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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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POWER UPRATES SUBCOMMITTEE

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OPEN SESSION

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WEDNESDAY, OCTOBER 5, 2011

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., J. Sam Armijo, Chairman, presiding.

SUBCOMMITTEE MEMBERS:

J. SAM ARMIJO, Chairman

SAID ABDEL-KHALIK, Member

SANJOY BANERJEE, Member

JOY REMPE, Member

WILLIAM J. SHACK, Member

JOHN D. SIEBER, Member

ACRS CONSULTANTS:

MARIO BONACA

GRAHAM WALLIS

DESIGNATED FEDERAL OFFICIAL:

PETER WEN

AGENDA

P R O C E E D I N G S

8:28 a.m.

CHAIRMAN ARMIJO: Good morning. This is a meeting of the Power Uprates Subcommittee. I'm Sam Armijo, Chairman of this subcommittee. ACRS members in attendance are Said Abdel Khalik, Bill Shack, Jack Sieber and Joy Rempe. I saw Dick Skillman around, but perhaps he's attending the other meeting. Dr. Sanjoy Banerjee will not be able to attend the morning session, but will attend this afternoon.

ACRS consultants are Dr. Mario Bonaca and Professor Graham Wallis. Peter Wen is the Designated Federal Official for this meeting. The purpose of this meeting is to review the extended power uprate request for Nine Mile Unit 2, the staff's draft safety evaluation and associated documents.

You will hear presentations from the Office of Nuclear Reactor regulation and the licensee, Nine Mile Point Nuclear Station, LLC. As shown in the agenda, some presentations will be closed in order to discuss information that is proprietary to the licensees and its contractors, pursuant to 5 U.S.C. 552(b), (c), (3) and (4). Attendance at this portion of the meeting dealing with such information will be limited to the NRC staff, licensee representatives, and its consultants and those individuals and organizations who have entered into an appropriate confidentiality agreement with them.

Consequently, we need to confirm that we have only eligible observers and participants in the room, and the closure of the public phone line for the closed portion. The subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate for deliberation by the full Committee.

The rules for participation in today's meeting have been announced as part of the notice of the meeting, previously published in the *Federal Register*. We have received no written comments or requests for time to make oral statements for members of the public regarding today's meeting.

The transcript of the meeting is being kept and will be made available, as stated, in the *Federal Register* notice. Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the subcommittee. The participants should first identify themselves and speak with sufficient clarity and volume, so that they may be readily heard.

We have several people on the phone bridge lines listening to the discussion. To preclude interruption of the meeting, the phone lines are placed on listen in mode. We will now proceed with the meeting, and I call on Ms. Louise Lunn of NRR to introduce the presenters. Louise.

MS. LUND: Thank you, good morning. I'm Louise Lund, the Deputy Director of the Division of Operator Reactor Licensing in the Office of Nuclear Reactor Regulation. I appreciate the opportunity to brief the ACRS Power Uprate Subcommittee this morning. In the interest of time, my opening remarks will be brief.

At this meeting, the NRC staff present to you the results of our very thorough safety and technical review of the licensee's application. The thoroughness of the review is supported by the fact that we had several pre-application meetings with the licensee, starting as early as September of 2008, in which the licensee scheduled an overall proposed EPU implementation plans were discussed with the NRC.

The NRC staff also performed an extensive acceptance review before initiating our detailed review of the application. We believe this helped with the efficiency and effectiveness of our review.

During the course of our review, the staff had frequent communications with the licensee, as well as two audits and numerous conference calls to discuss the EPU application and its supplemental responses to several rounds of requests for additional information, covering multiple technical disciplines.

Some of the more challenging review areas that you'll hear about today include steam dryer stress analysis, in which Nine Mile submitted its revised acoustic circuit model, thermal hydraulic stability analyses, interim methods, specifically the applicability of GE methods to expanded operating demands.

As presented in the draft safety evaluation, which was provided to ACRS a month ago, there are currently no open technical issues in the NRC staff's review of the licensee proposed extended power uprate application. We'd like to give our thanks to the ACRS staff, who assisted us with the preparations for this meeting, especially Peter Yarsky.

At this point I'd like to turn over the discussion to our NRR project manager, Rich Guzman, who will introduce the discussions. Rich.

