, we're following the issue, and we'll implement work with our fuel vendor, and implement the regulatory rule making as it evolves.

CHAIRMAN SIEBER: Is this a generic safety issue?

MR. AIKEN: Yes.

CHAIRMAN SIEBER: Anyone on the Staff, is this a generic safety issue? The research issue that's been identified for high burn-up fuel, and -

PARTICIPANT: There are minimum guidelines that are in place, but they're only interim, and that's in part why we didn't do anything at this time, because we were uncertain what the final rule is going to be.

MEMBER MAYNARD: It's a generic issue. I do not know if it's an official generic safety issue.

CHAIRMAN SIEBER: I don't know either.

MR. KAI: I don't believe it's a GSI, but it is a generic issue that is being addressed by the industry groups and the owner's group.

MEMBER POWERS: But, in fact, we know what the answer is, and have known now for several years.

Okay. I know what they've done.

MEMBER ARMIJO: Just back at page 22, and that's just a question, clarification. On this locked rotor event, you have a failed fuel limit 7 percent. I presume that's a different 7 percent than what Dana was just discussing. I don't know, but in the SPU and you've made it less than 7 percent, and your current is less than 6 percent as your design basis. And what is the mechanism of fuel failure that we're talking about here?

MR. KAI: Okay. This is a locked rotor in which we assume an instantaneous stop of flow of the

RCPs, so, essentially, for the short time period before reactor trip occurs, you essentially have three-quarters flow with 100 percent power generation. And that is going to cause you to go into DNB.

MEMBER ARMIJO: So this is a DNB -

MR. KAI: Yes. Correct. And, again, in order to calculate things like peak clad temperature, we make a very conservative assumption. We do not, obviously, credit heat transfer in the DNB mode, so that's been the result is us calculating these types of parameters. And it applies only for the few second duration prior to reactor trip, because once reactor trip is over, the event is over.

MEMBER ARMIJO: So you calculate if they reach DNB conditions, you define them as failed.

MR. KAI: Right.

MEMBER ABDEL-KHALIK: Okay. And one should not compare the current and SPU conditions, because they are just different methods? Otherwise, the results don't make sense.

MR. KAI: Okay.

PARTICIPANT: Yes, for the peak clad temperature, I think we changed to the PAD data. I think that was the main -

MR. KAI: The fuel performance there that we used, one of the - which is called PAD - we used Version 4, and that results in some improvement in these types of parameters. And so that's why you see that the current result is slightly better than the SPU results.

MEMBER BROWN: 250 degrees is just slightly?

MEMBER BANERJEE: That's puzzling. Why is that -- 1969 to 1718.

MEMBER ARMIJO: Yes.

MEMBER BANERJEE: I can't follow the logic. Can you repeat what happened there?

MR. KAI: This uses a code called PAD to calculate fuel performance. And it's gas pressure resumed, fuel temperatures is a function of kilowatts per foot, and they're going to a new version which results, in some transients, significant improvement, and others it doesn't make any difference.

MEMBER BANERJEE: Well, what changes the -- what is the physical reason? Is it gap conductance, or what is it? If you can get 250 degrees by just changing something -

MEMBER SHACK: It gives you the code.

MR. WANG: This is Guogiang Wang from Westinghouse, just a quick additional information. As

Mike and Sandy explained, we changed PAD from 3.4 to 4.0, which was issued -- Westinghouse was issued in 2000, July 2000. It is a new PAD model. Basically, the fuel temperature was improved. At the normal operating condition, the fuel -- both the average fuel temperature and the fuel surface temperature was about -- close to 100 degree Fahrenheit lower at the normal operating conditions. So in the VIPRE model, when we calculated the PCT, peak cladding temperature, we assumed from the beginning the ultra surface of the cladding gone to DNB, so it's film boiling. So given the fuel surface temperature from PAD 4.0 is about almost 100 degree Fahrenheit higher, and we specify a fuel to clad heat transfer coefficient of about 10,000 btu power per square feet.

MEMBER BANERJEE: What led to this reduction in fuel temperature between the two versions of PAD? The gap conductance? What changed it?

MR. WANG: That's the different model for calculating the fuel temperature. That's a different -- I'm not familiar with the PAD WCAP, so we need to get back to you. I don't know.

MEMBER BANERJEE: It wasn't the gap conductance. It was something in the fuel conductivity or something like that?

MR. WANG: Yes. I guess, but yes.

MEMBER BANERJEE: Integral lambda dictator or something?

MR. WANG: I'm not too sure. It would be in the WCAP.

MEMBER BANERJEE: All right. I think it's fine.

MEMBER ARMIJO: It's a model change.

MEMBER BANERJEE: Yes, it's a model change. And, presumably, it was justified with data.

CHAIRMAN SIEBER: If we could, I'd like to move on. We're very far behind, gentlemen.

MR. KAI: Okay. I don't know if you're thinking there is more that we need from ASTRUM.

MEMBER BANERJEE: I have a question on SB LOCA. Was there any -- first of all, these SB LOCAs, the temperatures are very low, so you looked at all the spectrum of break sizes and things like that. Right? What sort of break size is that happening on the current, is that the same 4-inch? Is it different?

MR. KAI: I'll ask Josh.

MR. AIKEN: While Josh is coming to the microphone, I do want to point out that we -- all of the methods we're discussing here are methods that have been reviewed and approved by NRC in various topical reports.

MEMBER BANERJEE: You're using -- yes, so it was a 4-inch cold leg break, the current one, as well?

MR. MILLER: I believe so, but I'm not 100 percent sure on that. There's a 3-4 inch range. It's very typical of a standard 412 plant, the results we're seeing here.

MEMBER BANERJEE: Let me ask you then, do you get into a refluxing mode at all with such a small break?

MR. MILLER: You do, but it's a very short limited time for this particular break size. And there could be some countercurrent flow limitations preventing that liquid from getting back into the core. It's very short-lived, I can tell you that.

MEMBER BANERJEE: Well, the question I'm really asking here is, if you check the -- you, obviously, have higher steam velocities because of this 7 percent uprate condition. Are you close to the flooding limit at all, due to the increased steam velocity at the steam generator tube inlet? Have you checked that?

MR. MILLER: The inlet plane or at the

tube themselves?

MEMBER BANERJEE: Tubes.

MR. MILLER: The flooding limit? The model shows some liquid hold up in the steam generators under these circumstances, and we have seen that. And I have not looked at the details of this analysis, but I'd be willing to bet if you went in there, you would see that for a relatively short period of time until the flooding mechanism breaks down, and the steam generators drain out. Typically, in the loop seal vents, and you get a strong velocity through the faulted cold leg steam generator. That kind of clears the situation up.

MEMBER BANERJEE: So NOTRUMP has an explicit flooding criteria?

MR. MILLER: Yes, it does.

MEMBER BANERJEE: Like Graham Wallis -

MR. MILLER: Well, the flux flow links representing the steam generator are based on a TRAC flow regime map. TRAC-M I believe it is.

MEMBER BANERJEE: I was hoping you'd just say the Wallis correlation.

(Laughter.)

MEMBER WALLIS: Then you could really criticize it.

(Laughter.)

MR. KAI: I'd like to move on.

MEMBER BANERJEE: Okay.

CHAIRMAN SIEBER: Please.

PARTICIPANT: Mr. Chairman, did you want us to conclude on our fuel in the safety analysis presentation at this time? Is there any additional topics you'd like us to address?

CHAIRMAN SIEBER: Well, I think we ought to get into the radiological results. But it seems to me, having to review all this, there's only a couple of important points. One of them is you use your ultimate source term that you applied for four years ago, and got. And when you applied for it, you applied for it at 6-1/2 percent SPU, so it almost covers the situation we're in right now. Most of the doses are far below the limits, except for small line break outside containment, which is on Slide 30. You may want to explain that to me. And, also, the thyroid dose, which you say is not applicable now.

MR. KAI: Okay.

CHAIRMAN SIEBER: And five whole body, which is not applicable. You can -- I think that covers the striking results.

MEMBER POWERS: The critical issue that

determines all those is what they use for a gap inventory. And I don't quite understand how they calculated their gap inventory.

CHAIRMAN SIEBER: Well, that's a good question to ask.

MEMBER ARMIJO: It probably was reduced when they made this code change to PAD, because it's a drop in the fuel temperature.

MR. KAI: Okay. Why don't we go over this? I think we can answer your questions. Let's start with the small break outside containment. This is one of the only accidents that was not converted to alternate source term when we did the alternate source term submittal. And this is why we have this dual column. The 30 thyroid and the five whole body corresponds to the standard method for calculating doses. At SPU we converted it to the alternate source term methodology.

As you can see, what the standard -- alternate source term is actually, gets an equivalent dose combining the thyroid and the whole body into a single dose. And the thyroid limit, as you can see we're pretty close to the thyroid limit. And that results in us being essentially at the 30 limit.

CHAIRMAN SIEBER: Well, the total

effective dose, the limit is 2.5, and you're at 2.5.

MR. KAI: Right.

CHAIRMAN SIEBER: So there's no margin there.

MR. KAI: Well, what we've done there is --this analysis is terminated by the operator initiating -- isolating the break, so what we tried to do is get the maximum time for the operator, so we've gone all the way out to -- exactly as to what we expect. What this is, is a line that does not automatically isolate, and there's small lines, like instrument lines, and so we just calculate what we can -- the maximum operator response time that we can live with, and still meet the limit. So that's the reason why it's right up in there. If you were to assume a primary time, and like I said, this is 100 percent driven by what you're going to -- how much you're going to give for the operator in terms of isolating the small line like the instrument line.

CHAIRMAN SIEBER: When you look at fuel handlinG actions, there's not much difference between your current situation and the SPU situation, and that's due to fuel exposure, I take it. Higher burn-up.

MR. AIKEN: This is Bill Aiken. Well,

with the new uprate, we were beyond the limits of the regulatory guidance for .183, the alternate source term. We went over the combined 54,000 and then the 6.3 kilowatts per foot guidance in the footnote, which says that the standard gap fractions that you're supposed to assume, you have to propose new gap fractions, and justify the new fractions.

We proposed to use the old Reg Guide 1.25 gap fractions, the 12 percent for the iodine, and to offset that, we had to incorporate this control room design modification to have the control room initiate into the filtered recirculation mode within 30 minutes to counteract the additional dose that was -

CHAIRMAN SIEBER: Do you have a pressurized control room -

MR. AIKEN: Yes.

CHAIRMAN SIEBER: Bottled.

MR. AIKEN: Bottled air. We do not credit that in our analyses, though. We do not credit the bottled air, so we almost take a penalty, if you will. Even though the bottled air system still exists, and it is pressurizing the control room one minute after the accident, we don't credit that pressurization.

CHAIRMAN SIEBER: Still used?

MR. AIKEN: Yes, it is.

CHAIRMAN SIEBER: So you just haven't abandoned it and replaced it.

MR. AIKEN: No. No, it is still being maintained and surveilled.

CHAIRMAN SIEBER: On radiological controls, why don't we take a couple of minutes for the PRA overview.

MEMBER POWERS: Oh, one question, on the iodine spiking, you use a factor of 500.

MR. AIKEN: For which accident?

MEMBER POWERS: Steam generator tube rupture.

MR. AIKEN: Correct.

MEMBER POWERS: Any qualms about that? That's what the Reg Guide tells you to do that, and that's how you did it. You didn't look at it.

MR. AIKEN: That's exactly right.

MEMBER POWERS: You didn't look at it. You don't have any experience with your plant on spiking.

MR. AIKEN: No. Just following the guidance.

MEMBER POWERS: Now there's a controversy around that, but I guess you're not responsible for that controversy.

MR. AIKEN: That's a good thing.

MR. KAI: Should I keep the PRA -

CHAIRMAN SIEBER: Well, let me tell you what I think I need to know. One of them is the comparison of CDF and LERF currently in the stretch power uprate, and the increase which is a table on Slide 33. And I would like a listing in human factor space, or human reliability space as to how the times have changed for required operation operator responses during the abnormal operating occurrences and the accident conditions. Can you give me those?

MR. AIKEN: Yes, sir. The short answer, I'll start with the short answer, is that none of them have changed. We did use a different analysis tool. We used the RELAP code instead of the MAAP code to reconfirm all of the operator action times.

MEMBER STETKAR: In the real world, the times are shorter, so in the real world the times did change.

CHAIRMAN SIEBER: Well, they have to, because -

MEMBER STETKAR: They are shorter. However you evaluated them before or today, in the real world, there's less time available.

CHAIRMAN SIEBER: What he's telling me is he isn't going to answer my question.

MEMBER STETKAR: That's right.

CHAIRMAN SIEBER: Clarify something, the human errors didn't change as a result of the potentially shorter times.

MR. AIKEN: No.

MEMBER STETKAR: Let me see if I can shortcut something, because I looked at what they did. You changed five numbers in the PRA. You increased arbitrarily the PORV challenge probability by 10 percent. That was a guess. You increased the plant-centered loss of off-site power frequency by 10 percent. That was a guess. You increased the general transient frequency by 10 percent. That was a guess. Those three numbers were increased by some thermal hydraulic plant stability justification.

CHAIRMAN SIEBER: Yes, I don't understand what justifies it. I think -

MR. AIKEN: The 10 percent number was just based on what DeNay had done. We were just following previous industry -

MEMBER STETKAR: The only operator action you changed was the operator error rate for bleed and feed cooling, which actually didn't change. You assumed it was 10 percent lower in the current PRA.

You just assumed it was 10 -- if it was 10 percent lower in the current PRA, the current PRA core damage frequency would be somewhat lowered. You didn't increase it for SPU, you decreased it for the current PRA. Is that correct?

MR. AIKEN: Yes. All of that is based on thermal hydraulic analyses.

MEMBER STETKAR: I didn't see any time window, so because you're using HCRORE, the time window is very critical for the operator error rate. So back to Jack's question, there is no time documented.

CHAIRMAN SIEBER: The only one I could find was the inject recirc.

MEMBER STETKAR: Hot leg recirc, that changed from 9 hours to 5 hours, but they didn't change that in the PRA.

CHAIRMAN SIEBER: Yes, because it changes -- before you could wait for shift changes.

MEMBER STETKAR: Now you can't.

CHAIRMAN SIEBER: Now you have to do it yourself.

MEMBER STETKAR: Right. The other numbers that were changed, and these are critical. I want to get this on the record, because we're short time. We can discuss it more if we have time, was that you decreased the off-site power recovery probability for a large number -- for all time windows in the current PRA. You decreased it by 10 percent. You did not change it for the SPU conditions. Again, there's no information about the time. You just said if the current time -- if the current recovery probability was less, your current core damage frequency would, in fact, be less than what you calculate now. And then you calculate a delta. This is kind of a backwards type of change in risk, because you've artificially reduced the current core damage frequency, artificially reduced the current large early release frequency, and then said well, now, here's the difference. Well, I could artificially reduce it to 10 to the minus 8, and it would show that it's a 6 times 10 to the minus 6 difference.

MR. AIKEN: Well, we have RELAP results.

MEMBER STETKAR: Well, what are the -- back to Jack's time. If you have all of those analyses, what are the differences in time? We'd like to see those tomorrow.

MR. WALLIS: Do the 10 percents have any justification whatsoever?

MR. AIKEN: No, other than that's what

DeNay used in their -

MR. WALLIS: So you might as well have guessed the change in the core damage frequency.

MEMBER STETKAR: Well, the 10 percent changes that they made results in a 6.5 percent percent, which is sort of the only thing you can argue about. A 6.5 percent change -

MEMBER BONACA: In core damage frequency.

MEMBER STETKAR: But from an artificially reduced current core damage frequency. It's kind of a -

MR. WALLIS: But it's all very artificial; you could have guessed 20 percent, and you got a 12 percent change in core damage frequency. It really is not an analysis at all, is it?

MEMBER STETKAR: It's kind of a sensitivity study, but to state that the core damage frequency increase is 4.0, E to the minus -- not 4.1, not 4.2, but 4.0 E to the minus 7 is specious.

MR. AIKEN: But you understand that we did do a relatively formal update of the model before we did the sensitivity study.

MEMBER STETKAR: I understand that you updated the model. It's just that in whatever I think we could read, there is no information to tell us what the differences might be, the real differences.

MEMBER POWERS: We have to get people to quit putting these bottom line numbers up, instead of putting up importance measures for us so we know what the critical components, and -

MEMBER STETKAR: Well, the only reason I wanted to bring this is they did make the point of saying that the most important parameters, at least that they looked at, were the operator action for bleed and feed cooling. That has a relatively high importance measure. In the details there is stuff in there.

PARTICIPANT: There was some justification for what we picked, not the 10 percent, but which things we picked to do the sensitivity -

MEMBER STETKAR: Yes. And the off-site power recovery time, the off-site power recovery probabilities is a function of time -- those are two that have relatively high -

MEMBER POWERS: Give them a gold star for pointing those things out, because lots of people give us these damned numbers that are meaningless.

MEMBER STETKAR: But, I mean, we've seen submittals where people actually do show you here's my time beforehand, here's my time after, and that's what the delta is, run through the model.

MR. KAI: We can get you that information pretty easily, I think.

MEMBER BONACA: But this code, RELAP-5, I mean, what did you do, you calculate success criteria, your -

MR. AIKEN: Yes, we did all the success criteria as part of the model.

MEMBER BONACA: Before the EPU.

MR. AIKEN: Pardon? No, that's what -- we only did that for the stretch uprate. That's why -

MEMBER BONACA: I don't understand the logic. I mean, they seem to have some kind of logic, so you did that for the power uprate.

MR. AIKEN: Yes.

MEMBER BONACA: And then?

MEMBER STETKAR: I'll give you a good example, Mario. As I understand it, they said the steam generator dry-out time currently is 37 minutes. Is that correct? I don't know what -- SPU conditions is 37 minutes. I found that in there. I don't know what the current steam generator dry-out time is under current conditions. Does anybody know what it is?

Now, apparently, in the PRA they used a 30-minute time window for operator actions to start auxiliary feedwater or something like that, so as long as 30 minutes is less than 37 minutes, they said well, their 30-minute assumption is justified. That's fine, but I'm not sure why -- how that translates into a change in the human error rate, because that 37 minutes is certainly less than what it is today.

MEMBER SHACK: Your PRA numbers are about a factor of 4 lower than when you submitted your license renewal application. Is that -

MR. KAI: Well, we've done a couple of major updates since then.

MEMBER SHACK: They just keep going down, huh?

MR. AIKEN: Well, a couple of things are happening. We're going through the process of incorporating the industry standard. We're updating the data on a more regular basis, incorporating better methods and models as we go. There's a lot more work being done in the PRA area now than there has been in the past, so that's been the net result, is that the initial risks have been shown to be conservative.

CHAIRMAN SIEBER: Any more questions? If not, I think that we'll take a short break for 5 minutes, but I would like after we return from the break to have NRR present their remarks on safety analysis for fuel, and safety analysis. We'll return at 5 minutes after 3.

(Whereupon, the proceedings went off the record at 2:57 p.m., and resumed at 3:03 p.m.)

CHAIRMAN SIEBER: Okay. We'll resume our meeting, and I'd like to call on NRR to do the fuel and safety analysis.

MR. PARKS: Good afternoon. My name is Benjamin Parks. I'm in the Reactor Systems Branch in NRR. I'm joined up here with John Lamb, our Project Manager, and Sam Miranda, also in Reactor Systems Branch. We worked together to review this power uprate.

As you can see from our review scope, we followed the guidance that was in RS-001, and our review focused on the topical areas covered by RS-001. I don't think that I need to run through the list. The last item on that list was Westinghouse methods. We reviewed an implementation of RETRAN and VIPRE, and we'll discuss a little bit about that, and answer some questions, if you may have some remaining.

Our review looked at, it says EPU evaluations, this is a stretch power uprate, we reviewed it to the EPU standard. We think RS-001 is a pretty powerful guidance document, and the licensee formatted the licensing report, so it was a natural fit, pretty thorough evaluation.

I guess for the fuel system design itself, about the only place in the licensing report, and I feel like this is worthy of pointing out, the licensee's contractor made the point that it's okay to, from a mechanical perspective, reinsert previously irradiated fuel assemblies of a different type other than RFA or RFA2. The Staff's evaluation, however, focused on what the nuclear and thermal hydraulic analyses were for, which was RFA and RFA 2 fuel. And as you notice from the licensee, the uprated core will be RFA2 fuel entirely, so that was the focus of our review. Okay?

What we observed in terms of the fuel and system design was a slight increase to the linear heat rate, and a slightly less peaked core design. Our safety evaluation tabulates an overview of the nuclear effects of the uprate and you see some of the reactivity coefficients change a little bit. I think we've already talked about that.

Basically, as is typical of a power uprate, licensing report and our review, we looked at a sort of reference core design that isn't necessarily what will be used at the power uprate, but it is an uprated core, and the cycle-specific, NRC-approved reload process, which is an NRC-approved Westinghouse process will confirm that they can live with the analyses that they set forth here, or will redo those analyses.

CHAIRMAN SIEBER: They may have to change that if they, during the process of refueling, find a damaged fuel assembly or something like that.

MR. PARKS: Correct.

CHAIRMAN SIEBER: And so that's why you do a separate reload safety analysis.

MR. PARKS: Right. You also want to make sure that if you changed any of your design parameters, or anything like that, or if something is implemented on a fit forward basis, you want to make sure that your safety analysis footprint, I guess your core design is going to fall within that.

So for the thermal hydraulic design, we're reviewing against GDC-10 and 12. We want to make sure that specified acceptable fuel design limits are not going to be exceeded as a result of anticipated operational occurrences, and we want to make sure that we have a stable core design.

For a Westinghouse reactor, a stable core design generally refers to a xenon transient. There are design features discussed in the licensing report reflected in our safety evaluation that explain trip features that sort of act to mitigate a xenon oscillation, should one occur. But, generally, a Westinghouse core is designed so that these types of oscillations where you're burning in and you're burning out xenon in various places is heavily dammed. Right.

And I'll just mention, I've heard some members before ask about Condition 1, 2, 3, 4. In this space, we are looking at Condition 1-4 ANSI scale events. In other words, Conditions 1 and 2 events are typically AOOs, anticipated operational occurrences, so we're looking for those. Typically, we're looking at violation to the departure from nuclear boiling ratio, or an acceptable pressurization result. And then 3 and 4 are more in accident space, either a limited amount of fuel damage, or an acceptable -

MR. WALLIS: So what is your acceptable DNBR?

MR. PARKS: The acceptable DNBR is going to be in the terms of this analysis, the design limit.

MR. WALLIS: So it's 1.2 then?

MR. PARKS: Yes, ish. The ish, right? Basically, I guess, my view on the acceptable DNBR here is, there is -- when we step into DNBR space in terms of fuel design limit, we're looking at the safety analysis limit, that upper 1.6, and then below that to the, I guess, the design limit. It's basically a basket safety margin. And as you've seen in some of the accident analyses where they encroach upon that 1.6 limit, they take a piece out of the basket, but it's a quantified amount of margin, so you're not just saying there's margin there so we can live with it. It's actually quantifiable, and we notice that it was quantified.

MR. WALLIS: Quantifiable in terms of say 1.5 being bigger than 1.2? Is that how you -

MR. PARKS: 1.5 being bigger than 1.2, and then accepting that we've, for a specific accident that has the .1 penalty, allocated that much margin. So I guess we used a bank account analogy, so that's in the balance sheet now. Okay? We're not just going to swipe the card and forget that we -

MR. WALLIS: This always puzzles me, because a 1.6 is a sort of arbitrary number picked by the licensee. They could have picked 1.5, and some other licensees pick other numbers.

MR. PARKS: It would leave them with less operational flexibility, and that's an agreement that's (a) sort of out of the scope of our review. I mean, we definitely consider it. We're more concerned about the safety limit DNBR, which is an enforceable quantity. But the point is, they are maintaining margin to it.

Obviously, for thermal hydraulic design, we're concerned about anticipated operational occurrences, and that begs the question of transient analyses and accident analyses. We went through this morning in a significant amount of detail the accidents and transients. I'll discuss, or Sam and I will discuss three specific points that were of interest to the Staff, and we had some interactions back and forth with the licensee about.

The first I'll tell you about is overpressure protection. I'll step through those slides. Sam will tell you about the inadvertent ECCS actuation, P-19 permissive. I apologize, our pictures aren't of the best quality, because we took them from the licensing report, but we do have pictures of the logic, if we need to talk about that more. And we'll talk about an interesting transient, rod withdrawal at power. If we need to move on in the interest of time, we'll be happy to do so. Overpressure protection.

CHAIRMAN SIEBER: Yes, make it as quickly as you can.

MR. PARKS: We're looking at Condition 2 Acceptance Criteria here, limiting the peak pressure. And we look at two trips, high pressurizer pressure, and OT-delta T overtemperature delta T. Typically, we look for accident analyses to credit the second trip, and this licensing report credits it first, but we confirmed through the RAI process that crediting either trip would be acceptable. And the result was that the peak pressure didn't exceed 2750.

MR. WALLIS: Is this psig or psia?

MR. MIRANDA: It's psia.

MR. WALLIS: psia?

MR. MIRANDA: psia.

MR. WALLIS: Strange thing to take absolute pressure.

CHAIRMAN SIEBER: The ASME code gives you psig, which gives you the steam time margin.

(Simultaneous speaking)

MR. MIRANDA: 2735 psig.

MR. WALLIS: That would be the ASME code.

MR. MIRANDA: Yes.

MR. WALLIS: Okay. Thank you.

MR. PARKS: Okay. So now I'll turn it over to Sam for the P-19 permissive.

MR. WALLIS: This is the vessel, which is in a containment which is not at atmospheric pressure. It seems a bit peculiar.

MR. MIRANDA: It's absolute pressure, and it's the -- what you get is the output of our analysis codes.

MR. WALLIS: That's right.

MR. MIRANDA: psia.

MR. WALLIS: But the ASME code has to be the difference between that and whatever surrounding pressure.

CHAIRMAN SIEBER: Four pounds and 2500.

MR. WALLIS: Well, I assume you know what you're doing. It would be nice if it were clearer.

MR. MIRANDA: Well, one of the unique features of this application is the licensee's implementation of a new permissive. What they're doing is adding a P-19 permissive that interlocks low pressurizer pressure with the cold load safety injection charging valves. And this relates to their desire to comply with the acceptance criterion for Condition 2 events, which stipulates that a Condition 2 event must not develop into a more serious incident.