MR. GUZMAN: Good morning. My name is Rich Guzman. I am the senior project manager in NRR, assigned to Nine Mile Point Nuclear Station. First off, I'd like to apologize. I'm having some technical difficulties in projecting the presentation onto the screen. So at this time I'd ask that you use a hard copy, the color copies that you have.

The first presentation is from the NRC staff binder, which is titled "Opening Remarks." During today's Subcommittee meeting, you will hear presentations from the Nine Mile Point Nuclear Station and the NRC staff. The objective is to provide you with sufficient information related to the details of the EP application, as well as the evaluation supporting the staff's reasonable assurance determination that public health and safety will not be endangered during the operation of this proposed EPU.

Before I cover the agenda items, I would like to go over some background information really of the staff review of the Nine Mile Point 2 EPU. On May 27th, 2009 -- well there you have my script --

(Laughter.)

MR. GUZMAN: All right.

CHAIRMAN ARMIJO: Just go ahead. Don't worry about it.

MR. GUZMAN: All right. As you see there, on May 27th, 2009, the licensee submitted its license request for Nine Mile Point 2 EPU. The increase would be 34.67 megawatts thermal, their current license thermal power, to 39.88 megawatt thermal. This would represent a 15 percent increase from their current license, and a 20 percent increase from their original license thermal power.

The staff's method of review was based on the RS 001, which is NRC's review plan for EPUs. As you know, it provides a safety evaluation template, as well as major C's that cover the multiple technical areas that the staff is to review.

There are no associated or linked licensing actions associated with this. Nine Mile previously submitted, and the staff approved two license amendments, mainly the maximum extended load line limit analysis, and the AST amendment in 2007 and 2008, respectively. Finally, there were numerous supplements to the application, responding to multiple staff RAIs. Overall, there were approximately 25 supplemental responses, which supported our draft safety evaluation.

The staff projects December 2011 to complete our review, and this would be in support of the licensee's scheduled implementation in the second quarter of 2012.

The next slide on your hard copy binder there covers the agenda items for the Subcommittee meeting. The morning will cover the fuel methods and the thermohydraulic design review areas, mainly the anticipated transient without scram, and the stability review. Then the afternoon will go into materials and the mechanical and civil engineering review, which will also include the steam dryer analysis.

Finally, at the conclusion of the meeting, as needed, we can cover any open items in preparation for a full Committee meeting. And also to note, there will be closed sessions during the latter parts of both the morning and afternoon sessions.

So if there's any proprietary information that needs to be discussed, it can be deferred over to the designated closed session for the agenda. This concludes my presentation. I would like to now turn the presentation over to the licensee, specifically Mr. Sam Belcher, who is the Senior Vice President for Operations for the Constellation Fleets.

And that said, I am going to eventually in parallel, as you guys just need, talk to the slides that is in your binder, I'll eventually get this posted on the wall. I apologize.

MR. BELCHER: Thank you. As mentioned, I am Sam Belcher. I'm the Senior Vice President of Operations for Constellation Energy Nuclear Group, and I'll be walking through a presentation. I don't know if it will make it up not the slide or not, but it is in your binder and it's titled "ACRS Subcommittee Presentation," and I'm on point 2, "Extended Power Uprate, October 5th."

I'll be walking us through a very high level overview, and then we'll get into more technical details as we move through the morning and into the afternoon. We'll start with an overview, followed by a discussion on the plant modifications necessary for the extended power uprate, anticipated transient with scram and stability discussion, and then, as mentioned the closed sessions for fuel methods, material mechanical civil engineering topics, and then steam dryer analysis also will be a closed session.

At a very high level, Nine Mile Point Unit 2 is a GE BWR 5, with a Mark II containment. Original license thermal power was 33.23 megawatts thermal. In 1995, a stretch uprate was done of 104.3 percent, which takes us to the existing license power limit of 34.67 megawatts thermal.

This amendment would take us to 39.88 megawatts thermal, with the intention to implement second quarter of next year. This is a constant pressure power uprate. Additionally, Nine Mile Point Unit 2 is not requesting any containment accident pressure credit to support ECCS positive suction head. I see some smiles there.

Also, no new fuel will be introduced as a part of this uprate. The current core and the EPU core will be GE 14 fuel consistently. Also, as mentioned previously, alternate source term has already been completed, and that was at the EPU power level as the base assumption. Also previously discussed is the maximum extended load line limit analysis, expanded operating domain as well.