And, typically, for an inadvertent ECCS actuation, this progression scenario consists of filling the pressurizer, opening the PORVs, which typically are not safety grade. And once the PORVs open with a ready solid pressurizer and they discharge water, then they're assumed not to reseat completely. So what we have then is a Condition 2 event, the inadvertent ECCS actuation developing into a more serious small break LOCA event at the top of the pressurizer.

Millstone's case is different in a couple of respects. First of all, they have this P-19 permissive, and they also have water-qualified PORVs. There is some history behind this acceptance criterion, too. In 2005, the NRC issued a regulatory issued summary, 2005-29, which reminded licensees that this criterion, this acceptance criterion that a Condition 2 event must not become a more serious event, is going to be something that the NRC examines in all license applications, license amendment applications. And knowing this, Millstone has acted proactively to put in this P-19 permissive, which it changes significantly their ECCS actuation scenario.

One more comment, since we talked about it this morning at length. There is a criterion, that's an informal criterion, not filling the pressurizer. Not filling the pressurizer for Condition 2 events is related to this. For example, in the loss of feedwater accident, where the reactor coolant swell that results from the loss of heat sink gradually fills the pressurizer, and continues to fill it until eventually the heat removal through the auxiliary feedwater system exceeds the decay heat generation rate. At that point, you reach your maximum pressurizer level, and, hopefully, that doesn't fill the pressurizer.

The objective in not filling the pressurizer is to eliminate the possibility of passing water through open PORVs. So not filling the pressurizer is a way to demonstrate that their Condition 2 event will not become a more serious event.

MR. WALLIS: Does the PORV now close?

MR. MIRANDA: The PORV if it release steam will close, if it relieves water we assume will not close.

MEMBER ABDEL-KHALIK: Even if it is qualified?

MR. MIRANDA: If it is qualified, we assume it will close. And Millstone takes credit for using the PORVs.

MEMBER ARMIJO: Well, that's some sort of unique design, or test. There's something that qualifies these particular valves?

CHAIRMAN SIEBER: Test.

MEMBER ARMIJO: Test.

MEMBER SIEBER: For valve. You have to look at the downstream piping, too, because that water slug's like a board.

MR. MIRANDA: Yes, that's half of it. The other half is qualifying the actuation circuitry to the safety grade standards. And Millstone is one of six plants that have water-qualified PORVs.

Millstone also has in its history an inadvertent actuation event occurring in 2005, and that event resulted in a water leak through the PORVs.

MEMBER MAYNARD: Did they close?

MR. MIRANDA: They closed. But afterwards they showed some leakage.

CHAIRMAN SIEBER: Usually, the stem bends a little bit.

MEMBER MAYNARD: Some leakage?

MR. MIRANDA: Some leakage. Yes.

MEMBER BROWN: Some or a lot, talking about GPE power, GPH, or -

MR. MIRANDA: I don't have the exact amount. I know they repaired one valve and replaced the other.

MR. WALLIS: I don't quite understand. A permissive is something where you let them take credit for something. Is that what a permissive is?

MR. MIRANDA: A permissive is -

CHAIRMAN SIEBER: It's an actuation signal that allows the trip signal to pass through.

MR. MIRANDA: Yes, it's an interlock.

MR. WALLIS: Physical change of some sort. What's it got to do with permission?

MR. MIRANDA: Well, you can -- it depends on what sign you take. You can either prevent something or permit it. You can say an interlock, which would be the negative, and the permissive would be the positive.

MEMBER MAYNARD: These conditions have to exist before that would come into effect, or -

MR. MIRANDA: It imposes a condition for something else to happen. In this case, you have to have low pressurizer pressure in order to -

MR. WALLIS: In some sort of logic it says if A, then B.

MR. MIRANDA: Yes. In this case, the permissive comes from the low pressurizer pressure reactor trip logic, in that you have two out of four low pressure bi-stables, which make up the low reactor coolant system pressure trip. It also generates the P-19 permissive. And that permissive is routed to the cold leg valves, which permit charging flow into the RCS.

MEMBER STETKAR: Sam, I forgot to ask this morning. Does Millstone have four cold leg injection valves or two?

MR. O'CONNOR: We have injections of four loops through two safety grade cold leg and actuation valves.

MEMBER STETKAR: Okay. A Train A valve, and a Train B valve?

MR. O'CONNOR: That's correct.

MEMBER STETKAR: Thanks.

MR. WALLIS: Well, they eject into two, each valves ejects into two loops?

MR. O'CONNOR: No, both valves provide flow to the four cold leg -

MR. WALLIS: Four.

MR. O'CONNOR: They're completely 100 percent.

MR. MIRANDA: Okay. So with this arrangement, where you have the permissive or interlock, however you want to call it, you don't have a situation now where a single fault will cause charging flow to enter the -

MR. WALLIS: It's a way to prevent inadvertent actuation of ECCS.

MEMBER SHACK: Well, inadvertent water going into the -- you can still get ECCS actuation.

MR. WALLIS: But the valves aren't open, so it doesn't happen.

MEMBER SHACK: All other ECCS functions would work.

MR. WALLIS: But ECCS doesn't get into the RCS.

MR. MIRANDA: That's true, but you do get charging flow to the reactor coolant pump seals.

MR. WALLIS: Yes. That's what we were talking about before.

MR. MIRANDA: Right. So what it amounts to is a very slow motion -

MR. WALLIS: Pressurizes very slowly.

MR. MIRANDA: Yes. It provides the operator a lot more time to act. And unlike a lot of other plants, what the operator needs to do in Millstone is to assure that the block valves to the PORVs are open so that the PORVs are available for function. In other plants that don't have this permissive, and don't have qualified PORVs, they need to end the ECCS flow.

MR. WALLIS: So this is something else that has to open in order for this to actuate?

MR. MIRANDA: So this diagram which was taken from the licensee's application shows, marked in the little balloon there, the new permissive. And you can see, it comes off the pressurizer low pressure bi-stables, and they make up the two out of four voting logic, which continue then to the reactor trip.

MR. WALLIS: So a permissive is a physical thing. Now, I thought a permissive was something that the NRC allowed, gave permission for, but it's nothing like that, at all. It's a functional, logical thing, which opens or closes things.

MR. MIRANDA: Yes.

MR. WALLIS: That makes it a lot clearer what it is.

MR. MIRANDA: Yes.

MR. WALLIS: It's a funny name to give it.

MR. MIRANDA: Yes. The Westinghouse RPS logic is full of permissives.

CHAIRMAN SIEBER: Yes, but they've been doing that for 50 years.

MR. MIRANDA: And this is where the

permissive goes. It goes into this ANT gate. You need that ANT condition to open the valves and permit the ECCS flow into the RCS.

MEMBER ARMIJO: Has any other plant used or have this permissive, or is this one-of-a-kind, first-of-a-kind?

MR. MIRANDA: Not that I know of. I think this is a first-of-a-kind.

MEMBER ARMIJO: Okay.

MR. WALLIS: And it doesn't induce some new event?

MR. MIRANDA: No.

MEMBER ARMIJO: No downsides, no -

CHAIRMAN SIEBER: Unless it fails, that's a new event.

MEMBER BONACA: Did they perform an evaluation of risk for the PRA, the configurations?

CHAIRMAN SIEBER: Okay.

MR. WALLIS: Well, there's a new mode of failure now, whether for the permissive doesn't permit, then the ECCS doesn't work.

(Simultaneous speaking)

MR. WALLIS: There is a new mode of failure at the ECC.

MR. MIRANDA: Well, it's kind of hard to

imagine a failure of a permissive when it comes from the two out of four voting logic.

MR. WALLIS: But it could still happen. It could still happen.

CHAIRMAN SIEBER: Yes, a card could burn out, but you've got two -- it meets the regulatory requirements, and defense-in-depth, and redundancy.

MR. MIRANDA: Failure to generate this permissive means also failure to have a reactor trip on low pressure.

MR. WALLIS: Which would not be very good.

MR. MIRANDA: Which should not be possible. It would not meet GDC, I think, 20.

MEMBER ARMIJO: I'm trying to understand this thing a little bit better. If there was no stretch power uprate, would the staff still look kindly on approving this permissive, or would you even get involved? Is this a good thing to do, even without SPU for this particular plant?

MR. MIRANDA: I believe it's a good thing to do for this plant, and other plants that have high pressure ECCS systems.

MEMBER ARMIJO: Improvement, independent of stretch power uprate.

MR. MIRANDA: Yes.

MR. PARKS: At this point, we're going to start talking about reactivity and power distribution anomalies, and I'd like to start before the rod withdrawal at power to follow-up on some discussion that you had about the Staff's acceptance of 200 calories per gram for the reactivity insertion accident.

I don't believe we have a generic safety issue that we pursued with this, but we are following it up. And I'd like to introduce Paul Clifford to discuss -

MR. WALLIS: How many calories per gram did you say?

MR. PARKS: 200. That's the Westinghouse acceptance criterion for the enthalpy addition.

MR. WALLIS: It just seems high, isn't it?

(Simultaneous speaking)

MEMBER POWERS: It just depends on whether your fuel has any burn-up or not. If it doesn't have any burn-up, it's not -- if it's new, it's okay. But if it has a little burn-up, then you've got a problem.

CHAIRMAN SIEBER: Let's move along.

MR. PARKS: Could you please summarize for us what we're looking forward to in terms of that accident and the acceptance criteria?

MR. CLIFFORD: As many of you are aware, Reg. Guide 177 has some deficiencies in it, and the staff has been -

MEMBER POWERS: Has for 25 years.

MR. CLIFFORD: Exactly. And the Staff has -- excuse me. Paul Clifford, NRR DSS. The Staff has been reluctant to -- I shouldn't say reluctant -- has been slow in revising Reg Guide 177.

I think it's important to realize that Westinghouse identified this deficiency really with the 280 calories per gram, which is the coolability limit. And they imposed an internal guidance of 200 calories per gram to preserve coolable geometry. And that, even now, we're looking at revising Reg Guide 177, and to date, the criteria would be 230 calories per gram, so the 200 calories per gram for coolability would remain conservative. Nowhere in the Westinghouse internal -

MEMBER POWERS: Be precise. It would remain conservative relative to 200, relative to reality it's wildly non-conservative.

MR. CLIFFORD: Well, we can't mix up PCMI failure with coolable geometry. The 200 calories per gram limit, the Westinghouse-determined limit, is not a PC -

MEMBER POWERS: It's not conservative for fission product release.

MR. CLIFFORD: Right. But the Westinghouse internal criteria of 200 calories per gram is not to prevent PCMI failure. It's not replacing the 100 calories per gram, or the 150 calories per gram, numbers thrown around like that. It's a maximum limit on coolable geometry, which is intended to prevent molten fuel from being dispersed into the coolant.

Now, the Staff has always relied upon the available margin in 2D methods to make up for the fact that we haven't introduced a PCMI-specific fuel failure criteria. And that's still our position today, but you need to note that Westinghouse still uses a very conservative method of using DNB as the point of failure for the cladding, and determining how many pins or how much RCS activity is available for release and for off-site doses. So the real question is, is DNB more or less conservative than what recent tests of Cabri or NSRR would tell us that PCMI failure would occur. And I believe that, up to reasonable burn-ups, mid- to high burn-ups, DNB would remain limiting relative to PCMI, and only high burn-up fuel that's heavily corroded and absorbed a lot of hydrogen would PCMI become more limiting. And you would have to ask yourself would that fuel be -- have the ummpf, the reactivity left in it to actually experience PCMI. And it may, depending on fuel management. But there is ample margin, as Westinghouse has stated, the 3D methods could clearly show less than 100 calories per gram delta.

MEMBER BONACA: That's an issue to figure out, I mean, if they used 3D neutronics with proper simulation with Doppler feedback and so and so forth, you would get much lower calories per gram deposition. And by leaving the criterion so high, then you're allowing methodologies which are obsolete. I mean, they used to use it 20 years ago, combination of point kinetics with the 3D axial shape, which is not 3D. I mean, so --

MR. CLIFFORD: That's correct. And by maintaining the DNB failure criteria, the dose calcs will maintain sufficient conservatism.

MEMBER POWERS: How do we know this? This is based on your gut feeling?

MR. CLIFFORD: The Staff currently relies upon Rule 0401 which is an operability assessment performed in 2003 to state based upon three-dimensional physics calculations that show the current fleet remains conservative relative to an investigation of the recent PCMI failures noted at NSR and Cabris.

MEMBER BONACA: It seems to me also that, I mean, physically DNB is a cladding issue, and the rod ejection, it's calorie deposition, and possible fuel damage, and burst, and so, I mean, I hear you, but -

MR. CLIFFORD: That is correct. We have issued interim guidance which we are currently imposing on new reactors, and we will be imposing that interim guidance once it's finalized over the next six to twelve months, and are going through the proper channels of backfit on current operating fleets.

MR. WALLIS: I'm trying to understand your argument. You're saying a lot of heat goes into this fuel, but because you don't have DNB, a lot of heat also comes out? Is that the argument?

MR. CLIFFORD: No.

MR. WALLIS: Therefore, it's okay?

MR. CLIFFORD: That's not what I'm saying.

MR. WALLIS: Well, what's the DNB argument got to do with so many calories per gram, and how are they linked together?

MR. CLIFFORD: They're using conservative test methods, and assuming that if the clad goes into DNB for even a split second, that that cladding fails, and you release all of the gap activity for your dose calculation. That's a conservative assumption.

Now, if you were to look at PCMI failure -

MR. WALLIS: Do you know how to calculate DNB in a transient like this?

MR. CLIFFORD: It's very conservative to assume that the steady-state DNBR methods are valid for the speed of this transient.

MR. WALLIS: All the correlations are based on a steady-state.

MR. CLIFFORD: That is correct.

MR. WALLIS: And you're saying it's conservative with regard to a transient, because those things don't have time to get up to steady-state or something, or what? Is this a gut feeling, or is it justifiable analytically?

MR. CLIFFORD: There's several -

MEMBER BONACA: What they've been using forever, all the vendors, is to be a point kinetics calculation, and then a static calculation of axial values, and then they combine the two of them, and they call it a 3D calculation. Actually, it's not.

It's a point kinetics calculation of some kind, and you do a DNB calculation to look at the thermal hydraulics of that. But you don't get the benefit of Doppler feedback, LOCA that you would have if you had a 3D simulation, too. And also, analytical methods. All the vendors have had it for a number of years, but it's expensive to use, and also you would have to change the regulation, I guess, because -- and so what happened is that nobody is using advanced methods. They're still using this hybrid static calculation where you don't get sufficient feedback to lower your peaking factor, so, therefore you have 200 calories per gram deposition. In reality, with the 3D calculation they show value at the order of 60-70 calories per gram peak. Until you tighten up the criteria, that is the indication that you have, nobody is going to use more advanced methods.

MR. CLIFFORD: That's true, but to get back to the question that Graham Wallis had. The experimental results show -- well, confirm that the 170 calories per gram radial average enthalpy is a point at which DNB failure occurs. And when you use our methods to calculate DNBR, I'm confident that it would be below 170 calories per gram.

MR. PARKS: Okay. So having heard on that issue, I want to share with you the rod withdrawal at power accident, or the transient analysis. At power is evaluated in the licening report. It considers, I believe, 1060 and 100 percent power, various rates of reactivity insertion. The results were acceptable, but there is a reference in the licensing report to a generic disposition of the potential for overpressurization associated with this transient. Basically, the high pressurizer pressure trip may not, if this accident is initiated at a low-power level, terminate the accident before there's overpressurization, so there's a generic evaluation that shows that this is not the case. We questioned that in the next slide.

The generic study was docketed for another licensee in concert with a different license amendment request, and it was performed as described here. Basically, it's for a four-loop Westinghouse plant to demonstrate that the positive flux rate trip will help terminate this transient before there's a pressurization problem. And the four-loop study, I was concerned, didn't cover Westinghouse's power level, so I asked about it.

I worked with Project Management and the licensee to conduct an audit, and we had some sensitivity studies that we reviewed of this accident. And, basically, what the licensee contended initially, and demonstrated with some sensitivity studies on this generic study, was that the potential for energy addition associated with the core uprate is mitigated by water-filled loop seals which are assumed in the accident and do not exist at the plant, water-filled loop seals on the pressurizer safety valve discharge piping. So because Millstone doesn't have those, there's no purge delay on its pressure relief, and so the safety valves relieve the pressure, and acceptably mitigate the consequences of the event.

MR. WALLIS: These are loop seals on the pressurizer discharge piping?

MR. PARKS: The safety valve discharge piping.

MR. WALLIS: You're just looking at the hydrostatic head in the loop seal?

MR. PARKS: Right. And there's -

MR. WALLIS: It's a rather small -

MR. PARKS: About a second and a half purge delay. And what you can see from the sensitivity studies -

MR. WALLIS: It's very small compared with the -

MR. PARKS: It is, but it's approaching 2700 psia, and so, in that case -

MR. WALLIS: Loop seal is worth what, 10 psi or something? What is it worth?

MR. PARKS: They contributed something like 20 pounds, and so eliminating -- basically, the energy addition and the pressurization associated with the increase in power level I think added about 20 to the peak pressure, and then eliminating the loop seals took down 20. The reason I was concerned about is because we're so close to 2750. Okay.

All in all, the results came out okay. We demonstrated and got the necessary information about Millstone 3 to show that we're reasonably assured that the positive flux rate trip will terminate this event, and the pressurizer safety valves will adequately mitigate the transient. In actuality, obviously, the PORVs will take care of this, so we covered that.

Okay. The next thing we'll talk about is LOCA. The licensee evaluated large breaks using ASTRUM. ASTRUM is the second, I guess, generation of a Westinghouse best-estimate LOCA analysis method. We previously, before this Automated Statistic Treatment of Uncertainties Method, before we approved that, we had approved what's called the Code Qualification

Document Methodology. So, basically, what we're talking about is a statistical process, and that's based on WCOVERT TRAC, both methods were based on WCOVERT TRAC. ASTRUM is 124 cases convoluted with uncertainties associated with the plant parameters. There is no change to the small break evaluations.

MEMBER POWERS: One hundred and twenty-four samples in a minor correlating analysis, and build the distribution, depends, of course, on what my uncertain parameters are. I have relatively broad uncertainty bounds on each of the quantiles. How do they pick within that quantile range what number to use?

MR. PARKS: Now we're stepping from the plant-specific application of ASTRUM to its original basis for approval, so I researched it, Dr. Powers. I'm not as familiar with that as I am with Millstone. What I understand is the documentation that we have, and that we reviewed, considered -- now I looked at the introduction document. I mean, this is volumes and volumes of information. It's a huge amount of paper. But the introduction, in my opinion, concisely summarizes the various parameters against which uncertainty is considered, and are convoluted. And my opinion was, and you're just confirming that a statistical method is okay. I wasn't re-reviewing the method. There was adequate information contained in the submittal for the Staff to review, and its contractors to review all of the statistics and conclude that 124 cases agree that that was acceptable.

MEMBER POWERS: It's not a bad number to run, but in the end, you want a number to compare against some criterion. And usually you pick -- what you select, a mean value, a 95 percentile, or something like that, is kind of up between you and God, whatever gets decided. I know of no analytic way to make that choice. But when you go to 124, it's not a huge number of analyses, not a bad number, not a huge number. And, typically, if I go to a quantile, say I decide I want to use the 95 percentile, I will find that I have a fairly substantial range that corresponds -- that I know the 95th percentile lies within that range. I just wondered, how do you pick a number out of that range?

MEMBER STETKAR: What you're saying is in the standard Monte Carlo analysis, you look at convergence of the mean. You run enough samples, and then you have some confidence that -

MEMBER POWERS: The distribution, and you can only determine that distribution to a confidence level, because it's not an infinite number of samples. And so you set up the distribution, and you say well, okay, I'm going to take the 90th percentile on my quantiles, and there's a range there. So what number do I pick? I mean, I'm just curious.

MR. PARKS: Well, when I approved, or recommended the approval of the implementation, I'm working from the fact that the ASTRUM method has already been reviewed and approved, so I didn't revisit the adequacy of the selection of 124.

MEMBER POWERS: I don't argue with that. I mean -

MR. PARKS: So I guess where I'm headed with the question is, I think it may be a question that's better answered by the folks at Westinghouse, because it's their method, and it's under their control.

MEMBER POWERS: I'm desperately hoping they charge forward here.

MR. WALLIS: I don't think there's any range there. I think you just assert that 1781, you have 95 percent confidence that that 1781 lies within the 95th percentile. That's all you can say. With this value you get, that's what you say. Some other day you may run the codes 124 times and get 1722 or something. You have the same argument you can say then. I don't see any range of anything, though.

MEMBER POWERS: You can say that 1781 lies within the 95^{th} percentile. I will believe that. But so might 1855 lie within the 95^{th} percentile. I mean, the 95^{th} percentiles do get pretty broad.

MR. WALLIS: Don't say anything about that.

MEMBER POWERS: Well, then I'm not very well informed here.

MR. WALLIS: No, you could be better informed. Sure.

(Laughter.)

MR. WALLIS: There's more information -

MEMBER POWERS: That's true almost throughout my life, Graham, that I could have been better informed.

MR. WALLIS: For information, but if you ask a specific statistical question, you get a specific answer. If you ask another question, you can get different answers, depending on what question you ask.

MEMBER POWERS: The question I'm really asking, Graham, is what is the temperature that I'm going to get in the event of one of these deleterious accidents.

MR. WALLIS: Well, you could ask what the mean is.

MEMBER POWERS: And if I want to be relatively conservative, I really want to know what the upper bound on that 95th range quantile is.

MR. WALLIS: You won't get enough of that, unless you run an infinite number of runs.

MEMBER POWERS: Well, I'm willing to say okay, give me the 95th percentile at 95 -

MR. WALLIS: You can say give me the 99th. You can ask for more and more.

MR. PARKS: I believe the point of the method is through 124 cases, there's a reasonable degree of confidence acceptable to the staff that we've identified a number under which -

MEMBER POWERS: I will accept that.

MR. PARKS: Okay.

MEMBER POWERS: What I'm asking is now, you, the staff, have looked at this methodology, so with 124 samples. I sampled, I know I have to a confidence level of about 95 percentile, I have sampled about 99 percent of the parameter space. Okay. And that's what the number corresponds to.

Okay. You have a distribution. Now you need a number. You need a number to compare against 2200 degrees Fahrenheit. Okay? And you say, I take the 95^{th} percentile number, but there's a range up here, and I'm asking how big is that range?

MR. WALLIS: Well, I don't think Westinghouse reports all of the data of the 124. If they did, you could start to do that. They just report the biggest number. Isn't that what they do?

MR. PARKS: Right.

MR. WALLIS: They don't say that the lowest number was 1200 or something, do they? They don't give you any other information. They just follow the rule.

MR. PARKS: That's how the method is employed. That's how we approved it.

MR. WALLIS: That's how it works. That's right.

CHAIRMAN SIEBER: It seems to me like we're not really getting anywhere with this discussion. I'd like to move on.

MR. PARKS: Okay. So the results of the analysis, you've already seen. This small break came in at 1193, and that was a four-inch break, I confirmed, pretty significant margin-ing on cladding oxidation and on core life.

MEMBER ABDEL-KHALIK: Does the 3-1/2 percent clad oxidation number include pre-accident oxidation?

MR. PARKS: Does it include pre -- I believe that it does, or that the limiting scenario is chosen at a point -- okay.

MS. ANTOINE: My name is Stephanie Antoine. I'm from Westinghouse, on the best-estimate large break LOCA. The 3-1/2 percent does not include the pre-transient oxidation. The ASTRUM methodology was approved without the pre-oxidation.

MEMBER ABDEL-KHALIK: Now, the 17 percent acceptance criterion does include pre-transient oxidation.

MS. ANTOINE: That is not in the statement that we, I believe, have in our analysis. The way that it was approved by the NRC was that to meet the 17 percent, we did not need to include pre-transient oxidation.

MEMBER SHACK: It does, but let's go on.

MR. PARKS: All right. Next slide, please. Westinghouse is implementing VIPRE and RETRAN to replace THINC-IV and LOFTRAN for the transient analysis methodology. In some cases, the LOFTRAN code is maintained, particularly for the steam generator tube rupture, a modified version is used to credit operator actions. I say here transients for the use of WRB2 is restricted, but I believe that that's a coincidence, not a reason to -

MR. WALLIS: What am I supposed to conclude from this? Is the VIPRE-RETRAN supposedly better or something? What do I conclude from this information?

MR. PARKS: RETRAN is a method that's, I know, based on LOFTRAN. It's the more current Westinghouse accident analysis method.

MR. WALLIS: Is the RETRAN 3D? Is that what this is?

MR. PARKS: No.

MR. WALLIS: The old RETRAN.

MEMBER POWERS: Yes, it's not nearly as good as the one that you have used.

MR. HEUGLE: Yes. This is Dave Heugle, Westinghouse. It's the old RETRAN-02.

MR. WALLIS: It's an old RETRAN.

MR. HEUGLE: Yes. It's not the RETRAN 3D.

MR. WALLIS: But what should someone reading this conclude? They've changed the code. Did they change it in order to get some advantage for themselves? Why did they do? What am I supposed to conclude from this?

MR. HEUGLE: The standard practice that we see when there's an uprate, Westinghouse typically implements their new -

MR. WALLIS: Because they get better numbers, or is it because it's a more reliable code, or what?

MR. MIRANDA: I believe that when we reviewed RETRAN, we compared the results to those obtained with LOFTRAN, observed that they were largely consistent, so it wouldn't be expected that you'd see significantly different results using one or the other.

MR. WALLIS: But you haven't run anything yourself, so you have no idea how good these codes are compared with something else?

MR. HEUGLE: This is Dave Heugle from Westinghouse. The reason we went with RETRAN was to align ourselves more closely with the utilities, and to also try and set ourselves up to allow us to take advantage of some improved methodologies down the road. But as a first step, what we did in the RETRAN submittal was to use the same methodology as we applied for LOFTRAN. And, as was stated, we compared those results to our LOFTRAN results, and for all the transients we showed that you get very similar results.

In addition, at the Ganay Beaver Valley extended power uprating, we also provided results that showed for actual plant transients, that the RETRAN model very closely matched for a number of different primary and secondary site conditions what we actually got from plant data.

CHAIRMAN SIEBER: Thank you. Why don't we do the summary now?

MR. PARKS: So we reviewed the transients and accident analyses that demonstrated acceptable results at the uprated conditions, confirmed that the fuel design remains acceptable to support the uprate, and the methods have been implemented acceptably. We reviewed conditions, limitations on the methods, and the technical basis underlying those conditions and limitations to make sure that the licensee was in compliance with those, and the technical basis.

CHAIRMAN SIEBER: Any questions?

MR. WALLIS: When you listened to the questions ACRS had this morning, did the same questions occur to you, or are we more critical than you are of this? You have given a pretty sure review here. Right? Did the same sort of questions get asked by you as they're being asked by, say, my colleagues here?

MR. PARKS: We spent some time -- we. I just promoted myself to an ACRS member, I apologize. The Committee seemed to discuss, heavily, comparison between RETRAN and LOFTRAN. Okay? And I asked a lot of those questions as draft RAIs, and ultimately removed them, having accepted the fact that as I dug into the methods themselves and noticing that the results from the various methods are largely consistent, that the implementation of the new analysis method, also given the fact that it's NRC-approved.

MR. WALLIS: But we asked questions, such as why is this number so much bigger than that number? Why is 1875 so much bigger than 1543, or something? Did you do that sort of thing, too? Did you get satisfactory answers?

MR. MIRANDA: I know I did. I asked questions like that, too. That's the first question you might ask when seeing the results. And you've got to know how it breaks down. And I have asked a couple of questions like that, and I've received plausible responses.

MR. WALLIS: I would think, you've got a long time to do this. You have a year to do it. We have -

MR. PARKS: No, we didn't have a year to do this.

MR. WALLIS: But, presumably, you asked many more questions than we do. I just want to make sure that you cover at least the kind of range that we cover, and probably many more. That's it.

MR. PARKS: We tried, I guess, to give you a sampling of things that we interacted with the licensee. I mean, I saw the rod withdrawal at power accident, for instance, and I didn't think there was quite enough information there, so I wanted to see more about that. So the answer to your question is yes, we asked the same types of questions. When we saw stuff that changed significantly, we asked about that. I asked about reactivity insertion rates.

MR. WALLIS: You don't feel that it's not an embarrassment that you had not asked the question when we asked it.

MR. PARKS: I'm sorry. I didn't hear the first part.

MR. WALLIS: You didn't feel any kind of embarrassment that you had not asked the question when we asked it. You heard our questions, you didn't say gee whiz, I wish I'd asked that. You didn't have any kind of that reaction.

MEMBER POWERS: Did you ever feel God, I'm glad I didn't ask such a stupid question?

(Laughter.)

MR. PARKS: Dr. Powers, I can't answer that question.

(Laughter.)

MEMBER MAYNARD: It's probably best to move on.

CHAIRMAN SIEBER: With that remark, I think that I appreciate your presentation. And I have a suggestion for the rest of the day. I've read through basically all the elements in the SER, and the application. And in my opinion, the electrical section is pretty simple, and I don't think that we need to review that. On the other hand, if any member objects, I'd like to know about that. And the flow-accelerated corrosion is also another area that is pretty standard in the industry. They're using standard methods and achieving the same results, so I suggest we accept -

MR. POWERS: When you say you get the standard results, that includes thinks like Surry?

CHAIRMAN SIEBER: No, Surry didn't use CHEKWORKS. On the other hand, containment analysis has some interesting features to it, particularly because of the sub-atmospheric containment. They are not asking for credit for containment pressure, which is a good thing. On the other hand, I think it would be good if Dominion would present their containment analysis, and the staff can follow-up with their analysis of Dominion's application. So if there are no objections, I'd like to change the agenda to do that.

MEMBER STETKAR: A minor monkey wrench. I had a couple of questions about their EQ of electrical stuff, in the main steam safety valve building.

CHAIRMAN SIEBER: Okay.

MEMBER STETKAR: That's the only -

CHAIRMAN SIEBER: You're talking about -

MEMBER STETKAR: The high temperature stuff. And I think I can get it resolved, I hope I can get it resolved real quickly. I don't want to make a big deal about -

CHAIRMAN SIEBER: Why don't you ask the question now and we'll see.

MEMBER STETKAR: I'll do that. Are the -- and I've seen them called -- are the main steam atmospheric relief valves, or I've seen them called atmospheric dump valves, called both things, at Millstone, are they safety-related equipment, or not?

MR. RUSSELL: Paul Russell, Operations. The atmospheric steam relief valves?

MEMBER STETKAR: Yes.

MR. RUSSELL: Our steam generator PORVs, if you will, they are air-operated, and because they're air-operated, we don't take credit for them.

MEMBER STETKAR: Okay. So they're not safety-related.

MR. RUSSELL: That's correct.

MEMBER STETKAR: So, therefore, their operators are not qualified for the steam environment. Is that correct?

MR. RUSSELL: They do get -- if we have a main steam line isolation, they do get isolated.

MEMBER STETKAR: No, no, no, no. I'm asking, if you had a steam line break in -- I'm assuming the operators for the steam line -- main steam PORVs are located in the main steam valve building. Is that correct?

MR. RUSSELL: Yes.

MEMBER STETKAR: Yes. Okay. So if you have a steam line break in that building, the operators for those valves are not qualified to operate in that environment. Is that correct? I'm not talking about the isolation valves, the motor-operated. I'm talking about the atmospheric reliefs, themselves.

CHAIRMAN SIEBER: If the line is broken, why do you need atmospheric relief? I mean, it's all coming out anyway.

MEMBER STETKAR: If you have a break in that building, these are upstream from the MSIVs. Is that correct?

CHAIRMAN SIEBER: They're between the MSIVs.

MEMBER STETKAR: They're between the steam generator and the MSIVs. Right?

CHAIRMAN SIEBER: Yes.

MEMBER STETKAR: So if I have a steam line break in the main steam valve building, and the MSIVs go closed, I can still use the atmospheric relief valve on that steam generator. Right?

MR. RUSSELL: That is correct. Yes.

MEMBER STETKAR: Okay. But the operators are not qualified for that steam environment. Is that right?

MR. RUSSELL: That's a question I have to defer to design.

MEMBER STETKAR: This is kind of -- I want to find out, this is sort of PRA-related issue, because since the steam temperatures are so much higher in that building right now, I want to find out if the PRA takes credit for using the atmospheric relief valve on the line with -- on a line that's now isolated for secondary heat relief, because I didn't notice that the -- it's a convoluted environment, it's a convoluted scenario. You get a steam line break, MSIVs close successfully; however, you -- because the break was in that -- is the main steam valve building -- I don't know anything about the plant. Do you have four separate enclosures, or do all the steam lines come through a single -

MR. RUSSELL: They basically come through one single -

MEMBER STETKAR: Single.

MR. RUSSELL: So there's not separate enclosures for each -

MEMBER STETKAR: So it's a single. You're talking about a single volume.

MR. RUSSELL: Yes.

MEMBER STETKAR: Okay. So what would happen is, you'd have a steam line break. You'd relieve a bunch of steam into there for some period of time. The MSIVs would go closed. Temperature would be up. The question is, does the PRA then take credit for use of the atmospheric relief valves to remove secondary heat for active cool-down during one of those events?

MR. O'CONNOR: Mike O'Connor. I think I can help you with that. The atmospheric relief valves are air-operated. We don't credit those components, because the air system is not safety-grade.

MEMBER STETKAR: In the safety analysis, I asked does the PRA take credit for it.

MR. O'CONNOR: So there's no operator action to use those for a cool down. There are other valves that are used.

MEMBER STETKAR: Okay.

MR. O'CONNOR: Did that answer your question?

MEMBER STETKAR: Yes. I mean, as long as the PRA doesn't take credit for the atmospheric reliefs.

MR. RUSSELL: I'm pretty sure it doesn't, but I'll confirm that when I --

MEMBER STETKAR: After a steam line break, I only care about the steam line, or a feed line break in the same area.

MR. KAI: Millstone 2 has two dump valves per generator, so there's actually eight valves, one is air-operated, and one is motor-operated. There are two totally separate valves on each generator.

MEMBER STETKAR: But all the operators are in this -

MR. KAI: Right. One is air-operated, one is motor-operated.

MEMBER STETKAR: Well, if you're going to back up and get to the motor-operated ones, then I'll ask you are the motor operators qualified to operate in 562 degrees -- I don't know. The atmospheric relief isolation valves and the bypass isolation valves were discussed to some extent. In fact, those are the ones that you're specifically insulating because you couldn't get them qualified to operate in that environment, so that they would close to isolate a break in the relief valve line. Recognizing that's limited to a three-inch break.

I'm worried about the operability of the relief function after a steam line break, or a feed line break, with successful -- subsequent successful main steam isolation, because that type of function was not discussed in any of the EQ discussion for that building. Jack, thanks. That's enough.

CHAIRMAN SIEBER: Okay.

MR. WALLIS: You didn't get an answer.

MEMBER STETKAR: That's -

CHAIRMAN SIEBER: Let's see if we can finish up on the containment analysis.

MR. COLLIER: Mike Collier back again to talk about containment analysis. To expedite our discussions, since I think you want to go directly to the results. I think that what I would recommend is that we start with Slide 8, unless you want to go and discuss the ones before. We'll start with 8, that gives the actual results. And I can talk in terms of what the initial conditions are, and what our results are, or would you -

MR. WALLIS: And they're all using different codes than you had before.

MR. COLLIER: Okay. We could start with 6. Six is the changes in methodology. Now, again -

MR. WALLIS: I have no idea what that means.

MR. COLLIER: Okay.

MR. WALLIS: That doesn't -- I don't think it matters, but I just -- you've changed the code. I need some assurance that it hasn't -- the numbers haven't changed because you've changed the code, they've changed, giving a fair representation of what's the effect of the uprate. That's all. How do I get that assurance?

MR. COLLIER: Well, in this case we benchmarked -- the work course code is GOTHIC.

MR. WALLIS: You benchmarked the new code against the old conditions?

MR. COLLIER: Yes, exactly. Here is the results. With the current analysis, assumptions we reproduced with GOTHIC to make sure that we got exactly the same answers.

MR. WALLIS: Okay. Thank you.

MEMBER BROWN: That's one of the circumstances where you benchmarked your new one against your results from the old code in the old plant.

MR. COLLIER: Yes. Correct.

MEMBER BROWN: The current plant.

MR. COLLIER: The current plant, correct.

MEMBER BROWN: With assumptions, et cetera. Which you didn't do with the reactor design-type codes.

MR. COLLIER: Correct.

MEMBER BROWN: Okay.

MR. COLLIER: And we are using the internal -- it was approved by the NRC, so we have done that benchmarking to make sure that GOTHIC will produce the same results as the current containment tool.

Slide 8 is the initial conditions. What we tried to do here is expand the range of initial conditions that we're going to assume for some additional operational flexibility, so we will look at, as you can see, a wider range of parameters. We used both ends, whichever is conservative, either the low end or high end, and that's the point of Slide 8.

Slide 9 gives the actual results, compares current to SPU for LOCA and steam line break.

MR. WALLIS: This containment liner temperature is an average temperature.

MR. COLLIER: Correct.

MR. WALLIS: But in reality, a big LOCA produces a jet, which could impinge on the liner, so there could be local places where the temperature of the liner is 500 degrees. Pressure is pretty uniform, but the temperature is certainly not. I mean, there's a jet which impinges on the liner. Is this considered at all there?

MR. COLLIER: No, we do not -

MR. WALLIS: Because there have been incidents where jets have impinged on liners, and the whole liner has buckled. This happens with a water hammer accident. It has happened. I just wonder why average temperature is an acceptable criterion. Maybe the staff has some comment on that. Why is average temperature an acceptable criterion for containment liner, when some spots could be much hotter?

MR. CARICONE: Hi, my name is Albert Caricone, working in -- actually, nuclear safety analysis. I was heavily involved in doing containment analysis. We pretty much followed the same methodology that was used before. It's a pretty standard limit, that you encapsulate in the liner temperature -

MR. WALLIS: So maybe the question is why is the average acceptable to the Staff?

CHAIRMAN SIEBER: Anyone here from the Staff to answer that?

MR. LOBEL: This is Richard Lobel from the Containment Systems Branch in NRR. The number isn't meant to be a maximum in the sense of the jet impinging on the liner. It's meant to compare with the criterion for a structural number for the liner maximum allowed temperature. And it's calculated in a conservative way for heat transfer from the atmosphere to the liner. The heat transfer coefficients are increased, and you make other assumptions that maximize the temperature on the surface of the liner.

MR. WALLIS: On the average.

MR. LOBEL: Yes.

MR. WALLIS: But if a local region gets much hotter, you'll get some -- you could get buckling of the liner.

MR. LOBEL: But it's really a one-dimensional type calculation.

MR. WALLIS: Yes, it is. I know.

CHAIRMAN SIEBER: Well, you end up blowing the liner -

MR. WALLIS: Why is that good enough?

MR. CARICONE: The other thing I was going to point out, that all the loop valves are in the steam generator valve cubic, our steam generator cubicles, so really they're not exposed to the containment liner, per se. You can always postulate the break that would, I guess, a jet that it would impinge on a liner, but the majority of the piping, RCS loop piping is pretty much enclosed.

MR. WALLIS: A large break LOCA could go

a long way.

MR. CARICONE: Correct. What I'm saying, that the lower levels of the containment are very much comparmentalized, I guess. I'm having a hard time pronouncing it.

MR. WALLIS: So there's something in the way?

MR. CARICONE: Correct. There are actually --

MR. WALLIS: Is there always something in the way?

MR. CARICONE: No, you can always postulate -- well, let's see. You have the reactor vessel, the hot legs come out. They're all inside the shield wall area, and then you -

MR. WALLIS: The shield wall helps you a lot.

MR. CARICONE: Right. And then you have the steam generator.

MR. WALLIS: That helps you, except you blow the insulation off it, and --

MR. LOBEL: Correct. And the steam generator compartment is a concrete structure.

MR. WALLIS: It's always been curious to me why the average temperature was acceptable as a criterion. I'm not sure I'm going to get an answer.

MEMBER MAYNARD: Well, typically, for the late model Westinghouse plants, which you guys are, all of the RCS piping is contained down below, into the bioshield area. And there really isn't any direct path of something to go to the containment. You could have steam in your secondary side -

MR. WALLIS: But that's different.

MEMBER MAYNARD: On the RCS side, that's typically all down within the bioshield where there's extra brick concrete wall between that -

MR. WALLIS: So from a LOCA you would never get a direct impingement on the containment wall.

MR. CARICONE: That is a true statement.

MR. WALLIS: That's a true statement?

MR. CARICONE: Right.

MR. WALLIS: But you might for a steam line break, or something?

MR. CARICONE: Steam line break, obviously, the upper portion of the steam generators are exposed to containment. There is a possibility that you might have impingements.

MR. WALLIS: What's the purpose of this temperature limit?

MR. LOBEL: There's a structural criterion, a temperature limit on the temperature of the liner. And this is just to show that for the LOCA, the steam line break, that you stay below that temperature.

MR. WALLIS: If you exceed it, what happens?

MR. LOBEL: If I were doing the review and I saw that the temperature -- well, first of all -

MR. WALLIS: What happens, if you get over 280 degrees what happens?

MR. LOBEL: Well, I imagine there's -

MR. WALLIS: Blows up or something?

MR. LOBEL: I imagine there's a lot of margin, but if I were the reviewer and I saw that situation --

MR. WALLIS: I think if it heats up uniformly, it just pushes against the concrete, so that's not a bad thing.

MR. LOBEL: Yes.

MR. WALLIS: What's the mode of failure you're worried about?

MR. LOBEL: I, as the containment reviewer, would go to the structural people, and I would say that I'm very close or over this limit.

MR. WALLIS: Well, I as an IE person would say, if it's uniformly heated, I wouldn't worry very much. If I got a really hot region of 20 feet diameter, and if the rest is cold, and that's 500 degrees, I might worry about what will happen.

MR. CARICONE: I believe -- you might want to correct me if I'm wrong, but I believe the issue is liner separation.

MR. WALLIS: Yes. That's what happens when it -- it does separate.

MR. CARICONE: Correct. I think, as you know, the liner is obviously attached -- when they're pouring the concrete -

MR. WALLIS: It's fitted to the concrete.

MR. CARICONE: Correct.

MR. WALLIS: AS long as they've been installed, it's all right.

MR. CARICONE: But if you separate the liner from the containment, my guess is that liner would start heating up more quickly.

MR. WALLIS: Really heat an area -

MR. CARICONE: Because you don't have, obviously -

MR. WALLIS: Cool off.

MR. CARICONE: I believe that that's the

criteria, but that was never defined as a local temperature criterion.

MR. WALLIS: But you're meeting the regulations. I understand that. I'm just sort of wondering -

MR. CARICONE: Right. I understand.

MR. WALLIS: And I don't know where this goes, why this is a regulation. What makes sense, but it seems to me that buckling of the liner because of overheating some region might make a more sensible criterion. I don't know if the staff is going to do anything with that or not, but it just seems to me -

MEMBER ARMIJO: I want to ask a quick question. Is the fact that the temperature of the liner is lower than -- at SPU than current, is that strictly the result of these model changes?

MR. CARICONE: Right. There was definitely a methodology change, and -- do you want me -- I suppose I could elaborate on it a bit more.

MEMBER ARMIJO: Oh, no. I just -

MR. CARICONE: I understand.

MR. KAI: One thing I did want to point out, which you mentioned, is that as to whether we're a sub-atmosphere containment. We are a lot like Beaver Valley, in that we were originally designed to be sub-atmospheric. In the early 1990s, we removed the requirement to return sub-atmospheric, so we no longer have a requirement, post-LOCA, to return sub-atmospheric.

We operate, our plant normally operates slightly sub-atmospheric, but we do not -- we're like Beaver Valley in that we've eliminated, and actually we did almost a full 10 years ago, the requirement to return sub-atmospheric. So I just want to make sure that we understand that we're not thinking that we're like the original design, where we were required to the same -

CHAIRMAN SIEBER: The original design said you went back sub-atmospheric within an hour.

MR. KAI: Right.

CHAIRMAN SIEBER: And you don't do that now.

MEMBER ABDEL-KHALIK: Which of these two accidents do the numbers that you give in the table correspond to, in terms of containment liner temperature?

MR. CARICONE: Containment liner temperature comes from the steam line break analysis. Is that what the question is, whether it's a steam line break or a LOCA?

MEMBER ABDEL-KHALIK: Yes, that's the -

MR. CARICONE: That is determined by the steam line break analysis. Matter of fact, it's a double-ended rupture of 1.4 square foot pipe, which is the maximum size break that you can have because of the venturis that we have at the steam generators for zero power. And that includes super heat, so it's governed by a steam line break, double-ended rupture of a steam line break at zero power. Does that answer your question?

MEMBER ABDEL-KHALIK: Yes.

MR. KAI: Okay. Any questions? You also have to look at a long term alarm pressure and temperature in containment, primarily driven -- we've had some discussions about EQ. What this graph shows is where, in fact, the EQ profile had to be changed to accommodate the SPU results. You can see there's a little red triangle. That was the sole change that we made. The current EQ profile, which assumed a 60 psi peak pressure lasting for, I don't know, a couple of hundred seconds, that is still bounding. So the part that needed to be changed to accommodated SPU is that little triangle that shows in the red. And this is shown in the following page.

Now, you were not meant to read all the

different cases. But the point here is that we run dozens of cases to make sure that this EQ profile that we are using is still bounded at SPU condition. So each one of these traces represents a different case with different initial conditions, and different assumptions to try and maximize pressure. So we run dozens of cases for LOCA.

Same thing on -

CHAIRMAN SIEBER: The design pressure containment is how much?

MR. KAI: Forty-five psi gauge.

CHAIRMAN SIEBER: Forty-five pounds?

MR. KAI: Correct, 45 psi gauge.

CHAIRMAN SIEBER: Okay. And the pressure profiles that you show in Slide 11, go up to 55?

MR. KAI: Absolutely.

CHAIRMAN SIEBER: Absolutely. Okay.

MR. WALLIS: Isn't temperature more a problem than pressure?

MR. KAI: Yes, and we've got that. Right. The next slide -

MR. WALLIS: Well, I was hoping you were going to get to one that's important here.

MR. KAI: Okay. Slide 12 gives the temperature, exact same situation. The blue lines shows the current EQ profile. The red part is the part that needs to be changed to accommodate SPU. And you could see that the change is from about 2000 to about 20,000 seconds, that the cool-down is slightly slower in that part. And if you look at the next slide, the same thing. We run dozens of cases to make sure that we have captured a bounding EQ profile for the containment.

MR. WALLIS: It looks rather like a complicated boundary.

MEMBER ARMIJO: Yes, it sure does.

MR. KAI: It is.

MR. WALLIS: Why didn't you take something simpler? Bigger.

MR. KAI: Well, again, remember how this is used. Right? Because it's used to match up with how the points are tested.

MR. WALLIS: You're going to test them with a profile like that, or something much simpler?

MR. KAI: We want this profile that we use bounded by what the equipment was tested for. And, like I said, they vary all over the place in how the equipment is tested.

MR. WALLIS: The test profile would be something much more simple.

MR. KAI: Yes, but they won't -- again, they typically run the test, some of them reduce the temperature earlier, some reduce it later, so the net result is we really need to get a profile that we are sure would be bounded by all of our tests.

MEMBER ARMIJO: If you make this change, do you have to retest some equipment?

MR. KAI: No.

MEMBER ARMIJO: Everything that you have is current, meet this new profile?

MR. KAI: Correct. Other than the issue that you discussed about the steam valve closing. This is all inside containment, but that issue we did need to take some mods for qualification. Any other questions?

NPSH, and I think this is another one that I'm going to have to explain kind of carefully. You have to understand how our ECCS and our containment system work. We're a lot like Beaver Valley. Initially, when we were designed, the recirc spray pumps, which are pumps that are used for recirc, that take suction from the sump, originally started on a timer from the CDA signal. Eleven minutes after the receipt of the signal, the pumps would start to take suction from the sump. In eleven minutes, most -- a significant fraction of that water -- number one, there's not much water on the floor in the containment, and most of it, or a significant fraction is coming from the RCS, which is hot, much hotter than what you're going to spray. So the current -

CHAIRMAN SIEBER: Do you have a punch spray system?

MR. KAI: Yes.

CHAIRMAN SIEBER: That injects first, so that water is in there.

MR. KAI: Right, but only 11 minutes of it.

MR. WALLIS: Why isn't the sump temperature about the same at SPU?

MR. KAI: Okay.

MR. CARICONE: He's getting to it.

MR. KAI: Yes. You're ahead of me, again. Okay. So that's our current system. Okay? And the analysis was based on 11 minutes. And you can see that the sump temperature is like 260 degrees. A CDA will occur within seconds of a large break LOCA, so it's on the order of 11 minutes after CDA. Your sump temperature is 260 degrees, and that's because it's mostly made up of RCS water.

We made a mod last cycle, and it's also

cited to in GSI-191, to delay the start of the recirc spray pumps until we reach a low-low water level in the RWST. Now the pumps are going to start approximately 30 to 40 minutes, much longer, they'll start much later. Now you have 30 to 40 minutes of point spray, spraying in, filling the sump with cold water, and that results in this reduction of temperature.

MR. WALLIS: You're required to resume this containment pressure of one atmosphere. Is that right?

MR. KAI: Correct.

MR. WALLIS: Even though the temperature is 260 degrees Fahrenheit.

MR. KAI: Right. We assume -- we take no credit for the back pressure followed by the event.

MR. WALLIS: I understand that. It's somewhat unphysical.

MR. KAI: Well, it's margin. Again, think of that as margin there. Okay? So that's why this temperature is lower. It's not that -

MR. WALLIS: I understand that.

MR. KAI: Because physically now we are starting RSS pumps much later.

MR. WALLIS: But if it's 260, the

containment pressure better be more than one atmosphere.

MR. KAI: Right. That's definitely true. And you could see on the pressure -- so the net result here is that -- and, again, therefore the SPU really has not affected the current design-basis calc for NPSH at all, because the temperature is well below. And not so much SPU, but it's reflected in the modification we made.

I know that this issue is associated with GSI-191. That's still in progress. We're still working on that. We want -

CHAIRMAN SIEBER: According to the SER, you said you were done with your modifications for GSI-191.

MR. KAI: We've made the modifications. That's correct. We have installed a strainer, but the analysis pieces are still not -

CHAIRMAN SIEBER: Are you done?

MR. KAI: No, we are not done with the analysis pieces. There are still open issues on some of the WCAPs that we are following, so it's not completely resolved. But, again, like I said, at the very early thing in the morning is that -- what we tried to do here in this view is make sure that this doesn't make it worse, or it doesn't affect what we're planning to do in terms of what's listed in GSI-191.

MR. WALLIS: So, presumably your pump intake is significantly below 24 foot 6?

MR. KAI: Correct.

MR. WALLIS: It has to be. Otherwise you'd be boiling in the pump inlet.

MR. KAI: Yes, it is. You're right.

(Simultaneous speaking)

MEMBER WALLIS: About 24

MR. KAI: The bottom of the containment I believe is minus 24.6. That's the level of the floor.

MR. WALLIS: The pump is -

MR. KAI: Lower than that.

MR. WALLIS: -- 20 feet below that or something?

MR. KAI: Much lower.

MR. WALLIS: Significantly below that level.

MR. KAI: Correct. The same thing with the pipe stress. We have to make sure that the temperatures that we are predicting will be handled by the piping and associated attachments, et cetera. We are making a couple of modifications to two of the hangers inside containment to make sure that we maintain, meet our ASME requirements. But we have looked at -- we have done a very comprehensive look at pipe stress, looking at all different combinations of temperatures that we can anticipate from the LOCA.

We still do a minimum containment analysis. That is an inadvertent actuation point spray, so it actually has absolutely no -- SPU has no real impact at all on that. That is not power-driven. It's a function of what the initial bounds are in the containment temperature, and you start the sprays, and that drives you to the minimum containment pressure. So that was not an SPU-impacted analysis at all.

CHAIRMAN SIEBER: Now your containment spray is actuated by a pressure sensor inside containment?

MR. KAI: Correct.

CHAIRMAN SIEBER: What is it set at, one pound?

MR. KAI: The point spray system is -

CHAIRMAN SIEBER: The one that comes on first.

MR. KAI: Five psi gauge. Correct.

CHAIRMAN SIEBER: Half?

MR. KAI: Five.

CHAIRMAN SIEBER: Five. Okay. They used

to be set at one pound, which gave you inadvertent actuation everytime somebody sneezed.