Finally, the New York state ISO has reviewed and approved the full EPU power uprate, with no grid modifications being necessary. The only modifications that I would note were revenue metering type modifications for the increased output. But nothing for grid stability or anything along those lines.

The first two phases of the EPU modification have been completed, and then the third and final phase of the modification will be completed in the second quarter of 2012, consistent with the refueling outage. At this point, unless there are some questions for me, I will turn it over to Dale Goodney, who is our lead engineer, to talk in more detail around some of the plant modifications required moving forward.

MR. GOODNEY: Okay, good morning. As Sam indicated, I'm Dale Goodney. I'm with Constellation Energy and the EPU lead, engineering lead for the EPU project. I'll provide an overview of EPU plant modifications. We'll cover the general approach, the review plant parameters and modification installation time line.

We'll summarize the major plant modifications and then we'll review other Nine Mile 2 plant improvements that are being implemented at the station.

In support of the license amendment request, a series of engineering studies were performed to determine the plant's ability to operate at EPU conditions, and to identify what modifications may be needed. These studies were developed by a team of Constellation engineers, industry consultants, GE Hitachi for the nuclear steam supply system, and Sargent & Lundy for balance of plant systems.

These studies analyzed the effects of the increase in steam flow, feedwater flow, electrical power output and reactor power on various plant systems and components. As Sam mentioned, this uprate is not increasing reactor pressure. Therefore, the evaluations were performed based on the methodologies outlined in the past and pressure power uprate licensing topical report.

The analyses were all based on the target power level of 120 percent of the original license thermal power. Each study included a review of relevant operating experience, both internal and external, and were applicable to results were incorporated into these evaluations.

Another element of the engineering evaluations were the margin reviews. Design and operating margins were identified and evaluated for both NSSS and balance of plant systems, to determine if there would be adequate margin under EPU conditions.

As a result of these reviews, over 20 physical plant modifications, mostly in the balance of plant area were identified and described in the license amendment request. The primary purpose of these modifications are to (1) restore material condition, (2) install instrumentation for data collection and analysis, or (3) upgrade or replace equipment to restore design and operating margins at EPU conditions.

This next slide shows the fundamental plant process parameters that would change due to the uprate, and compares the EPU conditions to the CLPP conditions. These parameters are the primary starting point for the evaluations that I just described, and they also form the key design inputs for the modifications that were developed for the power uprate.

The next slide is the modification installation time line, and as mentioned in our earlier slides, the Phase 1 and Phase 2 implementation is completed. Those are the modifications shown in the two left-hand columns. The remainder of the modifications will be installed prior to the end of the 2012 refueling outage.

On the next slide, or next few slides, will summarize some of these modifications. It will cover basically four general categories. Feedwater and condensate, steam path, electrical I&C systems, and auxiliary support systems.

DR. WALLIS: Can I ask you about the steam dryer? This isn't the -- you know, GE has a new design of steam dryer. That's not the one, right? It's the old steam dryer, modified because the analysis says you need to do so. So you're putting in strengthening at various places and perforated plates and so on?

So just to clarify, that's the old steam dryer, strengthened because of the results of analysis?

MR. GOODNEY: That's correct, and we'll be covering those modifications --

(Simultaneous speaking.)

MR. GOODNEY: Due to the higher feedwater flow requirements, the feedwater pumps will be modified with new rotating elements, new step-up gears and modified flow control valve trim. In addition, the heater drain pumps in motors were replaced in 2010, to increase the capacity of the pumps. These changes will provide the additional flow margin required for normal, off normal and transient conditions.

Reactor recirculation runback logic is being modified to maintain scram avoidance margin following a single feedwater pump trip. This will be accomplished by initiating the runback immediately on a feedwater pump trip, increasing the runback rate of the reactor recirc flow control valve, and -- the higher feedwater pump runoff flow capacity.

In terms of the steam path, the high pressure turbine we replaced with a monoblock rotor, new diaphragms and buckets to increase the steam flow, six relief valves located on the reheat piping will be replaced with valves with a higher set pressures to increase the steam relieving capacity.