MR. KAI: I'm pretty sure that the -- well, the transfers are outside containment. We have -- they're not actually located inside containment. Again, maybe, Mike O'Connor, you can tell me where they're located, but I'm pretty sure that the containment, the pressure transmitters are actually located outside containment.

MR. O'CONNOR: Right. I'm sorry. I misunderstood your question, but the actual location of the transmitter itself?

CHAIRMAN SIEBER: Yes.

MEMBER SHACK: Actuation set point.

MR. WALLIS: What does the spray draw from? Where does the water come from?

MR. KAI: Out of UST.

MR. WALLIS: Out of UST? How cold can that be?

CHAIRMAN SIEBER: How cold? Thirty-three.

MR. WALLIS: That will be 33 degrees?

MR. KAI: No, we have a spec that involves -- that requires the --

CHAIRMAN SIEBER: There's a minimum and maximum.

MR. KAI: Correct. It's 40-50 degrees is the range allowed on the -

MR. WALLIS: You have detainment coolers, do you? Fan coolers.

MR. KAI: We have a cistern that --

MEMBER WALLIS: The service water minimum temperature is 33 degrees. So you turned on the fan coolers, you could cool the containment to 33 degrees?

(Simultaneous speech.)

PARTICIPANT: The thing that's cooled by service water is our recirc spray pumps that take a suction on the containment sump and provide for long-term cooling to the core. Also take over for the spray system in containment. Those coolers are cooled directly by service water.

MR. WALLIS: But there are fan coolers in there?

PARTICIPANT: At this point in an accident, there would be no fans running. We wouldn't be using those.

MR. WALLIS: In terms of an inadvertent use or something.

PARTICIPANT: Now, the measurement of 33 degrees is with respect to the amount of cooling you can get to the recirculated water that's going back to the core.

MR. WALLIS: Well, the inadvertent use of the fan coolers couldn't bring the -- be a bounding event. That's all I'm trying to -

PARTICIPANT: No, it could not.

PARTICIPANT: Millstone does not have safety-related containment fan coolers?

PARTICIPANT: No.

PARTICIPANT: That is correct.

PARTICIPANT: We do have car fans that are cooled by CCP or CDS.

MEMBER ABDEL-KHALIK: What is the calculated value for the minimum containment pressure?

MR. KAI: We'd run at this 8 psi gauge, but remember what the actual calculated number is. That's in the FSAR. I can pull that out and pass it on to you.

Also, going back to your question, the car fans would normally be running so that would be something that we'd be wanting. I'm talking about for this particular depressurization scenario, depressurization, actuation of the point spray system.

MR. KAI: The last thing I was going to cover is sub-compartment, have looked at the impact of the releases, calculated for sub-compartment, and that does -- and those are the short-term pressurization effects.

One of the things that did in our original analysis is our original analysis can arbitrarily apply 10 percent margin on to the mass area, so in general, that margin was acceptable for most of the cases. There are a couple that we were not bounded, that we've re-analyzed, and so that the structural limits for -- and this problem with the pressurizer cubicle for the spray lines and the service line in the pressurizer cubicle.

In addition, we did credit leak before break to eliminate the large break, meaning to postulate large break for sub-component analysis. Millstone 3 was approved for large break, for leak before break, excuse me, but we haven't applied it to this analysis, even though we had gotten approval for leak before break. So we have submitted that credit as part of the SPU, and that has a big significant impact, and reduced the model analysis we do since, for example, the steam generator compartment where you have hot leg and cold leg piping, and you apply heat before break. They were recently analyzed to be shown acceptable, and now when you exclude that it represents significant margin.

PARTICIPANT: Mr. Chairman, that concludes our presentation on containment. We do have, as we're transitioning as our NRC Staff is coming up, their containment presentation, we do have a couple of answers to some of the questions. I'll just briefly address those, if I may.

One of the questions that was asked early on was what is our next most limiting component outside of the electrical generator. And the next for Millstone 3 at the current time is the HP turbine, and then after that there's several components that are limiting, so we're at the point now that we don't intend to do any other power uprates, because we have major components after this.

Another clarification that we'd like to make is, a question was asked about Millstone Unit 2. To be clear, Millstone Unit 2 was not the subject today. Millstone Unit 2 is a -- and it was asked whether we're doing a power uprate on Millstone Unit 2. Millstone Unit 2 is not a sister plant to Millstone 3. Millstone 3 is a four-loop Westinghouse unit. Millstone 2 is a two-loop CE-type design unit. Millstone 2 did a stretch power uprate, and I believe it was in the '70s time period, that was approved and implemented. It was approximately a 5.5 percent power uprate, so that was done in the past. And we do not intend to do a stretch power uprate on Millstone Unit 2.

Let's see we had another question. Dave Bucheit, you address the next one while I get your slide up.

MR. BUCHEIT: Yes, a couple of PRA questions I'd like to address. While Ron is putting the -- Bob, could you do me a favor and hand the Chairman that, please. Printed out a table from the license amendment request, and it includes all of the operator timing information. And explain to you how we use that information.

I alluded to the fact that we used RELAP to justify the times that were used in the analysis. Earlier we had used MAAP-4. What we did in every case was to run RELAP at the un-uprated, and at the uprated conditions assuming the operator action took place at that time, and determined whether or not there was core damage. And in each case there was not core damage, or the success criteria, steam generator dry-out, whatever it was, did not occur, so all we did was confirm that there was still margin in those numbers. We didn't confirm what the decrease in the margin was, which, I'm sorry, was really the question you asked. I can't answer that question. I can tell you what we did do.

MEMBER STETKAR: I looked at this. The one thing I think Jack mentioned also, is that the only clear time that I could find is there's an operator action called OAPHLR for hot leg recirc switch over, and in this table it says 538 minutes, which is approximately 9 hours. And in an other section of the LAR, it clearly states that that time is now reduced to five hours.

MR. BUCHEIT: Right.

MEMBER. STETKAR: So recognizing that the difference in PRA space between nine hours and five hours is insignificant for human reliability, but there is one instance where there was a substantial change in the time that's not reflected in this table. This table simply -- it's in the LAR we had. It's Table 213.2213-1 in the LAR.

CHAIRMAN SIEBER: But the only one that really changes is OAPFPW. All the rest of them stay the same.

MEMBER STETKAR: Some of them would change, Jack, because they take credit for -- they've taken credit for restoration of main feedwater.

Anything to do with steam generator dry-out times, or probably off-site power recovery times, which are generally related to secondary heat removal would change. How much they change, I don't know.

CHAIRMAN SIEBER: The RWST runs out of water quickly.

MEMBER STETKAR: Oh, yes. Sure. There are small changes in longer ones, but the more interesting ones from the PRA are typically in the 30 -- there are some very precise numbers in here, like 27 minutes, or 25 minutes. And those could change quite a bit. Not quite a bit, but they could change.

MR. BUCHEIT: But, again, we did do RELAP analysis at both un-uprated and uprated to confirm that when we assume the operator action meets the success criteria took place, that core damage did not result -- so we didn't get at the margin issue, but did confirm that the HEPs are still valid.

MEMBER STETKAR: Yes, but in terms of a delta, you don't really know what the delta is.

MR. BUCHEIT: Correct. Don't know what the delta is.

MEMBER STETKAR: And because -- the nice thing about the HCRORE methodology is that it is one -- because it's time reliability correlation, you could show directly what that change in the estimated human error rate would be.

MR. BUCHEIT: Yes. And, in fact, we'll probably -- we'll almost certainly have to do that when we bring the model up to the Reg Guide status. You had another question, sir, about whether or not we took credit for safety injection in any other scenarios. I looked into that, and the answer, indeed, is no.

There is manual override capability of this permissive, for feed and bleed, or something like that. They would still be able to actuate -

MEMBER STETKAR: Yes. I was just curious whether there was -- it would have to be pretty strange, but I just wanted to make sure that you looked into that.

MR. BUCHEIT: Yes, sir. That's all I have.

MR. BURNHAM: This is Robert Burnham. This morning we discussed at length the T hot spikes, and we were going to get some data for you. We've been in contact with the plant, and unfortunately, unless we have a spike in the recent plant process computer history, we can't get the trace, and we haven't seen one for a while.

What I can do is report what we had seen in the past through the engineering design changes and the valuations that were done. Initially, early in the cycle, we would see spikes of somewhat longer duration of 10 to 15 seconds, and they would be on the order of magnitude of a degree and a half to 3-1/2 degrees.

MR. WALLIS: That's in a positive direction. You had negative spikes, as well?

MR. BURNHAM: What we observed was that while the temperature was going up in one loop, it was actually going down in another loop.

MR. WALLIS: Down in another, so they're sort of symmetrical spikes?

MR. BURNHAM: Yes. We actually saw, for example, loop one in the hot leg, the temperature would go up, loop four the temperature would go down, and saw a similar reaction in loops two and three where they offset each other.

MR. WALLIS: We're talking about one or two degrees, maybe?

MR. BURNHAM: A degree and a half to 3-1/2 degrees. And that would be early in the cycle. Again, we haven't seen those for some time now.

Now, typically, what we have been seeing recently is a similar size spike of magnitude, again, a degree and a half to 3-1/2 degrees of a duration of approximately two to three seconds at the most. And that is what the T hot filter that we're putting in is specifically designed to filter out. And as we discussed this morning, that allows us to change the OPO to delta-T set points to gain margin back, because we're filtering out the spikes that were causing pre-trips and pre-alarms.

MR. WALLIS: If there was some kind of random turbulence, you would expect the spike size to vary a great deal. This seems to be a more regular thing, where the spikes usually are about the same?

MR. BURNHAM: Well, we see them -- the randomness we see is the degree and a half to 3-1/2 degrees. Usually, the signal is pretty stable, no more than that.

MR. WALLIS: It's a regular thing. It's not just a random turbulent thing which would be all over the place.

MR. BURNHAM: I'm not sure we have enough data to confirm either way. But, again, the magnitude is probably the more significant, and it's rarely seen, if at all, above 3-1/2 degrees.

PARTICIPANT: A follow-up to Dr. Wallis'

question on elevations. Again, containment floor elevation is at minus 26 four inches, first space center line elevation is minus 47 seven inches.

MEMBER WALLIS: Way down there.

PARTICIPANT: Way down.

MEMBER WALLIS: Do you have these long vertical pumps with the -

PARTICIPANT: I have the artist's representation over here if you want to take a look at it.

CHAIRMAN SIEBER: The pumps are typically hard to balance.

PARTICIPANT: I think we had answered it. So, Mr. Chairman, we have completed our presentations for the day. Thank you very much for the opportunity, and we believe we have closed out all open items that were introduced during the day.

CHAIRMAN SIEBER: Anyone have any final questions for the applicant? If not, I'd like to ask the Staff to do their companion presentation.

(Off the record comments.)

MR. SALLMAN: Good afternoon. My name is Ahsan Sallman. I'm a Reactor Systems Engineer in the Containment and Ventilation Branch, Division of Safety Systems. I was the reviewer of the SPU power uprate for the containment for Millstone 3. And, basically, we covered all of the RS-001 standard for the containment.

This slide presents a summary of what we reviewed, or what particular aspects of the containment we reviewed. We went through the RS-001, and we checked whether all the NRC-approved analytical methods were covered, and we had a lot of RAIs, and they were satisfactorily answered. We reviewed all the GDCs, and found that they were satisfied. SRP acceptance criteria was satisfied, and the Staff found that 10 CFR 50 requirements were okay.

The first aspect of containment review was the primary containment functional design, and we found that the licensee used Gothic methodology, which NRC has approved previously. There was an SE issue on that, and we accepted their methodology. The licensee used conservative initial condition for LOCA, and analyzed a spectrum of breaks for LOCA and MSLB.

The conclusions, short-term LOCA, MSLB, peak pressures and temperatures were bounded by the containment design conditions, and the long-term LOCA and MSLB pressure and temperature responses were okay, acceptable.

For the sub-compartment analysis, we have approved previously the leak before break criteria, and Millstone has used that in their analysis. And according to the NUREG-1838, and they used this criteria for the selection of five breaks. And there was sufficient margin in the differential pressure across the sub-compartment walls under SPU conditions.

We reviewed the mass and energy release analysis for LOCA and secondary pipe ruptures, and we found that there was a spectrum of break sizes that was analyzed by NRC-approved methods. And these are listed in this WCAP document that we have here. And so, also, secondary pipe breaks, energy release were used by NRC, the WCAP documents that are listed here.

For this same mass and energy release, LOCA and secondary pipe ruptures, the licensee used conservative assumptions and inputs to maximize the M&E release, and the Staff reviewed and agreed with the licensee's evaluation of LOCA M&E release.

MEMBER ABDEL-KHALIK: Did you do any independent confirmatory analyses of any of these calculations?

MR. SALLMAN: No, I did not, because there was nothing that I could see that would require such a review, because they were using acceptable codes and standards acceptable, the accepted codes that NRC has approved already. And the assumptions they were using were acceptable, inputs were acceptable, so that's the reason we did not go to independent review.

MEMBER ABDEL-KHALIK: So the fact that the methodology was changed in a lot of these analyses, which resulted in somewhat counterintuitive results, just simply because the methods were different, did not give you pause, or you didn't ask whether or not these changes are actually reasonable.

MR. SALLMAN: Well, one thing to mention is the Gothic containment analysis that we already have heard from the licensee that they used this new methodology which we have approved, plus one of the results that they talked about is the sump temperature was less than the previous sump temperature. That is also responded by the licensee that because of the change in the logic in a previous SE that was submitted, I mean, licensing document that was submitted, so those are the kind of things that, I guess, it was not necessary to review, do an independent calculation.

MR. LAMB: This is John Lamb. Just for your information, we asked 107 RAI questions, and approximately 16 of them came from Containment System, so it's about 15 percent. And when you were asking before about reactor systems, they asked about 38 questions, so about 40 percent or so. So you can see a bulk of our RAI questions came from Containment and Reactor Systems.

MR. SALLMAN: Next aspect of containment review was combustible gas control and containment, and was already approved, the hydrogen recombiners and monitoring system have been moved from the tech specs, so there was no impact of this aspect of the containment.

Heat removal system, the licensee did not take into account the containment accident pressure for calculation of the net positive suction head. They used the input parameters that were conservative or same as in the current analysis. And they used Gothic methodology to calculate a maximum sump temperature. So the heat -- the net positive suction head available requirement was met.

The next aspect of containment review was to see the minimum pressure analysis for ECCS performance capability. We found that the licensee used conservative initial conditions for calculating the minimum requirement containment back pressure, and the pressure transient bounds the transient used in the ECCS performance analysis, so it's unaffected.

We requested the licensee to see if they had reconsidered the Generic Letter 96-06, in which we wanted to find out that the licensee has reviewed overpressurization of the isolated water fill piping section and containment. And the licensee responded that they have reviewed, and there was no issue in that area. So that's the summary here.

I want to respond to one of the questions that was coming on the containment liner temperature of 280 degrees, I have seen in most plants, this is the temperature that they use. And the plant arrangement design is such that you don't find any direct jet impingement on the containment liner.

MR. WALLIS: Even from a steam break?

MR. SALLMAN: Yes. Because this is a very important design. The liner design is a very important part of the containment, and they don't want to have a liner exposed to 500 or something. So most of the plants I've seen, this is order of magnitude is 280 degrees, or close-by, I guess. That's the liner temperature. So the plant arrangement takes care of that.

MR. WALLIS: I'm trying to relate it to the zone of influence we have for the debris creation. And the zone of influence for some of these accidents actually goes way out to the containment liner, so, clearly, there is some kind of a jet that impinges on the liner, or at least it's assumed for that purpose. You're just saying it doesn't happen?

MR. SALLMAN: This has been the practice. You can see all FSARs, this is a liner temperature that is -- if it will have been an issue, it would have been considered very seriously.

CHAIRMAN SIEBER: Except for the very top of the steam generators, the crane wall, which is a circular cylinder of quite sturdy design, shields the liner from the working parts of the plant. So jet impingement, in my view, would be the rare occurrence, and possibly only from a rupture to the steam line. I think leak before break applies there.

MR. WALLIS: Have these folks installed the GSI-191 suction strainers?

MR. SALLMAN: I have seen the suction strainers have been installed, and -

MR. WALLIS: All been installed?

MR. SALLMAN: I think there were some issues in that, as they responded. The suction strainers has been considered, the pressure loss across the suction strainer has been considered in this analysis. I think it was already mentioned.

MR. COLLIER: This is Mike Collier. We have installed the advanced strainer last cycle.

MEMBER BANERJEE: What is -- do you have a buffer?

MR. COLLIER: We use TSP.

CHAIRMAN SIEBER: Any other questions? Okay. If there are no other questions, we are next at the point where we will receive public comments. Ms. Nancy Burton with the Connecticut Coalition Against Millstone would like 15 minutes to give us her viewpoint, so why don't we do that now. You want to put her placard up there. Okay. Go ahead.

MS. BURTON: Good afternoon. I'm Nancy Burton, and I direct the Connecticut Coalition Against Millstone. I thank you very, very much for the opportunity to be here. Thank you very much. It's been a very, very informative day. I have with me Arnold Gundersen, a nuclear safety engineer based now in Vermont. Mr. Gundersen worked at Millstone Unit 3 as Lead Licensing Engineer during the 1970s.

We have some handouts, which I'll distribute after I make a few comments. My comments are going to principally related to health, and Mr. Gundersen will be directing his comments for technical issues. And I hope that will be acceptable. Thank you. And we'll try to be very, very brief.

I just wanted to start by talking about Millstone Unit 2 for just one moment. We're here on Millstone Unit 3, but we've heard conflicting stories about whether we'll be here next on Unit 2. Just as a point of interest, Unit 2 suffered three unplanned shutdowns within a month recently, and has one of the most unreliable safety records in the industry. And we can only hope that the last gentleman who spoke from Dominion is correct, that there are no plans for an uprate for Unit 2.

I'm here principally to tell you that the organization I'm with, which consists of numerous statewide environmental organizations and safe energy organizations in Connecticut, as well as Millstone whistle blowers, and members of the community, we are absolutely opposed to this application.

With Mr. Gundersen's help, we have a calculation that if this uprate is approved, it will bring an additional $330,000 in profit to Dominion per year. That's what this is all about really, because there's no public need for this electricity. And my comments are directed to the complete failure of both the application and the NRC Staff review to address in any way the health effects on the population that lives in Southeastern Connecticut from this application.

The application does concede that there will be increases estimated at no less than 7 percent, and maybe much higher in the levels of radioactive materials released to the air and the water. This is, to a community which already suffers the highest levels of 12 categories of cancer in the State of Connecticut. And if we look at that figure of $330,000, and try to put a value on the little girl who died at seven months last year, had fourth stage liver disease in the immediate area, the numerous children who died of congestive heart failure moments after they were born in the immediate area of Millstone, the 16 workers we know of, or we've heard of at Millstone now suffering from cancer, the values are way, way, way off. They're askew, and this is not acceptable. The community does not accept seven, or eight, or nine, or ten, or twelve, or whatever percent more radiation dose to itself from this nuclear power plant.

I just want to mention that I will be circulating a declaration from Cynthia M. Besade, who grew up two miles from Millstone. Her father was a nuclear pipefitter at Millstone, and a whistle blower.

She has provided a statement in which she assesses the numbers of friends, neighbors, and associates in Niantic, in the area immediately around Millstone, who have suffered and died from cancer since Millstone began operations.

We keep a running count of people who are dying in this area, and we know there are clusters of brain cancer, breast cancer, childhood Leukemia, and bone cancer in an area that is celebrated for being a tourist attraction in our state. And this is entirely unacceptable. No rationale was given for this application other than to seek profit, and this body, we urge this body to ask what is the rush? There are so many unanswered questions here.

In Connecticut, we have a Department of Environmental Protection which presented the NRC and the NRC Staff with a series of questions. Those questions are still not answered, so I fail to see how the matter is at all closed. So we urge upon you a recommendation to the Full Committee denying this application. Mr. Gundersen will go into some of the technical and legal aspects of why that is necessary. And, at this time, I will turn the matter over to Mr. Gundersen.

MR. GUNDERSEN: It's Gundersen, S-E-N.

Get it right at the beginning, and I find I chase it a lot less. So thank you.

The NRC has a resume, but I used to be the Licensing Engineer at Millstone 3 back in the '70s when it originally was under construction. Of course, it was licensed in '86. I was long gone then, and was a Senior Vice President of Nuclear Energy Services at the time, and I was provided structural engineers at Millstone 3 at the time it was licensed.

Anyway, what we heard today was that the 61 other stretch power uprates, that was the number I heard, there was another quote that said that the Staff's review is based on experience gained on the other stretch power uprates. I heard an apology for burdening you with 26-day review cycle. And, again, I reiterate what Nancy just said, is what's the rush?

It's important to note -- I also heard several times we are a lot like Beaver Valley, and that there were peer uprates of 24 other Westinghouse four-loop plants. Just for the record, according to the NRC's web page, Beaver Valley was an extended power uprate review, not a stretch.

I need to talk about Millstone 3, and it really is not typical of the other reactors that have been considered by you in the past. I have my expert report that the NRC has already seen. Millstone 3 has an incredibly small containment. Initially, the number I used when I was analyzing this was 2.3 million cubic feet of free volume. Today, there was a number that was 1 percent smaller than that, 2.26 million cubic feet of free volume.

What I did was I used the NRC's web pages so there's nothing proprietary, and Dominion can stay in the room, and compared the reactors that have been - the Westinghouse reactors. And I didn't include as containment, and I didn't include the tiny ones. But anything over 2,000 megawatt thermal I looked at. And Millstone is the fifth smallest containment in the reactors, in the 25 reactors that fit that category at 2.3 million cubic feet.

Now why is that? It was originally sub-atmospheric, and if you go to the NRC's web page, you'll find that the only four-loop sub-atmospheric plant in the nation is Millstone 3. So its containment is incredibly unique. The volume is very small, sub-compartments are very tight. And when I was working there we recognized this. In '75, we all realized that the sub-atmospheric containment was not a great idea, but we didn't want to change it at that point, because it meant going back to square one with the licensing.

And, of course, when the reactor got built, it was within five years, the sub-atmospheric containment was essentially gone. We went from 10 psi to 14 psi in '91, if I remember right. But the containment, you're stuck with a 2.3 million cubic foot containment which is unlike any other four-loop Westinghouse plant out there.

In this, I urge you to take a look at the three tables, and they're all right out of the NRC's web page. Table Two in this report shows that Millstone is the fifth smallest reactor containment in the country. Then what I did was I -- in Table Three what I did was I compared the reactor power to the containment volume. And based on that criteria, Millstone 3 has the smallest containment in the country, dramatically smaller. That's the initial license reactor power. So even without the stretches of any of the reactors that ever have been licensed, the power output divided by the free volume is much lower than any other reactor in the country.

So then I went to the NRC's web page again, and Table Four of the handout takes all of these reactors and uprates their power, and again divides by the containment volume. And once again, Millstone is at the bottom of the list. So this is an incredibly small reactor, and of any reactor bigger than 2,700 megawatts thermal, Millstone is the smallest. So when you get down to the Indian Point 2s and Robinson's, smaller in that -- a couple that are smaller just cubic feet-wise, but when you then take the available energy that's going into that containment, Millstone winds up the smallest of that batch.

The other thing I need to note is that the -two other things and I'll be done - is that the NRC requirements say that a stretch power uprate is up to 7 percent. Millstone is actually above 7 percent. They round it up a thermal megawatt instead of down a thermal megawatt, and they're over by about 7.03 or something like that, so it actually exceeds the NRC's requirement for a stretch, and should be considered -as a legal basis that it should be considered as an extended power uprate, as was Beaver Valley at 8 percent.

So given it's not typical, and given it's, in fact, slightly above the 7 percent criteria, which is the threshold to be considered an extended power uprate, I'd like to suggest, especially in the area of the containment, that the NRC stand back and take a harder look. We just heard that there was "no independent confirmatory analysis by the NRC", on any of the information provided by Dominion. Given -- I know my sensors went up when I realized that this is the smallest containment in the world - why didn't the NRC do anything other than just review what Dominion provided?

I'm not saying Dominion provided was wrong. I don't know that, but what I am saying is that the NRC owed it to you, and to us, to civilians that they should have taken a harder look at this containment, because it was the only four-loop Westinghouse plant in the world that was a sub-atmospheric containment, and had the smallest free volume.

Okay. The Staff -- again, there seems to be a pressure on both you and the Staff. I heard the Staff say up here about an hour ago that they had less than a year to review the entire document. If it had been an EPU instead of an SPU, they would have had essentially another eight or nine months. What's the rush?

And last, but not least, I was involved in the uprate at Vermont Yankee, and it doesn't take much to rewind a generator. It can be done on-site with the existing copper, and some additional copper would be added. I would suspect that you'll see that this -- if the generator is the limiting factor in the next bit of power that can be gotten out of the reactor, given the cost of power on the grid right now, that Dominion will come back and try to squeeze another 2 percent out of a generator rewind without affecting the HP turbine. And I ask you, how are you going to treat that? Is that going to then make it an EPU? Is it 7 percent and 2 percent to get you to 9 percent, or is it 2 percent and it falls under the instrument error kind of a calculation.

In fact, we've got -- the containment was stretched in '91 when we went from sub-atmospheric 10 to sub-atmospheric 14. Now the reactor is stretched by another 7 percent, and we're looking at yet another couple of percent when you go for a generator rewind.

I guess that summarizes my - I have one other comment. The $300,000 was - the number I got in discovery on Vermont Yankee was $1 million in electric megawatt per year is the going rate for an uprate. The round-up to be over 7 percent is one thermal megawatt or a third of a electric megawatt, which is the $330,000. That's just the incremental, the actual outage, the actual uprate is probably generating on the order of $80 million a year for Dominion, not 330.

MS. BURTON: Thank you.

MR. GUNDERSEN: Thank you.

CHAIRMAN SIEBER: Any questions by any of the ACRS members?

MEMBER POWERS: Just one question. You spoke to the issue of confirmatory analysis. Staff did look at the computational method that was being used, and the inputs. Isn't that equivalent to doing a confirmatory analysis?

MR. GUNDERSEN: I guess given the short time that the Staff had to do it, and given that it wasn't typical, my answer would be no. Dominion had a handout here, it was Slide 17 of their containment analysis. And, again, I worked in that containment, and its space was incredibly tight. And it doesn't surprise me that a lot of the sub-compartment pressure issues became -- are becoming awfully significant.

Bullet One on Slide 17 of the Dominion presentation about half an hour ago shows that SPU mass and energy releases are bounded by the 10 percent margin provided in the current analysis, which tells me that until this uprate is approved, they had a 10 percent margin. Now they're throwing roughly 7 percent more energy into the containment, so what was adequate 10 percent margin before is now down to maybe a 2.9 percent margin, and that put the warning lights on when I was looking at this analysis. That was Slide 17 of the Containment section of the Dominion presentation. Thank you.

CHAIRMAN SIEBER: Any other questions?

MS. BURTON: Thank you.

CHAIRMAN SIEBER: Okay. Thank you very much.

MR. GUNDERSEN: Thank you.

CHAIRMAN SIEBER: Appreciate it. I want to discuss for a minute tomorrow's agenda. We are the first Subcommittee report tomorrow, and it's supposed to go from 8:35, following Bill Shack's remarks, to 10:45, which is two hours and 10 minutes. I think it would be very difficult to compress today's deliberations into two hours. On the other hand, we have all but three of our members, I think three, here today, and what we're doing is bringing everyone up to the same speed.

The agenda for tomorrow talks about the SPU overview by Dominion, and that's going to last 45 minutes. The fuel and safety analysis by Dominion, which would last a half an hour, fuel and safety analysis by the NRR, which last a half an hour, and closing remarks, which is 15 minutes, and that really covers the time allowed.

Now, the only thing that we covered beyond that was the containment analysis, and a few issues on electrical, and a few issues on the PRA. I'd like to ask any of the members would they object to just looking at the fuel and safety analysis at the Full Committee meeting, or do you want additional topics and information covered? Anybody have a comment?

MEMBER POWERS: Well, I think the fuel analysis section can be tightened up in its presentation a lot. I think they need to get to the point there, and get to bottom lines much more quickly than they did here. Here it was fine to go through it.

CHAIRMAN SIEBER: Yes. Well, it took them about four hours today, and that's a fair amount of time. On the other hand, we've all heard it. If we do that, my suggestion would be that we devote some time, like perhaps a half an hour, to our guest comments, which is the containment analysis. And if you can make those changes to tomorrow's agenda, we'd spend an hour and a half on fuel and safety analysis, and a half an hour on the containment analysis. Any other comments?

MEMBER MAYNARD: I would also suggest that, to me it's more important to talk about what are you doing for the SPU? Are you meeting the requirements and what your margins are. Either compare it to where you are currently, or if you're going to have a comparison, at least have some things that are more apples-to-apples. We spent a lot of time trying to talk about comparing an apple to an orange, and getting into an awful lot of non-productive dialogue.

CHAIRMAN SIEBER: Yes. I think I agree with you. When you look at the current ratings, and the current analysis, and compare it to the stretch ratings and stretch analysis, and in the process change assumptions and methods, the comparison is no longer valid. And that was the confusing issue today from 9:00 in the morning until now. And on the other hand, I don't think that the licensee or the Staff, either one of them, can change their presentations overnight, and I'm not going to ask them to do that, but I think your point is well taken.

Any other comments or remarks?

MEMBER MAYNARD: I would also just like to have part of the Staff's -- I would definitely like to know whether what we're doing is legal as far as the 7 percent of whatever.

CHAIRMAN SIEBER: Really, I'm not a master of the regulations, but whether it's a stretch power uprating or an extended power uprate determines what review standard the Staff uses. And at Beaver Valley, they used RS-001, at Millstone they used RS-001, so it's really -- it's irrelevant whether you call it a stretch or extended.

MEMBER MAYNARD: Not necessarily. You do have different review standards you may use. I do think it's important that there is -- what we're doing is consistent with the regulatory requirements. That's all I wanted to make sure of.

CHAIRMAN SIEBER: Well, we'll ask John Lamb to find that out overnight.

MR. LAMB: There was a legal, a request for a hearing. The ASLB denied that request. The Coalition Against Millstone filed an appeal, that appeal was responded to by Dominion and the Staff, and now it is with the Commission for their decision. And if you wish to see their legal opinion of the 7 percent argument, I can give you the ASLB's -

MEMBER MAYNARD: You've answered my question. I wanted to make sure that it had really been addressed. It may still be under contention or whatever, and that's fine.

MR. LAMB: I can give you the ASLB order that discusses the 7 percent, and you can see the legal opinion from them. And, obviously, they've appealed that.

MEMBER MAYNARD: And I just wanted to make sure -- I also agree with Dr. Powers that it is really irrelevant for what we do ourselves.

MR. LAMB: It's about 50 pages, but the section on that is about three or four pages on the 7 percent, the order.

MEMBER STETKAR: Jack, let me just ask you. Speaking for two of the members who are not here today who will be here tomorrow, both of whom are interested in PRA, and human reliability.

CHAIRMAN SIEBER: Glad they weren't here today.

MEMBER STETKAR: Yes, there's that, too. I don't know -- I'll ask everybody else in here for their opinion over whether we need -- there's nothing in the SER that discusses a word about the PRA. And this is certainly not a risk-informed application; so, therefore, the quality or any statements about the PRA in the application are relatively a moot point.

CHAIRMAN SIEBER: This is really a tech spec change.

MEMBER STETKAR: Yes. So, in that sense, in terms of governing time, I just wanted to bring it up in deference to -

MEMBER BANERJEE: Well, I would like to hear the answer to your question. The permissive, whatever it was. The injection system for the change in the logic.

MEMBER STETKAR: P-19 permissive?

MEMBER BANERJEE: Yes. Did you get an answer on that?

MEMBER STETKAR: Yes. They did. They answered it. And their answer is reasonable. They said there are no scenarios in the PRA that take credit for that. There are no scenarios in the PRA that take credit for high pressure injection during any condition that would not also have a low pressure in the pressurizer. In other words, that P-19 does not functionally disable injection for anything in PRA, and that seems pretty reasonable based on what -

MEMBER MAYNARD: I would recommend not putting it on the agenda tomorrow. It's not a risk-informed submittal. It's not affecting any decisions that would be made. And I think that the majority of the members heard the discussion today. You could fill -

MEMBER STETKAR: No, I just -- we've been characterized as a gang of three in the past. I want to speak up for the other two of the gang. That's all.

CHAIRMAN SIEBER: Well, you have.

MEMBER POWERS: A description of -- a term of endearment when you called -

MEMBER STETKAR: No, no, no. We recognize that.

CHAIRMAN SIEBER: Any other comments?

MR. WALLIS: In view of our reaction to the PRA, I'm not sure that the licensee would want to present it.

(Laughter.)

CHAIRMAN SIEBER: Any other comments? If not, it's 5:30, and we're early, believe it or not, and so we will adjourn.

(Whereupon, the proceedings went off the record at 5:22:08 p.m.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of: Subcommittee on Power Uprates

Name of Proceeding: Advisory Committee on Reactor Safeguards

Docket Number: n/a

Location: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Toby Walter
Official Reporter
Neal R. Gross & Co., Inc.

Dominion

Millstone 3 Stretch Power Uprate

ACRS Meeting
Overview

July 2008

Dominion

Introduction

J. Alan Price

Millstone Site Vice President

Ron Thomas

Project Manager

Project Team

- Significant Millstone Station And Dominion Corporate Engineering Staff Involvement. Not Turn-Key To An Outside Company.
- Full Time Dominion Team Members Included Representatives From Operations, Project Engineering, Nuclear Analysis & Fuel, Mechanical Engineering, Electrical Engineering, and Licensing.
- Operations Provided A Full Time Licensed Senior Reactor Operator During The Entire Project. Additional Operations Full Time And Part Time Personnel Were Added During The Past 15 Months As We Prepared For The Implementation Outage.
- Full Time Engineering Team Member Dedicated To Operating Experience And Margin Management.
- 22 Companies Supported The SPU Effort.
  - Shaw: Stone & Webster focused on the BOP and engineering program evaluation effort.
  - Westinghouse focused on the NSSS and accident analysis effort.
  - OEM Vendors included GE, Areva, TEI, Yuba, ALTRAN, ProtoPower.

Dominion

Licensed Core Power Level

- **The Current and Original Licensed Core Power Level For Millstone 3 Is 3411 MWt.**
- **Requesting Approximately A 7% Increase To 3650 MWt.**
- **This SPU Maintains The 2% Measurement Uncertainty Margin For Determining Core Power Level.**
- **Selection Of The New Power Level Was Based On Operations and Engineering Analysis That Showed No Major Modifications Necessary Up To 107% Power.**
  - Most Station Modifications Were Changes To Documentation, Calculations, Procedures And Drawings.
  - Hardware Changes: Replaced Feedwater Pump Turbine Rotor. Insulated 4 MOV motors in MSVB.
  - Control Function Changes.
  - Instrumentation Setpoint and Scaling Changes.

Dominion

Other Topics For Today

- Fuel and Safety Analysis.
- Containment.
- Electrical Power & Equipment Qualification.
- Flow Accelerated Corrosion.

Dominion

Millstone 3 Stretch Power Uprate

ACRS Meeting
Containment

July 2008

Containment Topics

- Analysis Summary.
- Analysis Methodology.
- Initial Conditions.
- Results.
  - EEQ Pressure Profile.
  - EEQ Temperature Profile.
  - Impact on RSS NPSH.
  - Piping Stress Analysis.
  - Minimum Containment Pressure.
- Sub-Compartment Analysis.

Dominion

Analysis Summary

- ☐ **Containment Analysis Methodology Updated To Current Standards.**

- ☐ **Significant Margin Remains Following SPU.**
  - 3.6 psi containment pressure margin.
  - EEQ profiles essentially unchanged.
  - No impact on current NPSH analysis.
  - Minimum pressure unaffected by SPU.
  - Subcompartment analysis remains bounding

- ☐ **Modifications Made To RSS Pipe Supports To Restore Stress Margins.**

Dominion

Containment Analysis Overview

- Current Long Term Mass and Energy release calculations have not been updated since original licensing.
- SPU long term mass and energy releases incorporates NRC approved methodology updates.
- Containment analysis changed to in-house NRC approved methodology.
- Because of changes in both mass and energy releases and containment methodologies, comprehensive sensitivity studies performed to assure limiting conditions identified.
- Original sensitivity studies repeated as well as new sensitivity studies performed consistent with current approved updated methodologies.

Containment Analysis Overview

- Ranges of initial conditions expanded for operational flexibility.
- Containment results used for a number of different component evaluations.
  - Containment minimum and maximum design pressure.
  - Maximum containment liner temperature.
  - Maximum pressure and temperature profiles for equipment qualification.
  - Maximum sump temperature at time of recirculation for pump NPSH.
  - Minimum and maximum temperature combinations for pipe stress evaluations.
- Bounding assumptions are dependent upon the component being evaluated.
- Reduction in cold leg temperature for SPU evaluated for impact on subcompartment analysis.

Dominion

Analysis Methodology

Methodology	Current	SPU
LOCA Mass & Energy Releases		
Blowdown	SATAN-VI	Unchanged
Reflood	WREFLOOD	Unchanged
Post-Reflood	FROTH	GOTHIC
Models	WCAP-9220 WCAP-6174 NS-TMA-2075 WCAP-8170	WCAP-10325 WCAP-8264 DOM-NAF-3

Dominion

Analysis Methodology

Methodology	Current	SPU
Steam Line Break Mass and Energy Releases	MARVEL	LOFTRAN
Containment Analysis	LOCTIC	GOTHIC
Subcompartment Analysis	THREED	Unchanged

Dominion

Analysis Initial Conditions

Parameter	Current	SPU
Volume, cu. ft.	2.26E+6	Unchanged
Pressure, psia	10.4 to 14.2	Unchanged
Temperature, °F	80 to 120	75 to 125
Humidity, %	50 to 100	0 to 100
SW Temperature, °F	33 to 78	33 to 80

Dominion

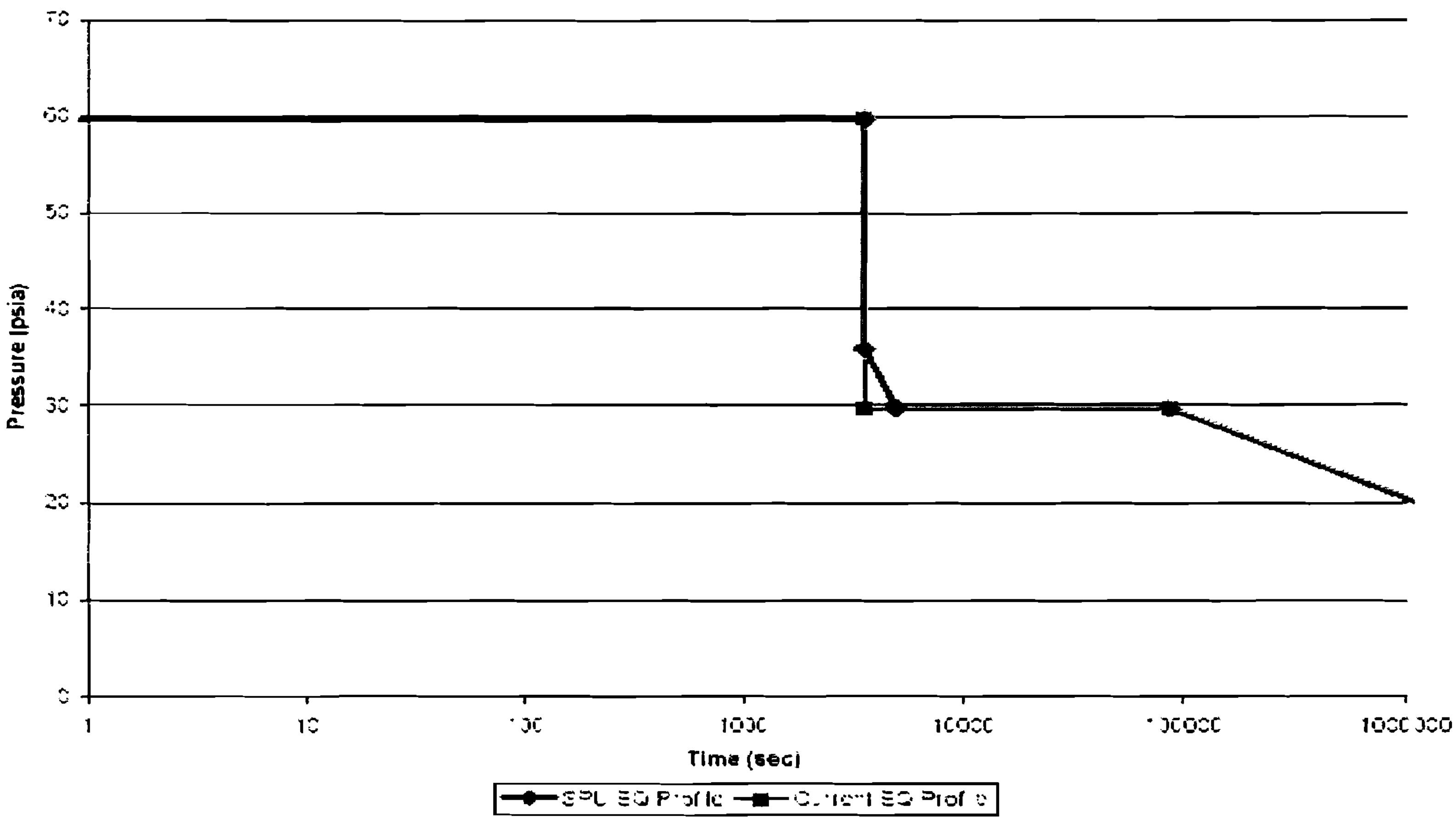
Results

Parameter		Limit	Current	SPU
Peak Containment Pressure, psig	LOCA	45	38.3	41.4
	SLB	45	34.14	38.15
Maximum Containment Liner Temperature, °F		280	255.9	241

Dominion

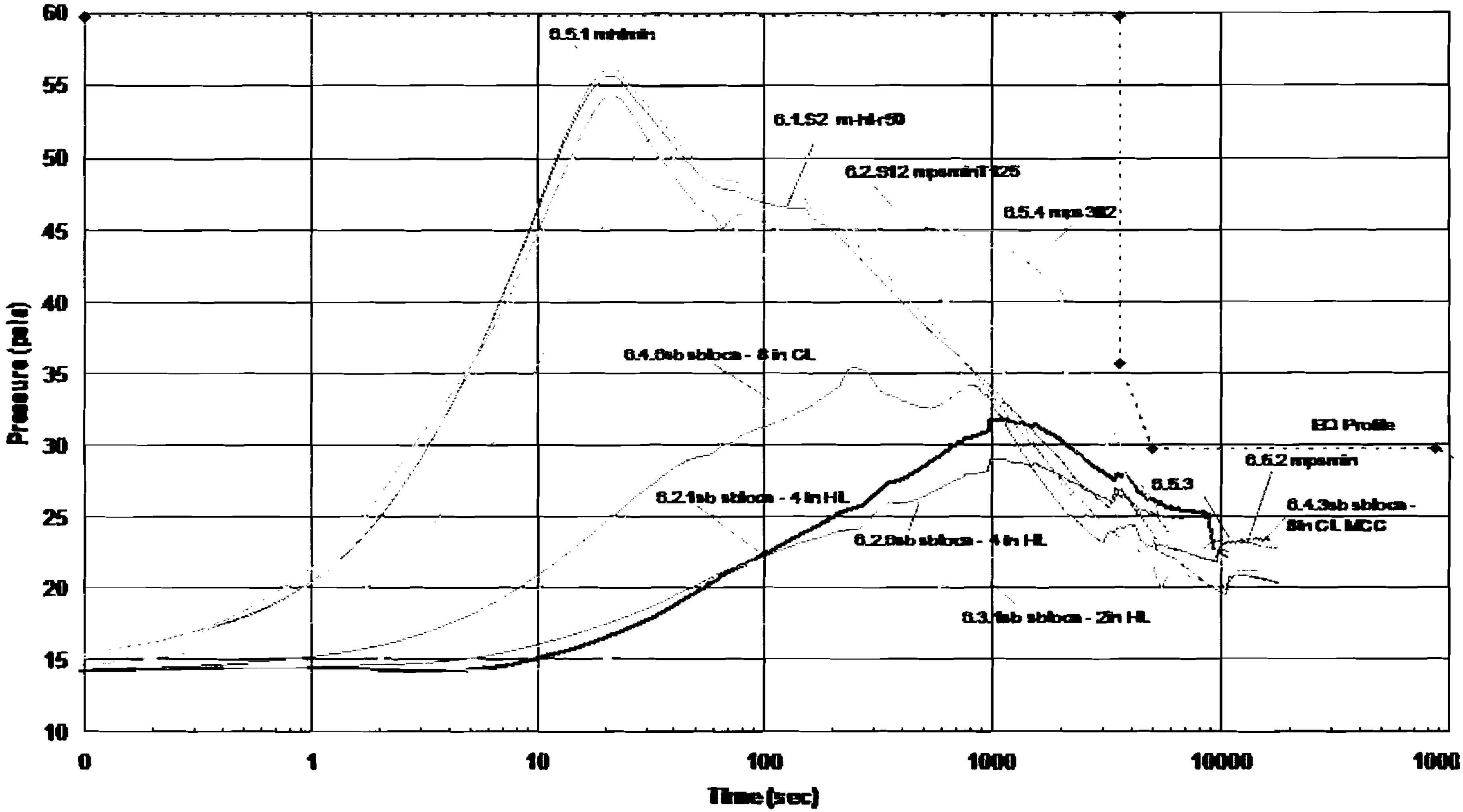
EEQ Pressure Profile

SPU Accident Pressure Profile



EEQ Pressure Profile

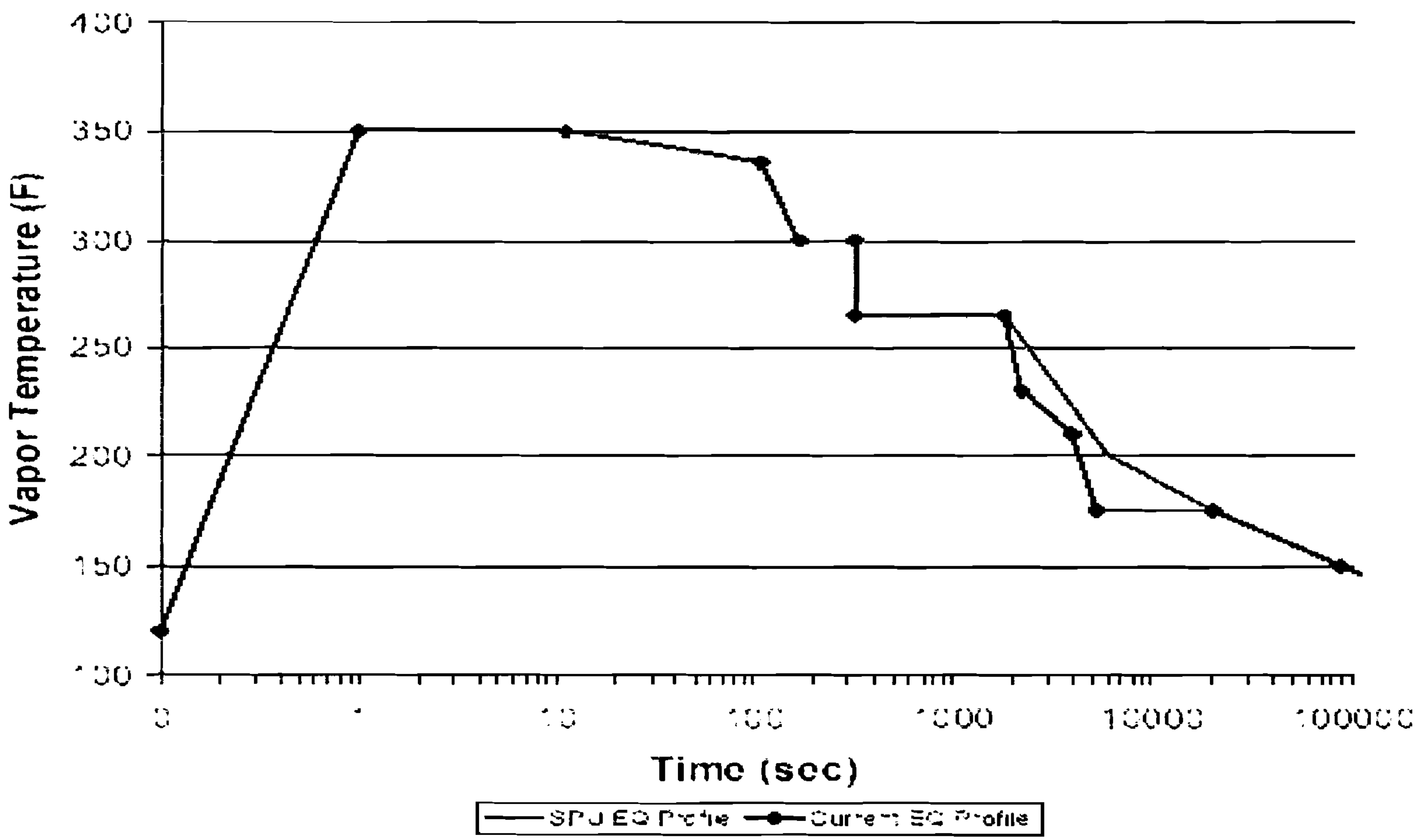
Figure 1 - LOCA Composite Pressure Profile



Dominion

EEQ Temperature Profile

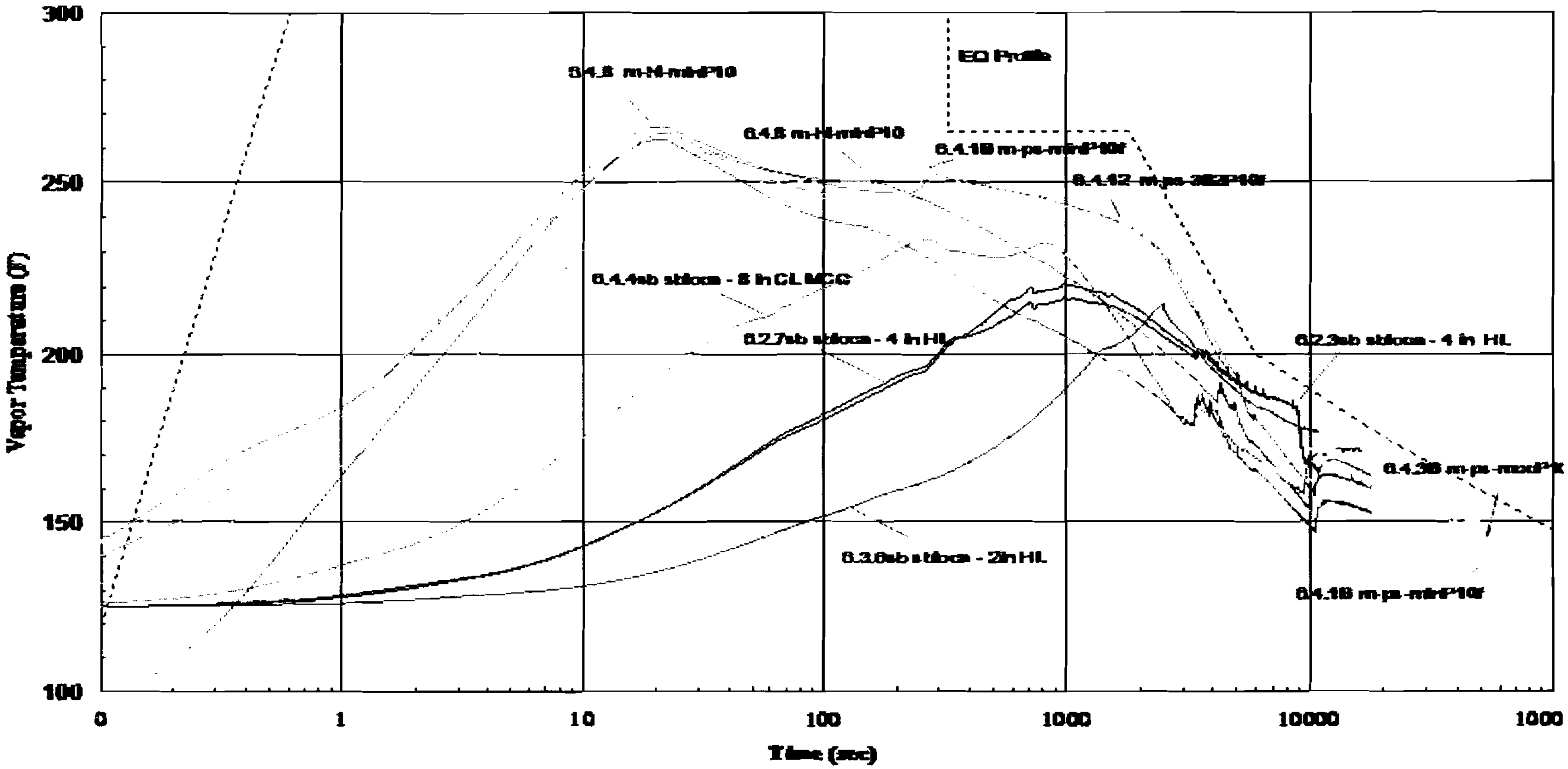
Containment Accident Temperature Profile



Dominion

EEQ Temperature Profile

Figure 2 - LOCA Composite Temperature Profile



Impact On RSS NPSH

Parameter	Current	SPU	Impact
Minimum Sump Elevation Above Elev -24'6", feet	4.33	4.33	Unchanged
Maximum RSS Flow, gpm	8220	8220	Unchanged
Maximum Sump Temperature, °F	260	225	Bounded By Current Calculation
Credit For Containment Back Pressure	No	No	Unchanged

Dominion

Pipe Stress Analysis

- **Limiting Conditions For QSS / RSS Piping.**
  - Heatup prior to spray.
  - Cooldown due to spray initiation.
  - Asymmetric temperature conditions due to postulated single failures.