Moisture separate reheaters on the fifth and sixth point feedwater heaters, will be rerated to higher pressures, and as you mentioned earlier, the steam dryer will be modified to provide the required structural margin at the higher steam flows, and we'll provide more details of those modifications in the afternoon session.

Two electrical modifications are needed to support the higher power output. The isophase bus duct will be upgraded by installing a higher capacity cooling system, and the coolers on the main transformers will be replaced with larger coolers, to provide additional thermal margin.

Instrumentation affected by the uprate include two tech spec instrument set points, the APRM flow-biased scram, and the main steam high flow isolation. Those changes are included as part of the license amendment request. The balance of plant instrument loops are being rescaled, as required, to accommodate the higher flows, temperatures and pressures under EPU conditions.

Due to the high heat load in the turbine building, the turbine building HVAC system will be modified to install four additional area coolers near the condensate and condensate booster pumps.

The turbine building cooling, although it does have adequate margins for EPU conditions, it's going to be modified to isolate retired loads to provide additional margin, and the system will be rebalanced to ensure that we get accurate cooling to all the power-dependent loads supplied by the system.

So that completes the preview of the EPU modifications.

CHAIRMAN ARMIJO: I have a question, not about modifications, but many years ago, a number of the BWRs had stress corrosion cracking problems in their recirc piping, core repiping, and a number of them did some replacements of the original type 304 stainless steel with an improved material, 316 nuclear grade.

I didn't, don't remember. What did you do at Nine Mile 2? Are you still using the original recirc piping?

MR. INCH: Yes, the original Unit 2 was, went into service in '87, and the piping was originally, you know --

CHAIRMAN ARMIJO: You'll have to speak a little bit louder, so that the microphone --

MR. INCH: The piping at Unit 2 was originally considered as upgraded piping. I believe it's 316. I'll have to verify that.

CHAIRMAN ARMIJO: Yes, if you could. So it was built at a time. By that time, people knew this was a better way to build it, and you just happened to be at the right place at the right time.

MR. INCH: It was a safe end replacement prior to service, where they replaced the safe ends with IGSCC-resistant materials. That was all done prior to service.

CHAIRMAN ARMIJO: Okay, all right. Thank you.

MEMBER SHACK: But just on that point, you do have a number of Class D welds left.

MR. INCH: Yes.

MEMBER SHACK: At least there's a discussion of that in the --

MR. INCH: The Class D welds are the similar metal welds, the safe end to nozzle welds. So that is the --

MEMBER SHACK: But they're what, ferritic to a normal carbon steel safe end? Is that --

MR. INCH: It's the stainless steel safe end to the low alloy steel nozzle. It's the similar metal weld, and that's the -- those are the category, considered Category D welds, per 8801, Generic Letter 8801.

CHAIRMAN ARMIJO: And some of those you've done a weld overlay, repair, mechanical stress improvement?

MR. INCH: There was one indication on one of the high pressure core spray lines, that a mechanical stress improvement was done in the early 90's. We've been monitoring that since then, with no growth.

It was an indication identified in one of the feedwater nozzles, approximately ten years ago. There was an overlay done on that. Otherwise, we're not tracking any --

CHAIRMAN ARMIJO: So with the exception of those two components, it's the as-built material?

MR. INCH: Yes.

CHAIRMAN ARMIJO: Okay, thank you. And when did you start the hydrogen water chemistry?

MR. INCH: Hydrogen water chemistry was started in, it was either 2000 or 2001. We implemented hydrogen water chemistry in combination with noble metals. So it's always been a noble metals hydrogen water chemistry application.

MEMBER SIEBER: I may have I missed it, but in my review of the material that was included, that Nine Mile Point 2 has a Mark II containment, which is the upside down lightbulb or ice cream cone, similar to the Mark I containments in containment volume, but the geometry was different.

Did you analyze the containment capability, insofar as you now have approximately 20 percent over the original design stored heat acumen environment?

MR. INCH: Yes.

MEMBER SIEBER: And if so, did that consider fuel failures, cladding oxidation and so forth? How far did you go in that analysis?

MR. INCH: The design bases analyses were redone for the higher megawatt thermal.

MEMBER SIEBER: Right.

MR. INCH: And decay heat levels, and

those were performed by GE, using their design bases methods for the higher power levels, and mainly because it's -- the peak pressure is governed by the short-term blowdown, and because it's a constant pressure power uprate, the peak pressure associated with that blowdown has not changed.

MEMBER SIEBER: Right, that's true.

MR. INCH: And the long-term response --

MEMBER SIEBER: Has to be increased.

MR. INCH: Long-term response was mitigated by the suppression pool cooling systems. There's significant margin built into the original design on those systems, that the original design analyses had not credited. So by actually crediting those systems capability, we were able to maintain the suppression pool temperature effectively the same in design bases space as current.

So there really wasn't any significant change in the long-term pressure temperature profile for the --

MEMBER SIEBER: Or change at all?

MR. INCH: Effectively, yes.

MEMBER SIEBER: I have to think about that.

MEMBER SHACK: While we're at it, is this a vented Mark II? Is there a vent on this Mark II?

MR. GOODNEY: Phil.

MEMBER SIEBER: Has to be.

MR. AMWAY: My name is Phil Amway, and I'm the extended power uprate operations lead, and I maintain an active senior reactor operator's license for the facility. Nine Mile Point 2 is able to vent the containment through a path that will divert the containment out directly to the stack, using a bypass around the PTS train. We have that capability.

CHAIRMAN ARMIJO: You know, this is a little bit off the scope, but you can't help it, because of the Fukushima events. How do you test those vents? Do you ever test them or that they --

MR. AMWAY: We have performed, and again, my name is Phil Amway. We have performed walkdowns of those procedures. We have procedures in place that line up that vent path. All the materials are staged to do so. We cover it in training. We have not actually physically made the alignments, to actually vent in that mode. But it is a fairly simple mechanical arrangement that could be done.

CHAIRMAN ARMIJO: Is there a rupture disc in that design or not?

MR. AMWAY: There is no rupture disc, no.

DR. BONACA: So it's a duct. Is it the hard piping or --

MR. AMWAY: It's hard piping. We actually bypass around any duct work that would be subject to the high pressure condition.

MR. BELCHER: If I may add, I'm Sam Belcher, the Senior Vice President for Operations for Constellation Energy Group. While we have processes and procedures and training in place, based on the recent events, we are looking in detail at improvements, not only at the Nine Mile Point site, but at our other sites as well.

I think there are lessons learned that will ultimately lead to us doing things differently. But we are looking at that.

CHAIRMAN ARMIJO: Okay, continue on.

MR. GOODNEY: No problem. This final slide covers other plant improvements that the station has implemented or is planning to implement, to restore material condition, improve margin, improve equipment reliability. Some examples are replacement of the third point feedwater heaters in 2010; the standby flow control relief valve margin was improved.

Cleaning tower upgrades were implemented. New feedwater pump seals will be installed, according to the feedwater pump modifications I mentioned earlier. One of the more significant improvements, the station plans to replace all 20 jet pump inlet mixers during the 2012 refueling outage. That will restore the equipment back to the original design performance, restore core flow margin, and address operating experience relative to flow-induced vibration.

Then finally, there have been several PRA-related risk reduction improvements, consisting of procedure changes and other minor modifications. As a result of these improvements, since 2008, the core damage frequency at Nine Mile Point has been reduced by 78 percent.

So that concludes my presentation on modification overview. Pending any questions, I'll turn this over to Phil Amway, to discuss power ascension testing.

MR. AMWAY: Thank you, Dale. Again, to reiterate, my name is Phil Amway. I'm the extended power operations lead. I'll be giving two presentations this morning. The first area is for the power ascension testing program. Under this topical area, we'll discuss the preparation of the program, approach to uprated power --

MEMBER ABDEL-KHALIK: I'm sorry. Can we go back to the previous slide?

MR. AMWAY: Sure.

MEMBER ABDEL-KHALIK: You indicated that you will replace the jet pump inlet mixers?

MR. GOODNEY: That's correct.

MEMBER ABDEL-KHALIK: Could you explain more of the rationale for that?

MR. GOODNEY: I apologize. Excuse me. You'd like to know the rationale behind replacing the jet pump inlet mixers, and whether that will impact the core flow measurement instrumentation.

MR. INCH: Oh, the jet pumps become fouled over years of operation, from a mechanism that I don't fully understand. But they call it a zeta potential, where you get deposits that affect the efficiency of the jet pumps. At Nine Mile, that's been occurring for several years.