- **Large Number Of Sensitivity Cases Analyzed,**

- **Stress Level A / B Criteria Applied.**

- **Modification To A Limited Number Of Pipe Supports Needed To Assure Stress Criteria Met For Asymmetric Temperature Conditions.**

Dominion

Minimum Containment Pressure

- Current Bounding Event Is An Inadvertent Containment Spray Actuation.
- Analysis is Independent of Core Power Level.

Dominion

Subcompartment Analysis

- For Most Scenarios, The SPU Mass And Energy Releases Are Bounded By The 10% Margin Provided In Current Analysis.
- SPU Analysis Credits Leak-Before-Break For Exclusion of RCS Piping Break In The Steam Generator Cubicle.
- New Analyses Performed For The Pressurizer Surge Line Break.

Dominion

Impact on Subcompartment Analysis

	Break	Basis For Acceptability
Pressurizer Cubicle	Spray Line	Mass and Energy Releases Bounded By 10% Margin
	Surge Line	Structural Analysis Performed To Demonstrate Acceptability
Steam Generator Compartment	Primary Side Breaks	Leak-Before Break Exclusion
	Feedwater Line Break	Conservatism In Critical Flow Correlation

ACRS Subcommittee on Power Uprates

NRC Staff Review of Proposed Stretch Power Uprate

For

Millstone Power Station, Unit 3

UNITED STATES NUCLEAR REGULATORY COMMISSION

July 8, 2008

Opening Remarks

Joseph G. Giitter

Director

Division of Operating Reactor Licensing

Office of Nuclear Reactor Regulation

Opening Remarks

- NRC staff effort
  - Requests for additional information
  - Supplements to application
- Challenging review areas included:
  - Fuel and core design analysis
  - Environmental Qualification
- Safety evaluation - no open technical issues

Introduction

John G. Lamb
Senior Project Manager
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Introduction

- Dominion Nuclear Connecticut, Inc. (DNC) is the licensee for Millstone Power Station, Unit 3 (MPS3)
- MPS3 Proposed Stretch Power Uprate (SPU)
  - 3,411 to 3,650 Megawatts Thermal (MWt)
  - Approximately 7% increase (239 MWt)
- Background
  - Licensed January 31, 1986
  - Approved License Renewal – October 2005
  - Operating License expires November 25, 2045
- Method of NRC staff review – RS-001 as guidance
- Schedule and Implementation

Topics for July 8, 2008

- Introduction and Overview of the SPU application
- Fuel & Safety Analysis
- Containment Analyses
- Electrical and Grid Reliability
- Flow-Accelerated Corrosion

Fuel and Reactor Systems Evaluation MPS3 SPU

Benjamin Parks and Samuel Miranda
Reactor Systems Branch
Leonard Ward, Ph.D.
Nuclear Performance and Code Review Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

Review Scope

- Staff reviewed the impact of SPU on
  - Fuel system and nuclear design
  - Thermal-hydraulic design
  - Overpressure Protection
  - Accident & Transient analyses
  - LOCA
  - ATWS
  - Westinghouse methods

Review Method

- Scope of EPU evaluations generally followed NRC-accepted, generic SPU guidelines and evaluations

- Analyses and evaluations are based on NRC-approved methodologies, analytical methods, and codes

- Followed the EPU review standard (RS-001)

Fuel System and Nuclear Design

- Evaluations:
  - Mechanical based on multiple fuel types
  - Nuclear/Thermal-hydraulic on RFA/RFA2
- Uprate effects:
  - Slight increase to linear heat rate
  - Slightly less peaked core design
- Licensee's evaluations demonstrate that acceptable core design may be achieved at uprated power level
- Cycle-specific analyses and evaluations will demonstrate compliance in accordance with NRC-approved reload licensing process

Thermal-Hydraulic Design

- Matrix 8 of RS-001 references General Design Criteria 10 and 12.
- Compliance with GDC 10 was shown by evaluations of DNB using ANSI acceptance criteria, Conditions I-IV.
- Compliance with GDC 12 is achieved by inherently stable core design, with trip features to shut down reactor in the event of xenon oscillations.

Accident & Transient Analyses

- Review included those transients covered in Matrix 8 of RS-001; results were acceptable as noted in staff's SER.
- Several accidents/transients warranted additional staff review:
  - Overpressure Protection
  - Inadvertent ECCS Actuation/P-19 Permissive
  - Rod Withdrawal at Power – Low Power

Overpressure Protection

- Limiting Overpressure event is Loss of Load/Turbine Trip
- Applicable ANSI Condition II Acceptance Criterion:
  - Limit peak pressure to 110% of reactor coolant system design pressure
- Two trips terminate event:
  - High Pressurizer Pressure
  - Overtemperature-ΔT

Overpressure Protection Continued

- Pursuant to staff request for additional information, licensee analyzed event crediting only the second (OTΔT) trip.

- Results of sequence crediting either trip were acceptable

- Peak pressure did not exceed 2750 psi (110% RCS Design Pressure)

Inadvertent ECCS Actuation

- Licensee will implement new permissive, P-19 Cold Leg Injection Permissive

AOO Acceptance Criterion

- "By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV type without other incidents occurring independently."
- NRC reminded licensees that this criterion is in the plant licensing bases, and therefore must be met (RIS 2005-29).

AOOs That Add Mass to RCS

- Inadvertent Actuation of ECCS can develop into a small break LOCA at the top of the pressurizer, if a PORV sticks open.
- In analyses, PORVs that are not qualified for water relief are assumed to stick open after they relieve water.

Millstone Unit 3 Operating Experience

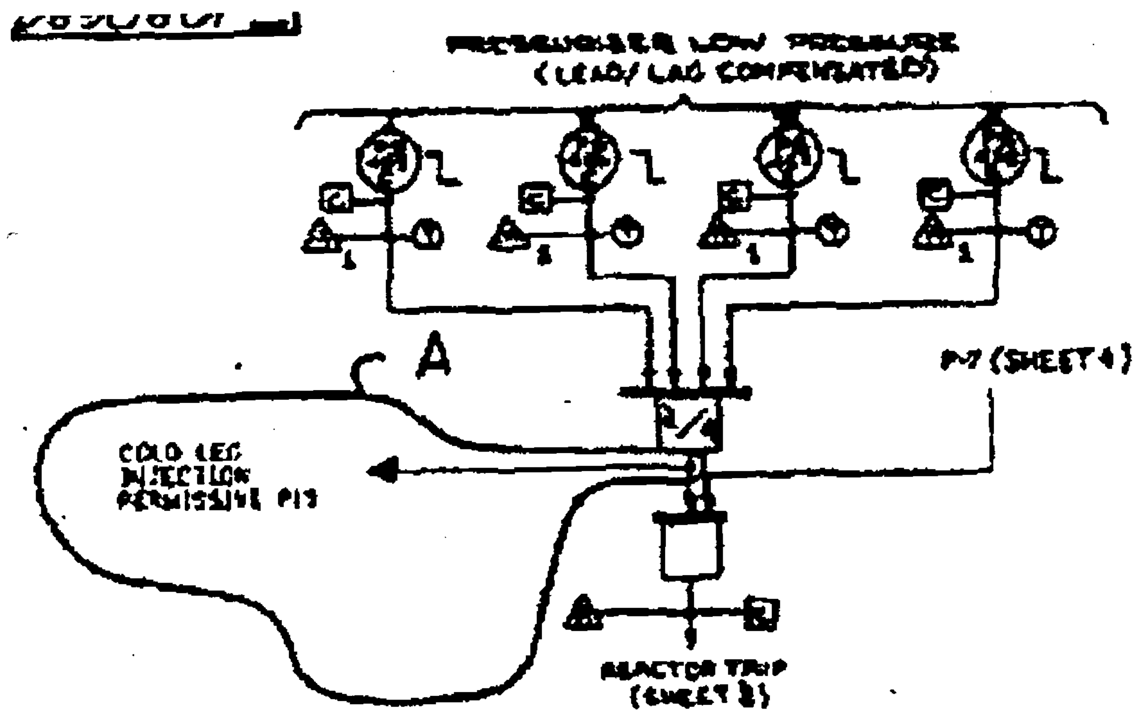
- Inadvertent actuation of ECCS incident occurred on April 17, 2005.
- Resulted in water relief through the PORVs

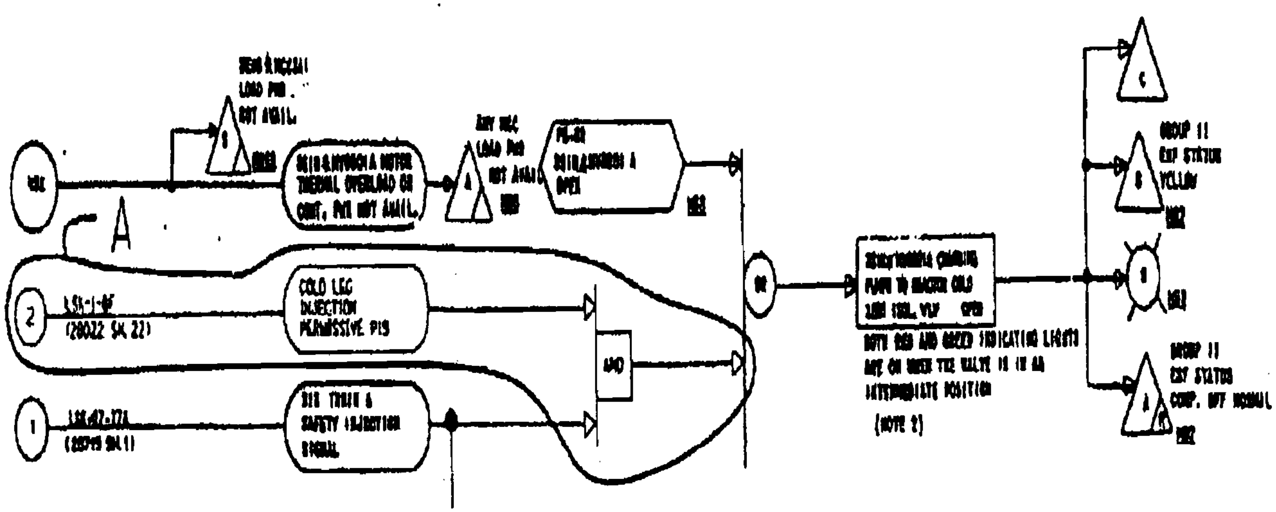
Millstone Unit 3

- PORVs are qualified for water relief
- P-19 Permissive interlocks the charging cold leg injection valves with a low pressurizer pressure signal coincident with an SI signal.

P-19 Permissive

- Charging cold leg injection valves do not open unless RCS pressure < low pressurizer pressure reactor trip setpoint and an SI signal is present.
- A single fault does not cause the cold leg injection valves to open. (P-19 would have prevented the incident of 2005.)


PRESSURIZER LOW PRESSURE
(LEAD/LAG COMPENSATED)
A
P-7 (SHEET 4)
COLD LEG
INJECTION
PERMISSIVE P13
REACTOR TRIP


A
COLD LEG
INJECTION
PERMISSIVE P19
(20022 SH 22)
AND
OR
GROUP 11
ESF STATUS
YELLOW
BOTH RED AND GREEN INDICATING LIGHTS
ARE ON WHEN THE VALVE IS IN AN
INTERMEDIATE POSITION
(NOTE 2)
GROUP 11
ESF STATUS
COMP. OFF NORMAL

Rod Withdrawal At Power

- Rod withdrawal at power evaluated in Licensing Report with acceptable results
- LR referenced a generic disposition of the potential for RCS overpressurization, given a RWAP initiated at a low power level
- Staff questioned the generic evaluation

Low Power RWAP – Generic Study

- Westinghouse evaluated the potential for overpressure conditions following a RWAP initiated at a power level where the high neutron flux-low setting can be blocked.
- Evaluation pertained to plants with water-filled loop seals on pressurizer safety valve discharge piping.
- Millstone 3 does not have water-filled loop seals; pressure relief would occur earlier.

Details of Generic RWAP Evaluation

- Performed for 4-loop Westinghouse plant
- Total power less than Millstone 3 SPU
- Pressurizer level lower than Millstone 3
- Remaining input parameters conservative relative to Millstone 3 SPU

Westinghouse Study of RWAP at Millstone 3

- Remove seal purge delay on pressurizer safety valve
- Increase core power level
- Increase pressurizer initial water level

Westinghouse Study of RWAP at Millstone 3 Continued

- Results confirmed that eliminating seal purge delay compensated for increased liquid volume in pressurizer and increased nuclear power addition capability

- Conclusion: Positive Flux Rate Trip terminates transient and Pressurizer Safety Valves mitigate pressurization effects.

LOCA

- Large Breaks evaluated with ASTRUM Best Estimate Method (Change from BART/BASH Appendix K Method)
- Small breaks evaluated using NOTRUMP (no change)
  - SBLOCA results show significant margin to regulatory limit

LOCA Results

	Small Break	Large Break	Acceptance Criterion
Peak Clad Temp, °F	1193	1781	2200
Local Cladding Oxidation, %	0.05	3.5	17
Core Wide Oxidation, %	0.01	0.12	1.0

Westinghouse Methods

- Licensee implements VIPRE/RETRAN to replace THINC-IV/LOFTRAN
- Use of LOFTRAN is maintained in some cases
  - Steam Generator Tube Rupture
    - Modified version of LOFTRAN to credit operator action
  - Transients where use of WRB-2M is restricted

Summary

- Transient and accident analyses demonstrate acceptable results at uprated conditions
- Fuel design remains acceptable to support the uprate
- Methods implemented acceptably

Containment Review

Ahsan Sallman
Containment & Ventilation Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

Containment Review

- Primary Containment Functional Design
- Subcompartment Analyses
- Mass and Energy Release
- Combustible Gas Control in Containment
- Containment Heat Removal
- Pressure Analysis for ECCS Performance Capability
- Reconsideration of Generic Letter 96-06

Summary of Staff Review

- RS-001, “Review Standard for Power Uprates,” was followed as guidance
- Applied NRC-approved analytical methods
- RAIs were satisfactorily answered
- Applicable GDCs were satisfied
- SRP acceptance criteria were satisfied
- Met 10 CFR 50 requirements

Primary Containment Functional Design

- Application of GOTHIC 7.2a methodology to MPS3 approved by SE, dated August 30, 2006
- Conservative initial conditions for LOCA and MSLB
- Analyzed a spectrum of breaks for LOCA and MSLB

Primary Containment Functional Design Continued

- Conclusions
  - Limiting short-term LOCA & MSLB peak pressure & temperature are bounded by the containment design conditions
  - Limiting long-term LOCA & MSLB pressure & temperature responses are evaluated to be acceptable from the standpoint of EQ

Subcompartment Analyses

- NRC has approved leak-before-break (LBB) methodology for MPS3 contained in the license renewal SE – NUREG-1838
- Used LBB criteria for selection of pipe breaks
- Conclusion
  - Sufficient margin in the differential pressures across the subcompartment walls under SPU conditions

Mass and Energy Release Analyses for LOCA & Secondary Pipe Ruptures

- Analyzed a spectrum of breaks for LOCA based on NRC-approved methods: LOCA blowdown & reflood (WCAP-10325-P-A & WCAP-8264-P-A) and post-reflood (DOM-NAF-3-0-0-P-A
- Analyzed a spectrum of secondary breaks based on NRC approved methods in WCAP-8822, WCAP-8822-01-P-A, WCAP-8822-02-P-A, and WCAP-7907-P-A

Mass and Energy Release Analyses for LOCA & Secondary Pipe Ruptures Continued

- Used conservative assumptions and inputs to maximize M&E release
- Conclusion
  - Staff reviewed and agreed with the licensee's evaluation of LOCA M&E release

Combustible Gas Control in Containment

- SER, dated June 29, 2005, removed hydrogen recombiners & monitoring system from Tech Specs as per 10 CFR 50.44 and RG 1.97
- Conclusion
  - SPU does not impact combustible gas control in containment

Containment Heat Removal

- Containment accident pressure was not used for calculation of NPSHA for RSS pumps
- Input parameters are conservative or the same as the current analysis
- Used GOTHIC methodology to calculate the maximum sump temperature

Containment Heat Removal Continued

- Conclusion
  - RSS pumps NPSHA requirement is met

Pressure Analysis for ECCS Performance Capability

- Used conservative initial conditions for calculating the minimum containment backpressure transient
- Calculated containment pressure transient bounds the transient used in the ECCS performance analysis
- Conclusion
  - ECCS performance capability is unaffected by SPU

Reconsideration of Generic Letter 96-06

- GL 96-06 states, "Thermally induced overpressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system operability requirements."

Reconsideration of Generic Letter 96-06 Continued

- Licensee reviewed GL 96-06 for piping system penetrating containment along with its relief valves as a part of SPU system design pressure & temperature evaluation
- Conclusion
  - No hardware changes are necessary for SPU conditions

Summary

- Applicable GDCs were satisfied
- SRP acceptance criteria were satisfied
- Met 10 CFR 50 requirements

Electrical Systems

Sheila Ray
Electrical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Electrical Systems Regulations

- 10 CFR 50.49
  - Environmental Qualification
- 10 CFR 50.63
  - Station Blackout
- 10 CFR Part 50, Appendix A, GDC-17
  - Electrical Power Systems

Electrical Systems Evaluation

- Loading on safety equipment remains bounding
- Current analyses remain bounding
  - AC Distribution System
  - EDGs
  - Switchyard
  - DC System
  - Station Blackout
  - Power Block Equipment

Environmental Qualification

- Existing environmental qualification remain valid for all areas except Main Steam Valve Bldg
- Additional analysis was performed due to the environmental changes in the MSVB. All equipment remains qualified in accordance with 10 CFR 50.49.

Grid Stability

- Safe operation under increased electrical output and increased plant load
  - Voltage studies indicated no adverse impacts.
  - The grid remained stable for all analyzed contingencies.

Summary

- The Electrical Engineering Branch staff found the following areas acceptable for operation at uprated conditions:
  - Environmental Qualification
  - Offsite Power Systems
  - Onsite Power Systems
  - Station Blackout

Flow-Accelerated Corrosion

Matthew Yoder
SG Tube Integrity & Chemical Engineering Branch
Division of Component Integrity
Office of Nuclear Reactor Regulation

Flow-Accelerated Corrosion

Regulatory Evaluation

- Generic Letter 89-08, "Erosion/Corrosion Induced Pipe Wall Thinning"
- EPRI NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program"
- Design code minimum wall thickness

Flow-Accelerated Corrosion

- Some changes in variables that affect FAC.
  - Primarily velocity and temperature
- Components inspections will increase as a result of the increased FAC rate at SPU conditions.
- CHECWORKS computer models are being updated prior to implementing the SPU.
- At SPU conditions, the FAC program remains consistent with industry guidelines.

Summary

- Staff concludes the licensee has addressed changes in the plant operating conditions on the FAC analysis.
- Staff concludes the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed SPU.

Staff Conclusion

- The staff concludes that there is reasonable assurance that the health and safety of the public will not be endangered by the proposed SPU.

EXHIBIT A

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of
DOMINION NUCLEAR CONNECTICUT INC.)
MILLSTONE POWER STATION UNIT 3) Docket No. 50-423
LICENSE AMENDMENT REQUEST)
STRETCH POWER UPRATE)

DECLARATION OF ARNOLD GUNDERSEN SUPPORTING CONNECTICUT COALITION AGAINST MILLSTONE IN ITS PETITION FOR LEAVE TO INTERVENE, REQUEST FOR HEARING, AND CONTENTIONS

I, Arnold Gundersen, declare as follows:

1. My name is Arnold Gundersen. I am sui juris. I am over the age of 18-years-old. I have personal knowledge of the facts contained in this Declaration.

2. I reside at 376 Appletree Point Road, Burlington, Vermont.

3. The Connecticut Coalition Against Millstone has retained me as an expert witness in the above captioned matter.

4. I have a Bachelor's and a Master's Degree in Nuclear Engineering from Rensselaer Polytechnic Institute (RPI) cum laude.

5. I began my career as a reactor operator and instructor at RPI in 1971 and progressed to the position of Senior Vice President for a nuclear licensee. I am a vetted expert witness on nuclear safety and engineering issues. My more than 37-years of professional nuclear experience include and are not limited to: nuclear

safety expert witness testimony; nuclear engineering management and nuclear engineering management assessment; prudency assessment; nuclear power plant licensing, licensing and permitting assessment, and review; nuclear safety assessments, public communications, contract administration, assessment and review; systems engineering, structural engineering assessments, cooling tower operation, cooling tower plumes, nuclear fuel rack design and manufacturing, nuclear equipment design and manufacturing, in-service inspection, criticality analysis, thermohydraulics, radioactive waste processes and storage issue assessment, decommissioning, waste disposal, source term reconstructions, thermal discharge assessment, reliability engineering and aging plant management assessments, archival storage and document control technical patents, federal and congressional hearing testimony, and employee awareness programs.

6. My Curriculum Vitae delineating my qualifications is attached.

7. My Declaration is intended to support Connecticut Coalition Against Millstone's Petition For Leave To Intervene, Request For Hearing, and Contentions.

8. The Five Contentions my Declaration supports are:

   A. The proposed power level for which Dominion Nuclear has applied to uprate Millstone Power Station Unit 3 exceeds the NRC Stretch Power Uprate (SPU) regulatory criteria.

B. The design margins for the Millstone Unit 3 Containment, which help to protect public health and safety, have been significantly reduced by license amendments granted in 1991, and Dominion's proposed power increase, if granted, will further reduce Containment margins designed for safety.

C. When compared to all other Westinghouse Reactors, Millstone Unit 3 is an outlier or anomaly. Dominion's proposed uprate is the largest percent power increase for a Westinghouse reactor. Additionally, Millstone Unit 3 also has the smallest Containment for any Westinghouse reactor of roughly comparable output.

D. Construction problems due to the unique Sub-Atmospheric Containment Design, coupled with the impact upon the Containment concrete by the operation of the Containment Building at very low pressure, very high pressure and very low specific humidity, place the calculations used to predict the stress on that concrete Containment in uncharted analytical areas.

E. The impact of flow-accelerated corrosion at Dominion Nuclear's proposed higher power level for Millstone Unit 3 have not been adequately analyzed and addressed.

9. As an expert witness, who happens to hold both a Bachelor's and Master's degree in Nuclear Engineering, have more than 35-years of nuclear industry engineering experience, and as a former Northeast Utilities employee worked on Millstone Nuclear Power Station Unit 3, in my professional opinion the Dominion Nuclear application fails to satisfy *any of the NRC criteria* to be accepted as a Stretched Power Uprate. A thorough review of the evidence presented by Dominion Nuclear and compared and contrasted with NRC Stretched Power Uprate requirements clearly shows that the Dominion Nuclear Stretched Power Uprate application should in fact be treated as an Extended Power Uprate (EPU) application.

10. According to the NRC, there are two criteria[1] that must be met for a licensee to be considered for a Stretch Power Uprate (SPU):

   A. An increase in the reactor power that is "**up to 7 percent**" and
   B. "**... are within the design capacity of the plant**"
   C. Furthermore, the NRC states that achieving a Stretch Power Uprate "**depends on the operating margins included in the design of a particular plant**". [Emphasis added]

11. In my opinion, the magnitude of Dominion Nuclear's proposed power increase, the uniqueness of the initial Millstone 3 Power Plant Containment design, the Containment's unusually small size, and the fact that the design margins of the Containment have already been dramatically reduced by changes made to

[1] www.nrc.gov/reactors/operating/licensing/power-uprates

Millstone 3 in 1990 by Northeast Utilities, makes it necessary for the NRC to conduct the more thorough and intensive Extended Power Uprate review.

12. Dominion Nuclear has characterized this proposed increase in power at Millstone Unit 3 (Millstone Power Station Unit 3) as a Stretch Power Uprate (SPU), and Dominion Nuclear claims that Millstone 3 meets all the criteria for a Stretched Power Uprate. According to Dominion's letter filing for the power increase:

> "DNC developed this LAR utilizing the guidelines in NRC Review Standard, RS- 001, "Review Standard for Extended Power Uprates." In addition, requests for additional information (RAIs) regarding SPU and Extended Power Uprate (EPU) applications for other nuclear units were reviewed for applicability. Information that addresses many of those RAIs is included in this MPS3 SPU LAR. RS-001 states that a SPU is **characterized by power level increases up to 7 percent and does not generally involve major modifications.** Plant modifications are addressed in Section 1.0 of the License Report (LR) (Attachment 5) and are not considered to be major. Since the requested uprate is 7 percent and does not involve major plant modifications, it is considered to be a Stretched Power Uprate."[2] [emphasis added]

13. Contention 1: To begin with, the Dominion Nuclear application fails to satisfy the first NRC criteria[3] that the NRC has set the power limit for SPU's at "**... up to 7% ...**". Yet Dominion Nuclear notifies its acceptance of the NRC's specific criteria in stating "**...a SPU is characterized by power level increases up to 7 percent** ... ". Most importantly, Dominion's proposed power increase at Millstone Unit 3 in fact exceeds the seven percent limit established by the NRC and accepted by Dominion Nuclear.