There's several options available. Ultra high pressure cleaning is an option that was considered, and but there's essentially the new mixers we're putting in are identical to the original design. So it restores the jet pumps to a new condition, and so that's what is going on.

So therefore, the instrumentation is not affected. It just basically restores the performance to the original performance.

MEMBER ABDEL-KHALIK: But the relation between the driver flow, the jet pumps and the actual core flow will change as a result of that modification; is that correct?

MR. INCH: It will be restored to the design bases, drive flow design basis and ratio, but it's not a change to the design. So operational procedures, every refuel outage, do a new baseline for where those jet pumps are, to establish the correlation between dry flow and core flow.

Then that's put into the instrumentation, and it's all proceduralized, because it does change over time. So the procedures account for that change.

MEMBER ABDEL-KHALIK: So that when the operators, the current procedures for knowing where they are on the power flow map, they use the driver flow or they use the direct total core flow, as measured from the 20 jet pumps?

MR. INCH: Operations has core flow. Phil.

MR. AMWAY: We use total core flow direct indication.

MR. INCH: When we plot our --

MR. AMWAY: Yes.

MEMBER ABDEL-KHALIK: But in that process, you use the relationship between the driver flow and the core flow, which you say you empirically calibrate every outage?

MR. INCH: Yes.

MR. AMWAY: Every outage, and it's part of our start-up test program as well. We will do the core flow calibration, which will calibrate the dry flow to the jet pump flow.

MEMBER ABDEL-KHALIK: How much has that calibration changed since the jet pump inlet mixers were replaced? Oh, you have no idea.

(Simultaneous speaking.)

MEMBER ABDEL-KHALIK: How much has that changed over the years, as a result of fouling?

MR. AMWAY: It has changed gradually over the years. It's actually a reactor engineering procedure. It's done at the conclusion of each outage, once we get the full rated power. We'll do that procedure and the trend has been, the acceptance criteria of that procedure is as long as the calibration is within two percent, no additional action is required.

It's about every third outage we actually have to go in and make a change, and adjust the gains on the dry flow to match the core flow.

MEMBER ABDEL-KHALIK: So the allowable deviation between the two flow indications is two percent you said?

MR. AMWAY: Two percent.

MEMBER REMPE: Do you expect the new plants that you're replacing to have similar fouling characteristics?

MR. INCH: The new mixers, we hope to be able to manage the fouling a little bit better. The plan is to they'll have a coating on them, that will resist fouling. It's not 100 percent, but it should reduce the rate of fouling.

MEMBER SHACK: What is this magic coating?

MR. INCH: That's proprietary. I can't -- we can talk about that in closed session, I guess.

MEMBER SHACK: Yes.

CHAIRMAN ARMIJO: You may want to do that.

MEMBER SHACK: But otherwise its geometry is identical, with the exception of a coating to surface treatment of some sort, to minimize the fouling rate.

MR. INCH: To try and minimize future fouling, yes, right.

MEMBER ABDEL-KHALIK: But generally th trend is that the actual core flow will likely be less than the indicated core flow. Is that the -- or is it the other way around?

MR. INCH: Right now, the design M ration for the jet pump mixers is -- at EPU, it will be about 2.28.

MEMBER ABDEL-KHALIK: No, no, no. I'm asking about the effect of fouling, and you're allowing a two percent deviation between or two percent variation on the calibration, in the empirical calibration between driver flow and actual core flow.

MR. INCH: I believe what Phil's referring to is just the instrumentation tolerances --

MEMBER ABDEL-KHALIK: That's right.

MR. INCH: That are built into the design bases. The fouling occurs over very long periods of time, over many years in the cycle.

It's a very gradual process, and the frequency for the calibrations will maintain and ensure that the relationship between dry flow and core flow is accurate to within the design tolerances at all times. But it's not something that occurs suddenly. I'm not sure if I'm --

MEMBER ABDEL-KHALIK: I'm just trying to

understand the direction of the trend.

MR. INCH: Okay.

MEMBER ABDEL-KHALIK: Okay, and the time line associated with that trend. You say that you need to do that roughly every third outage?

MR. AMWAY: About every third cycle. We actually have enough mismatch between the two measurements that we actually adjust the gains of the dry flow.

MEMBER ABDEL-KHALIK: So that sort of gives you an indication of how quickly the core flow is being impacted as a result of fouling? Or is it just drift?