---

[2] Letter, Dominion Nuclear to NRC, SPU Filing, February 2007
[3] www.nrc.gov/reactors/operating/licensing/power-uprates

14. Millstone Power Station Unit 3 is currently licensed to operate at 3411 thermal megawatts (MWt). This number signifies how much heat the reactor is generating and is accurate to four significant figures (numbers).

- The proposed power level of 3650, for which Dominion Nuclear has applied, exceeds the NRC 7% limit that would qualify the power uprate for the less rigorous review of a Stretched Power Uprate.

- Dominion Nuclear has applied for a power increase to 3650 MWt, which is a full 300 KW above what is allowable by the NRC regulations for a Stretch Power Uprate.

- Let's look at the math. Multiply the current licensed power by the NRC's maximum allowable 7% SPU increase. The calculation total equals 3649.7 MWt, which is below the reactor power level of 3650 MWt for which Dominion Nuclear has applied. <u>3411 x 1.07 < 3650</u>

- The 7% NRC limit is accurate to two significant figures. When multiplying a two significant figure number by a four significant figure number *mathematical methodology demands the calculation be rounded down <u>not up</u>* as Dominion Nuclear has done in its application.

- By rounding its proposed reactor power level to a higher power level the requested Dominion Nuclear reactor power increase exceeds the regulatory limit for a Stretched Power Uprate (SPU). Thus, this unscientific rounding up of the thermal megawatt power to a higher power level causes the reactor power to exceed the legal Stretched Power Uprate limit of "up to 7 %" by a full 300 KW.

15. The mathematical evidence shows that Dominion Nuclear proposed power level increase for its Millstone Power Station Unit 3 exceeds the 7% regulatory limit clearly established by the NRC. Therefore, it is my opinion that the Dominion Nuclear's Millstone Unit 3 *is disqualified* for a Stretched Power Uprate.

16. Moreover, while on the face, this mathematical discrepancy may not appear to be a huge number, the 300 KW discrepancy between the NRC 7% limit and Dominion Nuclear's application for a 3650 megawatt thermal increase at Millstone 3 is a significant number that will yield approximately an additional $1 Million in profit for each additional electric megawatt produced per year.

- In other words, industry data[4] shows that the profit from each megawatt of electricity generated from uprated power increases the profit yield to each electric generating corporation by approximately $1,000,000 per year.
- Therefore the data show us that by rounding up the power level increase at Millstone 3 in excess of 7%, Dominion Nuclear's Millstone Power Station Unit 3 will earn additional profits of approximately $330,000 each year until 2045.
- Stated in total dollars, the round up to a power increase in excess of 7% will yield Dominion Nuclear an extra $10,000,000 during the

[4] *Condenser Long Term Plan*, Enrico Betti, Vermont Yankee, Memo FILE UND2002-042 07; MSD 2002/002.

uprated license extension to 2045.

17. In the first place, according to the NRC document *Approved Applications for Power Uprates*[5], the NRC has never allowed a Westinghouse reactor to be licensed for a Stretched Power Uprate with a power level increase as great as that proposed for Millstone Unit 3 by Dominion Nuclear. In the second place, no other Dry Containment[6] Westinghouse reactor with a reactor power level greater than 2000 MWt has been granted a Stretched Power Uprate beyond 6.9 percent.

18. Table 1, inserted below, which is entitled Westinghouse Uprates Ranked in Ascending Order, is a list of all Westinghouse Dry Containment reactors whose thermal power exceeds 2000 MWt.

19. Table 1 ranks the Stretched Power Uprate from smallest to largest, and the NRC data provided in Table 1 shows that no other reactor of this type has ever been granted a Stretched Power Uprate in excess of seven percent like Dominion Nuclear has proposed for Millstone Power Station Unit 3.

---

5 NRC *Approved Applications for Power Uprates* http://www.nrc.gov/reactors/operating/licensing/power-uprates/approved-applications.html

6 A Dry Containment is a cylindrical structure with a hemispherical dome that relies solely on its large volume to contain the initial release of radioactive steam after an accident, and to reduce the peak accident pressure. It is a robust passive structure without any additional active mechanical means by which to mitigate immediate post accident pressure. Dry Containment does not rely upon ice or water suppression, nor is it maintained at a large sub-atmospheric pressure in order to reduce the peak accident pressure.

Westinghouse Uprates Ranked in Ascending Order

Name	Initial power	Power Uprate %	Current Power
Indian Point 2	2758	1.4	2797
Commanche Peak 1	3425	1.4	3473
Commanche Peak 2	3425	1.4	3473
STP 1	3800	1.4	3853
STP 2	3800	1.4	3853
Diablo Canyon 1	3338	2	3405
Diablo Canyon 2	3338	2	3405
Salem 1	3411	3.4	3527
Salem 2	3411	3.4	3527
Robinson 2	2300	4.5	2403
Shearon Harris	2775	4.5	2900
Vogtle 1	3411	4.5	3564
Vogtle 2	3411	4.5	3564
Wolf Creek	3411	4.5	3564
Turkey Point 3	2200	4.5	2300
Turkey Point 4	2200	4.5	2300
Callaway	3565	4.5	3725
Braidwood 1	3411	5	3581
Braidwood 2	3411	5	3581
Byron 1	3411	5	3581
Byron 2	3411	5	3581
Farley 1	2652	5	2785
Farley 2	2652	5	2785
Indian Point 3	3025	6.2	3213
Seabrook	3411	6.9	3646
Millstone 3	3411	7.01	3650

Table 1

20. Contention 2: The current application by Dominion Nuclear fails to meet the NRC's second criteria for a Stretched Power Uprate application, because the Millstone Power Station Unit 3 already had its design margins dramatically reduced.

21. According to the NRC, achieving a Stretch Power Uprate "...**depends on the operating margins included in the design of a particular plant.**"[7] [emphasis added] Dominion has stated that since the Millstone Power Station Unit 3 application "...does not involve major plant modifications, it is considered to be a SPU". Dominion has erroneously neglected to consider the significant reduction in structural **operating margins** already in place at Millstone Unit 3 prior to its application for a power uprate.

22. The Millstone Power Station Unit 3 Containment structure and its requisite systems have already been "stretched" by previous changes to its design basis when the Containment was converted from Sub-Atmospheric Containment to Dry Containment more than a decade ago. I believe that the proposed changes to Containment systems and structures that have already been reanalyzed and fine tuned once over a decade ago constitutes a dramatic decrease in "...the **operating margins** included in the design of a particular plant."

23. The Containment is the safety related building, which houses the nuclear reactor. As such, it "contains", or in other words collects, the steam and

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[7] NRC *Approved Applications for Power Uprates* http://www.nrc.gov/reactors/operating/licensing/power-uprates/approved-applications.html

radioactive material that may be released from the reactor after an accident. Please see the photo below of the inside of the Millstone Power Station Unit 3 Containment during initial fuel load in 1986.

24. As the Northeast Utilities lead licensing engineer on Millstone Power Station Unit 3 during the 1970s, I was responsible for coordinating all of the analysis for the PSAR (Preliminary Safety Analysis Report), which formed the original design basis of the Millstone Power Station Unit 3 including its Containment. This interface was among Millstone's structural mechanical, electrical, construction, and operations personnel as well as the architect Stone & Webster and the NSSS vendor Westinghouse. Millstone Power Station Unit 3 was originally designed to be "Sub-Atmospheric Containment." [In this instance my testimony is that of a fact witness[8] in addition to my overall testimony as an expert witness in this Declaration.]

25. The unique design approach of the Sub-Atmospheric Containment maintained the pressure inside the Containment at a "negative pressure" with respect to the atmosphere. Thus the difference between the pressure outside the Containment and inside the Containment (pressure differential) was approximately four pounds. Speaking as an expert witness nuclear engineer, this pressure

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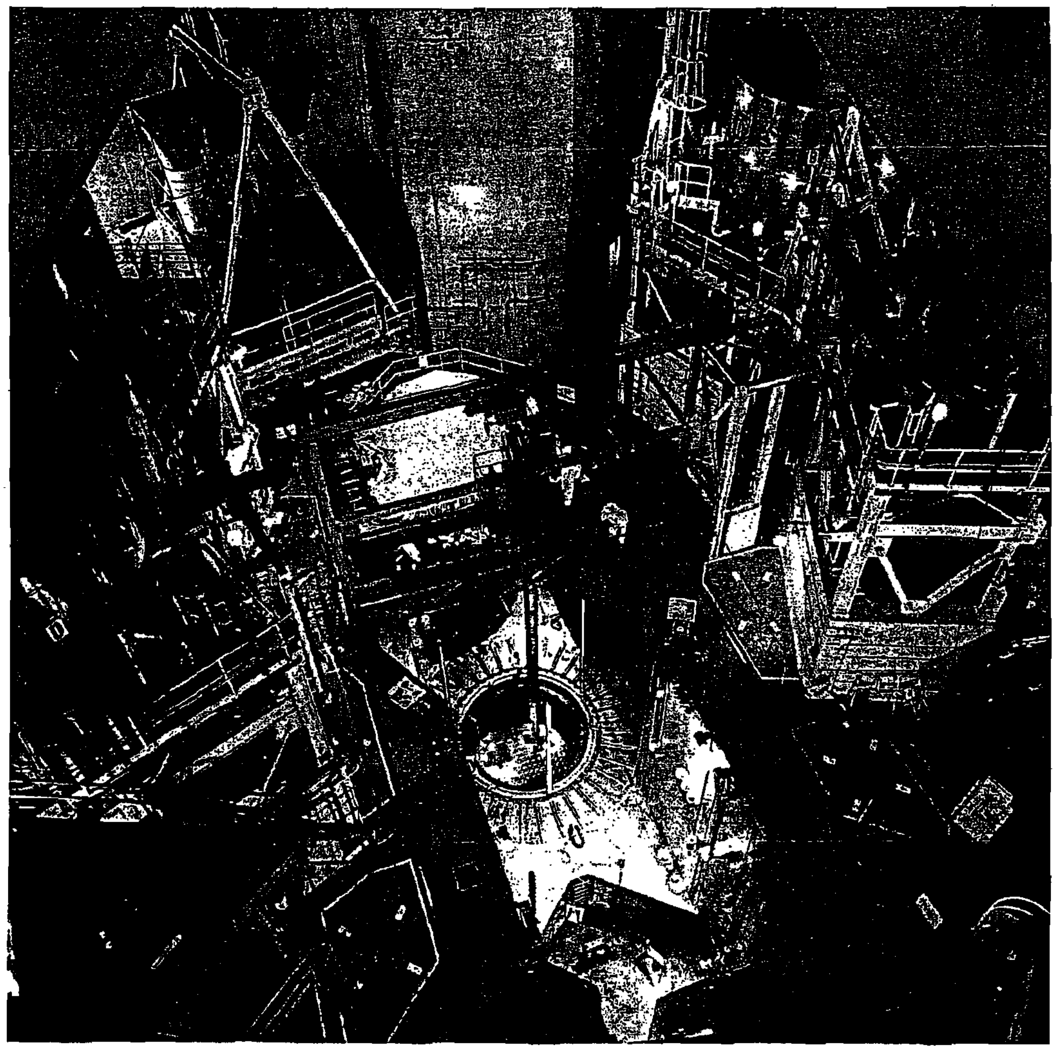
[8] According to the Department of Justice United States Attorneys' Manual Title 3, Chapter 3-19.111 An expert witness qualifies as an expert by knowledge, skill, experience, training or education, and may testify in the form of an opinion or otherwise. (See Federal Rules of Evidence, Rules 702 and 703). The testimony must cover more than a mere recitation of facts. It should involve opinions on hypothetical situations, diagnoses, analyses of facts, drawing of conclusions, etc., all which involve technical thought or effort independent of mere facts. And according to Chapter 3-19.112 Fact Witness A fact witness is a person whose testimony consists of the recitation of facts and/or events, as opposed to an expert witness, whose testimony consists of the presentation of an opinion, a diagnosis, etc http://www.usdoj.gov/usao/eousa/foia_reading_room/usam/title3/19musa.htm#3-19.111

differential is quite dramatic for a structure of this size. According to the NRC Sourcebook[9], page 4-26,paragraph B, Sub-atmospheric Containment, Millstone Unit 3 was the only Westinghouse four-loop plant in the nation to have Sub-Atmospheric Containment.



26. Due to critical engineering and operations concerns during my employment as

[9] NRC Sourcebook, page 4-26, paragraph B

the lead licensing engineer for Northeast Utilities on Millstone Power Station Unit 3, both the engineering and operations staff at Northeast Utilities (NU) expressed sincere regret as early as 1975 regarding NU's decision to design and build this unique Sub-Atmospheric Containment.

27. Critical issues of concern to both the engineering and operations staff regarding the Sub-Atmospheric Containment were:

A. The operations staff working within the Containment was repeatedly subjected to the adverse effects of the high temperature and low oxygen.

B. The small size of the Containment Building severely limited space for equipment and also complicated accident analysis.

C. Significant construction problems relating to the placement of concrete and rebar were caused by the Containment's small size.

D. Minimal analytical data regarding the long-term strength of the building's concrete and its continual exposure to the combination of high temperatures, low pressure, and low specific humidity within the sub-atmospheric Containment as it aged lead to doubts and questions regarding the strength of this critical safety-related structure in the event of a nuclear accident.

28. Despite these major concerns, NU decided in 1976 to continue with the licensing process for Millstone Unit 3 as a Sub-atmospheric Containment rather than risk delaying the license by changing the design. At the same time, the company made the strategic decision to modify Millstone Unit 3's license to

operate, by converting the Containment to a standard "Dry" Containment, but only after the nuclear power plant became operational because it is easier to amend a power plant license after a plant is operational.

29. Millstone Power Station Unit 3 began generating power in 1986, and at that time had Sub-Atmospheric Containment. However, Millstone Unit 3's original design basis with its one-of-a-kind four loop Sub-Atmospheric Containment was modified after it became operational in 1986.

30. The purpose of this one-of-a-kind four loop Sub-Atmospheric Containment was to lower peak design pressure[10] in case of a nuclear accident and to rapidly reduce out-leakage[11] after an accident.

   A. More specifically, the Containment Building is designed to capture steam, energy, and radiation after an accident. In order to capture this post-accident energy, the Containment pressure increases. Thus, Containment Buildings are designed to specific pressure levels that must be considered during all power level design changes.

   B. At Millstone Unit 3 the 1975 initial peak Containment design pressure was 39.4 psig[12].

   C. However, prior to Millstone Unit 3's start-up[13], NU reanalyzed the peak pressure and dropped it to 36.1 psig.

   D. Then on February 26, 1990, NU applied to modify the Millstone Power

[10] Maximum pressure inside the Containment after a design basis accident
[11] Leakage out of the Containment
[12] psig - pounds per square inch, gauge
[13] Amendment 17 to FSAR

Station Unit 3 license by changing the design basis pressure of the Containment from 9.8 psia to 14.0 psia[14].

31. When NU applied for the 1990 license change, it claimed that the sole basis for the change was to reduce the risk of injury to operations personnel who struggled to work at the reduced pressures inside this unique Containment. Such an environment is roughly equivalent to working at the top of the Grand Teton Mountains in temperatures in excess of 100 degrees.

    A. On page 2 of the initial application, NU stated, "... very little is known about the health effects of people working in high-temperature, low pressure environments."

    B. While it is true that this was indeed a staff concern dating back to 1975, it was only ONE of other equally important concerns.

    C. Another major staff concern was the fact that the Containment concrete is being exposed to these very same conditions and there is no data to review regarding the ability of concrete to withstand such a unique high-temperature low-pressure environment. Disturbingly, NU was silent on this major concern throughout its application to modify its license and convert the Sub-Atmospheric Containment to Dry Containment.

32. These changes to the design of Millstone Unit 3's one-of-a-kind Containment actually changed the design basis for the plant.

    A. From the time the initial PSAR was filed with the NRC, the peak accident pressure of Millstone Unit 3 was repeatedly *fine tuned* by NU.

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[14] psia - pounds per square inch, absolute

B. From a nuclear engineering standpoint, the critical concern in my mind is that each time a new Containment pressure analysis was derived, NU applied less conservative assumptions in order to achieve more operational flexibility and decidedly increasing public exposure to radiation if there were an accident.

C. In order to accomplish the 1990 modification of Millstone Unit 3, NU changed numerous design criteria and further reduced design margins by taking further credits for systems that were in the original accident scenario design basis.

33. On page 5 of the application to increase Millstone Unit 3's Containment pressure, Northeast Utilities acknowledged that these modifications to the original design "…constitute an Unreviewed Safety Question."[15]

A. In this February 26, 1990 application to the NRC, NU requested to increase the design basis for the normal pressure inside the Containment from 9.8 psia to 14.0 psia, which resulted in the increase of the post-accident peak Containment pressure from 36.0 to 38.57 psig.

B. Since Millstone Unit 3 was originally designed with this unique Sub-Atmospheric Containment Design, in the event of an accident the Containment was designed to leak radiation to the environment for only an hour until it was able to drop the pressure back down and once again

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[15] An unreviewed safety question means a change which involves any of the following: (1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; (2) A possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) The margin of safety as defined in the basis for any technical safety requirement is reduced. *http://www.nuclearglossary.com*

contain any radiation releases inside the Containment Building.

C. The 1990 modifications changed the ability of the Containment Building to release radiation for only an hour and instead allowed the Containment to leak at 0.65 weight percent per day after an accident.

D. Bypass leakage was also increased from 0.01 to 0.042 weight percent per day as a result of the change, and the modification to the Containment pressure increased the calculated exposure to a person at the Exclusion Area Boundary from 16.8 rem to 19.5 rem.

34. Contention 3: Earlier in this Declaration, I also mentioned that the Millstone Power Station Unit 3 Containment has what is considered a *small* Containment. To illustrate the fact that Millstone Unit 3's Containment is small in comparison to other Westinghouse designed nuclear reactors, I evaluated data from the publicly available "NRC Sourcebook" and compiled information regarding 25 Westinghouse Reactors, which all have "Dry" Atmospheric Containment[16].

35. Table 2, inserted below, shows, in ascending order by size, the free Containment volume (in millions of cubic feet) of these 25 Westinghouse Reactors.

A. The Containment for Millstone Unit 3 clearly stands out as one of the smallest such Containment Buildings in the country.

[16] Since they are not comparable with Dominion Nuclear's Millstone Power Station Unit 3, I have not included the Westinghouse Reactors with Ice Containments, or several three-loop Reactors with Sub-Atmospheric Containment in the compilation. Also, not included for the same reason are decommissioned reactors and reactors whose thermal power is less than 2000 MWt.

B. For that matter, the only nuclear power plants with a Reactor Containment that is smaller than Millstone Power Station Unit 3 have power outputs that are 800 to 1200 MWt less than the power output of Millstone Unit 3 *prior to the Dominion's proposed uprate.*

C. Moreover, of the 11 identical 3411 MWt Westinghouse four-loop Reactors, Millstone is smaller by as much as half a million cubic feet.

36. The ratio of the initial licensed power level to the Containment Volume at each of the same 25 nuclear reactors is clearly shown in Table 3. This ratio comparison is the real indicator of Millstone Unit 3's small Containment. By applying these ratio criteria in comparison with all 25 reactors, Table 3 clearly shows that Millstone Power Station Unit 3 has the smallest Power to Volume ratio of any Dry Containment Westinghouse reactor in the nation.

37. Dominion Nuclear's proposed 7+% power increase to Millstone Power Station Unit 3 widens even further the size gap between Millstone Unit 3 and the other reactors, thus making Millstone Power Station Unit 3's Containment even "smaller" in comparison to every other Dry Containment Westinghouse reactor in the country.

38. Table 4 shows how the initial licensed power levels of all 25 reactors adjusted as a result of NRC approved "stretch" increases.

A. Accordingly, I have adjusted the power level number for Millstone Unit 3 in order to reflect the amount proposed by Dominion Nuclear's application to uprate Millstone 3's power.

Ascending Comparison of Containment Volumes

Name	Volume xE6	Initial power
Turkey Point 3	1.55	2200
Turkey Point 4	1.55	2200
Farley 1	2.03	2652
Farley 2	2.03	2652
Robinson 2	2.1	2300
Millstone 3	2.35	3411
Shearon Harris	2.5	2775
Wolf Creek	2.5	3411
Callaway	2.5	3565
Indian Point 2	2.6	2758
Indian Point 3	2.6	3025
Salem 1	2.6	3411
Salem 2	2.6	3411
Vogtle 1	2.7	3411
Vogtle 2	2.7	3411
Seabrook	2.7	3411
Diablo Canyon 1	2.83	3338
Diablo Canyon 2	2.83	3338
Braidwood 1	2.9	3411
Braidwood 2	2.9	3411
Byron 1	2.9	3411
Byron 2	2.9	3411
Commanche Peak 1	2.98	3425
Commanche Peak 2	2.98	3425
STP 1	3.3	3800
STP 2	3.3	3800

Table 2

Containment Volume Compared to Initial Power

Name	Volume xE6	Initial power	Initial Power/ Volume
Indian Point 2	2.6	2758	1,060.8
Robinson 2	2.1	2300	1,095.2
Shearon Harris	2.5	2775	1,110
Commanche Peak 1	2.98	3425	1,149.3
Commanche Peak 2	2.98	3425	1,149.3
STP 1	3.3	3800	1,151.5
STP 2	3.3	3800	1,151.5
Indian Point 3	2.6	3025	1,163.5
Braidwood 1	2.9	3411	1,176.2
Braidwood 2	2.9	3411	1,176.2
Byron 1	2.9	3411	1,176.2
Byron 2	2.9	3411	1,176.2
Diablo Canyon 1	2.83	3338	1,179.5
Diablo Canyon 2	2.83	3338	1,179.5
Vogtle 1	2.7	3411	1,263.3
Vogtle 2	2.7	3411	1,263.3
Seabrook	2.7	3411	1,263.3
Farley 1	2.03	2652	1,306.4
Farley 2	2.03	2652	1,306.4
Salem 1	2.6	3411	1,311.9
Salem 2	2.6	3411	1,311.9
Wolf Creek	2.5	3411	1,364.4
Turkey Point 3	1.55	2200	1,419.4
Turkey Point 4	1.55	2200	1,419.4
Callaway	2.5	3565	1426
Millstone 3	2.38	3411	1,433.2

Table 3

Containment Volume Compared to Uprate License Power

Name	Volume xE6	Initial power	Power Uprate %	Current Power	Current Power/V
Indian Point 2	2.6	2758	1.4	2797	1,075.76923
Robinson 2	2.1	2300	4.5	2403	1,144.28571
Shearon Harris	2.5	2775	4.5	2900	1,160
Commanche Peak 1	2.98	3425	1.4	3473	1,165.43624
Commanche Peak 2	2.98	3425	1.4	3473	1,165.43624
STP 1	3.3	3800	1.4	3853	1,167.57576
STP 2	3.3	3800	1.4	3853	1,167.57576
Diablo Canyon 1	2.83	3338	2	3405	1,203.18021
Diablo Canyon 2	2.83	3338	2	3405	1,203.18021
Braidwood 1	2.9	3411	5	3581	1,234.82759
Braidwood 2	2.9	3411	5	3581	1,234.82759
Byron 1	2.9	3411	5	3581	1,234.82759
Byron 2	2.9	3411	5	3581	1,234.82759
Indian Point 3	2.6	3025	6.2	3213	1,235.76923
Vogtle 1	2.7	3411	6.2	3564	1,320
Vogtle 2	2.7	3411	6.2	3564	1,320
Seabrook	2.7	3411	6.9	3646	1,350.37037
Salem 1	2.6	3411	3.4	3527	1,356.53846
Salem 2	2.6	3411	3.4	3527	1,356.53846
Farley 1	2.03	2652	5	2785	1,371.92118
Farley 2	2.03	2652	5	2785	1,371.92118
Wolf Creek	2.5	3411	4.5	3564	1,425.6
Turkey Point 3	1.55	2200	4.5	2300	1,483.87097
Turkey Point 4	1.55	2200	4.5	2300	1,483.87097
Callaway	2.5	3565	4.5	3725	1,490
Millstone 3	2.35	3411	7.01	3650	1,553.19149

Table 4

39. An examination of Table 4, inserted above, shows that the new Power to Volume ratio created by the proposed uprate indicates that Millstone Unit 3's Containment would be even "smaller" if Dominion's proposed power increase is approved.

40. A smaller Containment does not mean that the physical Containment has shrunk in size, but rather that more reactor power, and, in the case of an accident, more radioactive releases are being squeezed by volume into the same small Containment Building as a result of this proposed power increase.

41. If approved, Dominion's power increase to Millstone Unit 3 would be the largest ever power uprate approved to Millstone 3's unique Containment with the "smallest" volume ever licensed as discussed above.

42. What is the net effect of increasing the reactor power in this unique very small Sub-Atmospheric designed Containment? I believe that the proposed power increase at Millstone Power Station Unit 3 means that in the event of a nuclear accident at Unit 3, more than 7% additional energy must be absorbed into this one-of-a-kind Containment.

43. I believe that Core samples from within the Containment should be analyzed to assure that the Containment's integrity has not been jeopardized by operating Millstone Unit 3 under these conditions during the first four years of its operational life during the time period while concrete curing shrinkage is

known to occur.

44. In addition to my concerns regarding Millstone Unit 3's operation beyond its design basis due to the analytical tweaking of its one-of-a-kind Sub-Atmospheric Containment, I am also concerned about the reactor power level Dominion has applied in its new analysis in order to support the proposed increase application.

A. Specifically, Dominion Nuclear used a 7.01 percent increase as the basis for energy added to the Containment during an accident. As I have already shown in this Declaration, that 7.01 percent exceeds the NRC limits for consideration for a Stretched Power Uprate.

B. More importantly, Millstone Power Station Unit 3 already has a history of exceeding its licensed reactor power. According to the NRC Integrated Inspection Report on Millstone[17], Dominion Nuclear was cited for:

> "failure to maintain reactor core thermal power less than or equal to 3411 megawatts thermal (MGTH). Specifically, during performance of turbine overspeed protection system testing, the Unit 3 reactor's four minute power average exceeded 3479 MWTH." [Unit 3's license limit is 3411 MGTH also written MWt]

C. This higher power level, for which Dominion Nuclear was cited, is a full 2% higher than level of power Millstone Unit 3 is licensed to produce.

[17] Inspection Report on Millstone, ML 080380599, February 7, 2008 for the period 10/012007 to 12/31/2007, Pages 4, 5, 21, and 22

D. Such a power level increase would also increase the energy available in an accident scenario by the same additional two percent.

E. Given Dominion's history of exceeding its licensed power level, it is my opinion that any analysis of Millstone Unit 3's Containment should use a 9% additional power level in order to most accurately reflect the condition of this one-of-a-kind Containment to withstand any additional pressures during an accident.

45. Contention 4: In its 1990 licensing application to change its Containment pressure, NU never mentioned its staffs' previous concerns about possible stress to the Containment's concrete due to the impact of its operation at high temperatures, low pressures, and low specific humidity. While it is a well known fact throughout the industry that concrete continues to shrink for up to 30-years as it matures after being poured, I was unable to uncover any NU or Dominion studies the long term impact Millstone Unit 3's concrete Containment due to its unique high temperature, low pressure, and low specific humidity environment.

46. Since nothing about this proposed change is either simple or standard, it is therefore my professional opinion that an Extended Power Uprate (EPU) review is more appropriate than a Stretched Power Uprate (SPU) review.

47. Furthermore, the Containment analysis for Millstone Unit 3 is further complicated by the fact that for the first four years of its operation, Millstone Power Station Unit 3 operated at the high, temperature, low pressure, low specific humidity unique to its Sub-Atmospheric Containment and therefore which may have compromised the structural integrity of the concrete.

48. In addition to being the lead licensing engineer at for NU at its Millstone Unit 3 nuclear plant during the 1970s, I have also been both a vice president and the senior vice president of a company that provided goods and services to Millstone 3 during the 1980s.

A. In my capacity as an officer of the firm contracted to conduct structural analytical support to Millstone Unit 3 during its construction phase, I oversaw a group of sixty structural engineers at the Millstone Unit 3 site in 1984.

B. Engineers reported to me during the construction phase informed me of other structural problems involving Millstone Unit 3's unique Containment.

C. Due to the design of this Containment, the size and amount of rebar near major Containment penetrations created strategic geometry problems in the ability of the construction contractors to pour adequate amounts of concrete around the rebar in this tight configuration.

D. This unique Containment design placed an enormous amount of rebar in several different directions around the Containment penetrations[18], making it extraordinarily difficult for concrete to slip by the rebar. Concrete voids between the rebar were a major concern. To "solve" this problem, NU qualified a procedure for the construction workers to apply long vibrating shafts into the rebar to get the concrete to slide around the rebar and create a heterogeneous block without voids.

E. This vibration method caused the sand to separate from the concrete if applied too long, and would create voids if applied for too short of a time.

F. While the procedure was qualified and construction workers were trained in how to operate the vibrating rods, my structural engineers were concerned that there was no way to test the Containment penetrations after the concrete had hardened to assure there where no voids.

G. The complex geometry at penetrations and the presence of concrete and steel intertwined made any ultrasonic exam impossible.

H. Core drilling was, of course, impossible, as it would weaken the Containment.

I. Given the structural limitations of the original design, and given that licensing changes in 1990 modified the Containment, it is imperative that this license modification be given a more thorough investigation than what is normally provided during a *Stretch* Power Uprate approval

[18] Containment penetrations - Locations through the Containment wall where pipes like steam lines and feedwater lines enter and exit the Containment.

process.

49. Contention 5: Flow Accelerated Corrosion is another critical issue that should be considered the review of Dominion's proposed power increase application.

A. Dominion's proposed power uprate will change Millstone Power Station Unit 3's reactor coolant flow by approximately 7%.

B. It will impact the flow in and out of the reactor and the steam and condensate/feedwater flow on the secondary side of the plant will also be increased by 7%.

C. These flow increases in turn increase "Flow Accelerated Corrosion" thus causing pipes to wear out much faster.

D. This Flow Accelerated Corrosion is a non-linear phenomenon, and in my opinion is a significant risk due to the application of a 7% power increase on a plant that is already in the second-half of its engineered design life.

E. Disturbingly, in its application, Dominion did not propose hiring any new personnel at Millstone Power Station Unit 3 to deal with *flow accelerated corrosion* following the unit's proposed power uprate. This despite the fact that components will require more inspections because an uprate will cause those components to wear out much faster.

F. In general, Flow Accelerated Corrosion increases the likelihood of pipe failure.

G. Equally important, given Millstone Power Station Unit 3 exceeded licensed power less than a year ago, is the concern that pipe already worn thin by the seven percent power increase might break when power is increased further.

H. I saw no evidence that the Containment has been analyzed to withstand this increased energy.

50. I believe that Millstone Unit 3's program for assessing Flow Accelerated Corrosion in Dominion's proposed uprate of the plant fails to comply with 10 CFR50 Appendix B, XVI which states:

<u>10 CFR Appendix B to Part 50</u> – Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, XVI. Corrective Action that reads:

> "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management."

51. The power increase at Millstone Power Station Unit 3 will be accomplished by increasing the flow of water through both the primary and secondary sides of

the power plant. This increased flow through the pipes causes pipes to wear out faster by a phenomenon called Flow Accelerated Corrosion (FAC).

52. The basic two causes of FAC are erosion-corrosion of the pipe walls and cavitation- corrosion of the pipe wall. Electrolytic attack may also occur. Wall thinning from FAC is non-linear and is a local issue, caused by local geometry like Elbows and flow restrictions, local turbulence, and local metallurgical conditions (welds and impurities) in the pipe. Once local corrosion has started, changes in turbulence in the local area can intensify the corrosive attack. This localized nature of the corrosion is evident in a FAC pipe failure at the Surry plant in 1986. There a feed-water elbow had holes in one area, yet the nearby pipe wall was much less worn. Similar FAC piping failures have occurred at San Onofre in 1991 and 1993, Fort Calhoun in 1997, and Mihama in Japan in 2004. While this is an *old issue*, it has not been resolved, and instead has continued to plague the nuclear industry for more than three decades.

53. Due to the localized nature of the FAC, it is difficult to predict where and when a piping component might fail. The difficulty in developing accurate predictive models for FAC is the reason why, as recently as 2004, several workers were killed at Japan's Mihama I nuclear power plant. While prediction of what might fail is difficult, it is certain, however, to say that the rate at which piping components will wear out as a result of the proposed increase in power at Millstone 3 will exceed the 7 percent power increase due to the non-linear nature of FAC.

54. In my opinion, Dominion's application does not adequately address the guidance of NRC NUREG-1800, which requires that a FAC program address the scope, analytical tools, benchmarking of the computer model, preventative activities, what is monitored, what is inspected, trend analysis, acceptance criteria, operating experience, inspection techniques as well as data collection.

55. Furthermore, I believe Dominion's proposed License amendment for Millstone Power Station Unit provides inadequate information to determine if Millstone Nuclear Power Station Unit 3 has the management systems and staff in place to properly evaluate FAC if NRC approves Dominion's proposed power increase to the plant.

A. The application did not discuss the increases in staff necessitated in order to maintain the plant in a safe condition if the proposed power increase is approved.

B. Clearly the increase in the increased corrosion rates caused by the proposed 7% power level increase will require extra analysis, extra inspection, and extra maintenance, yet the application is silent on the need to increase Millstone Unit 3's inspection and maintenance staff.

56. Without such programmatic and staffing information, I am unable to further assess the adequacy of any actions Dominion Nuclear might have to mitigate

the consequences of Flow Accelerated Corrosion caused by the proposed power uprate at Millstone Nuclear Power Station Unit 3.

57. In conclusion: following a complete review of the evidence presented and by relying upon my nuclear safety and nuclear engineering experience in my review of the documents referenced herein above, it is my professional opinion that the issues discussed above are serious safety considerations germane to the subject of the license application in this case. Similarly after reviewing all the evidence presented, it is my professional opinion that Dominion Nuclear is ill prepared to increase the power at Millstone Nuclear Power Station Unit 3. Finally, since Dominion's proposed power increase is above NRC regulatory criteria and given the new stresses upon the one-of-a-kind formerly Sub-Atmospheric Containment, I believe that the evidence clearly shows the entire application should be given the more rigorous review of the Extended Power Uprate License Evaluation.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this day, March 15, 2008 at Burlington, Vermont.

3/15/08

Arnold Gundersen, MSNE

Dominion

Millstone 3 Stretch Power Uprate

ACRS Meetings
Fuel & Safety Analysis

July 2008

Dominion

Fuel and Safety Analysis Topics

- Fuel Design.
- Nuclear Design.
- Initial RCS Conditions.
  - Pressurizer Level.
- Safety Analysis Summary.
  - Methodologies.
  - DNBR Margin & Results.
  - RCS / SG Overpressure Results.
  - Pressurizer Overfill.
  - Design Basis Results.
- Radiological Consequences.
- PRA.

Dominion

Fuel and Nuclear Analysis Overview

- No change in fuel design.
- Core will be 100% RFA-2. There are no mixed core issues.
- SPU achieved through an increase in feed batch size.
- Reduction in peaking factor design limits to increase DNBR margin.
- Predicted end-of-life fluence has decreased because the incorporation of more recent surveillance capsule data offsets power increase.

Dominion

Fuel Design

Parameter	Current	SPU
Fuel Type	Robust Fuel Assembly (17x17 RFA-2)	Unchanged
Burnable Poison	Integral fuel burnable absorber (IFBA)	Unchanged
Blankets	Annular pellets in axial blankets	Unchanged
Maximum Enrichment	5 weight percent	Unchanged

Dominion

Nuclear Design

Parameter	Current	SPU
Core Power, MWT (7% increase)	3411	3650
Radial Peaking Factor (3% decrease)	1.70	1.65
Local Peaking Factor	2.60	Unchanged
Most Positive MTC, pcm/degree F < 70% power > 70 % power	 +5.0 0.0	Unchanged
Shutdown Margin, %	1.30	Unchanged

Dominion

Nuclear Design

Parameter	Current	SPU
Average Linear Power Density, kW/ft	5.445	5.827
Feed Fuel Batch Size	72-76 out of 193	80-84 out of 193
Burnable Absorber (IFBA) Rods	7400	8200
18 Month Cycle Effective Full Power Days	510	510
18 Month Cycle Burnup, MWD/MTU	19800	21200

Dominion

Initial Conditions Overview

- ☐ Currently analyzed for a single nominal temperature at 100% power with no margin for coastdown.
- ☐ SPU analyses performed for a 8°F nominal temperature band at 100% power and 10°F coastdown for added operational flexibility.
- ☐ SPU operation selected at the same nominal temperature as current operation.
- ☐ Modest increase in hot leg temperature will have a small impact on the life of SG tubes and other hot leg Alloy 600 components.
- ☐ Modest decrease in cold leg temperature will have a modest improvement in the life of Reactor Vessel Head penetrations and other cold leg Alloy 600 components.
- ☐ Pressurizer level chosen to balance margins for operation and for design basis transients.

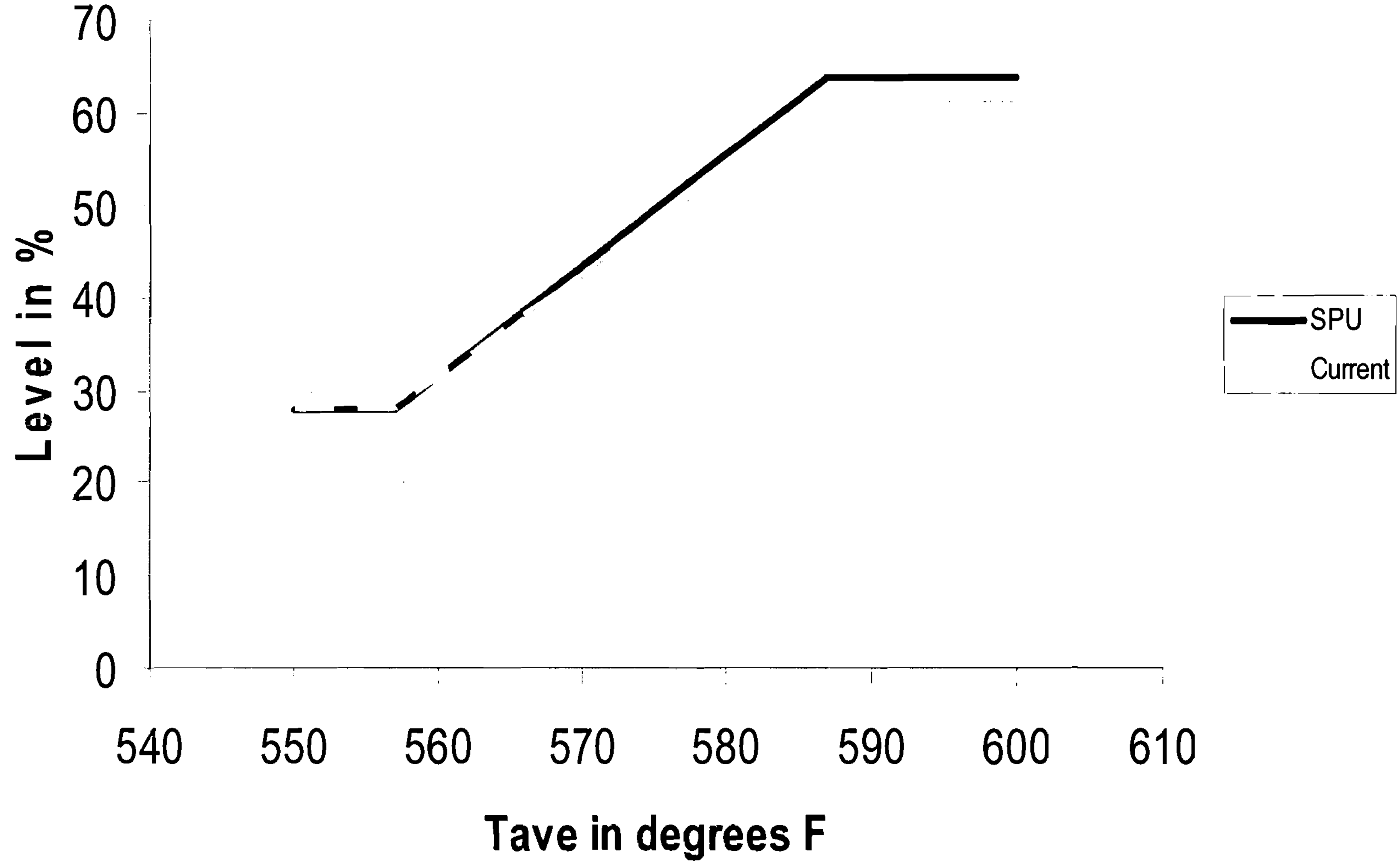
Dominion

Initial RCS Conditions

Parameter	Current Design	SPU	SPU Max Tave	SPU Min Tave	SPU Coastdown
NSSS Power, MWt	3425	3666	3666	3666	3666
Reactor power MWt	3411	3650	3650	3650	3650
Pressure, psia	2250	2250	2250	2250	2250
Hot Leg Temp, °F	618.3	617.4	622.6	615.1	605.6
Tave, °F	587.1	587.1	589.5	581.5	571.5
Cold Leg Temp, °F	555.6	556.8	556.0	547.6	537.0
Thermal Design RCS flow, gpm	363,200	NA	363,200	363,200	363,200
Min Meas. Flow, gpm	372,000	398,912 (Best Estimate)	379,200	379,200	379,200

Dominion

Pressurizer Level



Dominion

Safety Analysis Summary

- All plant specific safety analyses re-analyzed at SPU conditions.
- Significant Safety Analysis Margins Remain After SPU.
  - 11.7% DNBR margin.
  - 419 °F LB LOCA PCT margin.
  - 1007 °F SB LOCA PCT margin.
  - 3.6 psi containment pressure margin.
- Margins Achieved Through Plant Modifications.
- Methodologies Updated To Current Approved Standards.
- SPU has small impact on currently approved AST radiological analyses.
- PRA Results Show SPU Has Minimal Impact On Risk.

Dominion

Safety Analysis Methodologies

Methodology/Codes	Current	SPU
Transient Analysis	LOFTRAN	RETRAN
Thermal & Hydraulic	THINC	VIPRE-W
Rod Ejection Rod withdrawal from Subcritical	TWINKLE FACTRAN	Unchanged
SGTR	LOFTR2	Unchanged
SBLOCA	NOTRUMP	Unchanged
LBLOCA	BART/BASH	ASTRUM

Dominion

DNBR Margin Overview

- Identified as a Margin Management Issue.
- Current DNBR margin used to address Upper Plenum Anomaly.
- Modifications will address Upper Plenum Anomaly and re-establish DNBR margin
- Preliminary analyses used to establish target SPU DNBR margin.
- Final analyses resulted in small change to target SPU DNBR margin.

Dominion

DNBR Margin

Parameters	Current	SPU
DNBR Correlation	WRB-2	WRB-2M
Min. Meas. Flow, gpm	372,000	379,200
Radial Peaking Factor	1.70	1.65
Modifications	N/A	☐Elimination of auto rod withdrawal ☐Installation of electronic filter on hot leg temperature measurement ☐Decrease in power range high neutron flux setpoint from 118% to 116.5%

Dominion

DNBR Margin

Parameter	Current	SPU
DNBR Correlation	WRB-2	WRB-2M
DNBR Correlation Limit	1.17	1.14
DNBR Design Limit	Determined by statistically combining instrument and correlation uncertainties.	
Safety Analysis Limit	1.39	1.60
Generic Margin, %	Ratio of Design Limit to Safety Analysis Limit	
Penalties		
Instrumentation Bias and Rod Bow Penalties, %	Penalties for factors not addressed in VIPRE modeling.	
Rod Withdrawal from Power penalty, %	NA	3.2
Total, %	Sum of all penalties, generic and plant specific.	
Available DNBR Margin	Difference between generic margin and penalties	

Dominion

DNBR Results

Parameter	Current	SPU
Increase in FW Flow	2.31	1.88
Steam Line Break – Hot Zero Power (W-3 – Limit 1.45)	1.64	1.72
Steam Line Break – Hot Full Power	1.919	2.099
Turbine Trip	2.51	2.10
Loss of Flow	1.757	1.737
Rod Withdrawal from Subcritical Below 1st grid (W-3 - limit 1.3)	1.417 (3 RCPs)	1.306 (2 RCPs)
Rod Withdrawal at Power	1.381	1.55
Inadvertent Opening of PORV	1.584	1.874

Dominion

RCS/SG Overpressure Overview

- ☐ SPU has no significant impact on RCS/SG Overpressure events.
- ☐ Margins are essentially unchanged.

Dominion

RCS/SG Overpressure Results

	Pressure, psia			
Transient	**Current**		**SPU**	
	RCS	**SG**	**RCS**	**SG**
Limit	2750	1320	2750	1320
Turbine Trip	2731	1320	2729	1302
Bank Withdrawal at Power	Bounded by Generic Analysis	1310	Bounded by Generic Analysis	1295

Dominion

Pressurizer Overfill Overview

- Identified as a Margin Management Issue.
- Initial Pressurizer level selected to balance the margin to letdown isolation for routine reactor trips and margin to Pressurizer overfill for design basis transients.
- Current limiting event is the Inadvertent ECCS Actuation at power.
- Hardware modification proposed to significantly reduce the severity of the Pressurizer overfill rate for this event.
- Modification eliminates the Inadvertent ECCS Actuation as the limiting event. The new Pressurizer overfill limiting event changed to the CVCS malfunction event, currently considered bounded and not explicitly analyzed for Millstone Unit 3.

Dominion

Pressurizer Overfill

Parameter	Current	SPU
Pressurizer volume, cu. ft.	1800	Unchanged
Initial Pressurizer Level, %	61.5	64.0
Modifications	NA	ECCS Cold Leg Injection Valve Permissive

Dominion

Pressurizer Overfill Results

	Parameter	Current	SPU
Max Pressurizer Volume (cu. ft.)	Limit	1800	1800
	Loss of Feedwater	1061	1731
Time for Pressurizer Safety Valves to Open With Water Relief (minutes)	Inadvertent ECCS When PORVs Are Available	8.7 (water solid)	30.4 (water solid)
	Inadvertent ECCS When No PORVs Are Available	10.5	70.4
	CVCS Malfunction 1 Charging Pump	Not Analyzed	19.6
	CVCS Malfunction 2 Charging Pump		10.4

Dominion Design Basis Overview

- All design requirements are met at SPU conditions.
- In general, SPU has a small impact on the results.
- In general, safety analysis margins are essentially the same with significant margin remaining after SPU.
- The only significant change is the margin to hot leg saturation for the limiting feedwater line break.
- Reduction in margin to hot leg saturation due to the increase in decay heat associated with the SPU power level.
- Due to generic issues unrelated to SPU, initiation of two-path post-LOCA recirculation is reduced from 8-9 hours to 3-5 hours.

Dominion

Design Basis Results

Event		Limit	Current	SPU
Steam Line Break	Pins in DNB, %	0	0	0
Feedwater Line Break	Min. Margin to Hot Leg Saturation, °F	0	22	2.4
Locked Rotor	Peak Clad Temperature, °F	2700	1969	1718
	$Zr\text{-}H_2O$ reaction, %	16	0.5	0.22
	RCS Pressure, psia	3214.7	2652	2616.6
	Failed Fuel, %	7	< 6	< 7

Dominion

Rod Ejection Results

Event	Limit	Part of Cycle	HFP Current	HFP SPU	HZP Current	HZP SPU
Max Fuel Stored Energy, cal/g	200	Beginning	181.5	175.8	150.9	152.4
		End	170.6	173.7	148.9	158.3
Fuel Melt at the Hot Spot, %	10	Beginning	8.92	4.66	0.0	0.0
		End	5.71	6.86	0.0	0.0
Max Clad Average Temperature, °F	3000	Beginning	2258	2251	2624	2684
		End	2161	2224	2682	2899
Reacted Zirc, %	16	Beginning	0.90	0.91	2.65	3.01
		End	0.73	0.88	2.82	4.39

Dominion

Design Basis Results

Event		Limit	Current	SPU
SGTR	Margin to Overfill, cu. ft.	0	306	698
SBLOCA	Peak Clad Temperature, °F	2200	1009	1193 (4 inch CLB)
LBLOCA	Peak Clad Temperature, °F	2200	1974	1781 (DEGCLB)
	Local Oxidation, %	17	4.55	3.5
	Core Wide Oxidation, %	1	< 1	0.12

Dominion

Radiological Overview

- Alternate Source Term methodology submitted in 2004 and approved by the NRC in 2006.
- 2004 submittal included 6.5% power increase in anticipation of SPU.
- Alternate Source Term methodology resulted in significant increase in available radiological dose margins.
- For SPU, all events have been re-analyzed to take into account the additional 0.5% power increase.
- SPU impact on radiological analysis is small.
- For some events, changes were made to the radiological analysis assumptions to streamline the analyses and eliminate unnecessary restrictions.

Dominion

Radiological Consequences

Parameter	Current	SPU
Methodology	Alternate Source Term	Unchanged
Modifications	N/A	Automatic initiation of control building pressurized filtration mode upon receipt of CBI signal

Dominion

Radiological Results

Event		Limit	Current	SPU
Steam Line Break / Pre-Accident Spike	EAB	25	0.091	0.096
	LPZ	25	0.036	0.044
	Control Room	5.0	1.2	1.6
Steam Line Break / Concurrent Accident Spike	EAB	2.5	0.36	0.40
	LPZ	2.5	0.18	0.22
	Control Room	5.0	3.0	3.6
Locked Rotor	EAB	2.5	2.3	2.4
	LPZ	2.5	0.37	0.44
	Control Room	5.0	3.2	3.9

Dominion

Radiological Results

Event		Limit	Current	SPU
Rod Ejection / Containment Releases	EAB	6.3	0.87	0.51
	LPZ	6.3	0.48	0.26
	Control Room	5.0	0.83	1.5
Rod Ejection / Secondary Side Releases	EAB	6.3	0.12	0.12
	LPZ	6.3	0.015	0.016
	Control Room	5.0	0.053	0.051

Dominion

Radiological Results

Event		Limit	Current	SPU
SGTR / Pre-Accident Spike	EAB	25	2.1	2.2
	LPZ	25	0.18	0.20
	Control Room	5	3.0	3.3
SGTR / Concurrent Spike	EAB	2.5	0.9	1.0
	LPZ	2.5	0.09	0.2
	Control Room	5.0	1.3	1.7

Dominion

Radiological Results

Event		Limit	Current	SPU
Small Line Break Outside of Containment	EAB	2.5 TEDE	N/A	2.5
		30 Thyroid	21	N/A
		5 WB	1.5	N/A
LOCA	EAB	25	7.5	5.4
	LPZ	25	1.8	1.1
	Control Room	5.0	1.9	3.4

Dominion

Radiological Results

Event		Limit	Current	SPU
Fuel Handling Accident	EAB	6.3	2.4	2.7
	LPZ	6.3	0.13	0.15
	Control Room	5.0	4.9	4.8
Fuel Handling Accident Drop of Non-Fuel Object	Control Room	5.0	Not Analyzed	4.3

Dominion

PRA Overview

- Self assessments and Owners Group Peer Review evaluations have been performed for the Millstone 3 PRA model.
- As part of the SPU project, changes were made to address a number of the findings of these assessments.
- PRA model enhancements are continuing with the goal of full compliance with industry standards.
- No specific impacts were identified as result of SPU. Postulated impacts were assumed to determine SPU sensitivity.
- Results show SPU will have no significant impact on risk.

Dominion

PRA Results Summary

PRA Results	Current	SPU	Increase
CDF (/yr)	6.2E-6	6.6E-6	4.0E-7
LERF (/yr)	5.2E-7	5.4E-7	2.0E-8

Dominion

PRA Evaluation

- **Initiators.**
  - 10% increase in PORV challenges postulated.
  - 10% increase in Loss of Offsite postulated due to unforeseen switchyard reliability issues.
  - 10% increase in plant transients due to operating experience.
- **Success Criteria Validated at SPU Conditions.**
- **Human Reliability Analysis.**
  - 10% increase in failure postulated for feed-and-bleed.
- **SPU Modifications Have No Significant Impact.**

Dominion

Summary of PRA Changes

PRA Sensitivity	Change	Current	SPU
Consequential Small LOCA Due To Stuck Open PORV	Increased PORV Challenge Probability by 10%	7.7E-2	8.5E-2
Loss of Offsite Power (LOOP)	Increased Frequency by 10%	8.3E-3/yr	9.1E-3/yr
General Plant Transients	Increased Frequency by 10%	9.6E-1/yr	1.1E+0/yr
Operator Action To Establish Bleed and Feed	Increased Probability by 10%	4.9E-2	5.5E-2