MR. AMWAY: It's just looking at the total -- I mean some of that could be drift, some of that could be fouling. I'm just looking at the total measurements of drive to driven flow when we do that procedure. It's not really looking at specific factors that may input to that deviation of two percent.

MR. INCH: I can give you a feel for some of the numbers. The original, when we first started the plant up in '87, the original calibration for dry flow was approximately 41,000 GPM for a rated core flow, which is 108-1/2, 108-1/2 million pounds per hour. That's the relationship. So M ratio, drive to driven of approximately 2.4 or 2.5, I believe, at OLTP.

Now, for to achieve 108, the rated core flow, 108 million, we need 46,000 GPM drive flow, and that's occurred over 22 years. That's the -- it's a gradual change. It is affected by the stretch uprate in effect. When we did the stretch uprate in 1995, that was the original design, five percent uprate.

Then fuel type has some effect on it. C Core DP has some effect on it. So as the, you get some of the newer fuel design, you have a higher two phase pressure drop. So that, the jet pumps have to work a little bit harder. So that's in the mix with some of those relationship changes.

CHAIRMAN ARMIJO: So do you expect the fouling rate to be greater with the higher flows at EPU?

MR. INCH: Again, our flows are really not higher. They're the design flows. So the fouling rates really shouldn't change from what it's been historically. The change for power uprate, changes to the core DP are slightly. So we need basically it's a 1.9 percent effect on the dry flow relationship.

MEMBER ABDEL-KHALIK: But if the fouling

is some sort of a deposition of some, let's say an iron oxide or some other material on those mixers, you're putting a lot more water with all that same material through, over a given period of time. So I would expect the fouling to be faster.

MR. INCH: It's important that -- we're really not putting any more -- the core flow stays the same, and the dry flow really is the same. Now what we're doing is put it back to the original core relationship and efficiency of the jet pump. So the rate of fouling should be equivalent.

MEMBER SHACK: And with your magic coating, less.

MR. INCH: Well, hopefully yes.

CHAIRMAN ARMIJO: Okay.

DR. BONACA: Just one question I had regarding the vent. Do you have that venting procedures?

MR. AMWAY: Yes, we do have venting procedures. It's part of our emergency operating procedures, support procedures. But we do have those in place.

DR. BONACA: All right, thank you.

MR. AMWAY: All right. If I may continue on under power ascension testing, I'm on Slide 17.

The topical areas that I want to address under power sensitive testing are preparation, approach, schedule, the test plan and the acceptance criteria and actions.

Under the preparations, our objective of the start-up test program is to demonstrate satisfactory equipment performance, ensure we have a careful, monitored approach to EPU power level, and to ensure that we meet established requirements.

We define the roles and responsibilities in the master start-up test procedure. We have used industry benchmarking to confirm that our test program matches similarly uprated BWRs, and also that our test plan and implementing test procedure development is consistent with industry standards.

We will also perform operator training on the power ascension test program, including the test procedures that will be performed. The approach is similar to that used for other BWRs that have implemented extended power uprate, and that is incremental testing approach. We collect baseline data at 75, 90, 95 and 100 percent of current licensed thermal power.

Once we rise above the 100 percent current licensed thermal power, we will perform data acquisition and incremental steps of one percent, and an analysis of two and a half percent. Every five percent plateau is a major testing window that includes the active as well as the passive testing, and there is an NRC data review with those five percent incremental levels.

DR. WALLIS: And in doing this, you have instrumented the steam lines? They go and look at the fluctuations and that sort of thing.

MR. AMWAY: That is correct. That's part of the ascension program. Power ascension testing approach for Nine Mile 2 does not include large transient testing. The basis for that is the substantial industry operating experience from uprated plants that have experienced large transient post-EPU implementation, and also Nine Mile Point specific data for large transients that have occurred at the station at 104.3, which is the uprate, stretch uprate power level.

We were able to use that data to accurately project, using the analytical methods that are available today, such that we fully understand how the plant will respond post-uprate for large transients.

MEMBER ABDEL-KHALIK: If you'll go back again to the previous slide, please.

MR. AMWAY: Sure.

MEMBER ABDEL-KHALIK: Have you had any experience with the SRV leaks?

MR. AMWAY: With SRV leaks, and I may ask George to provide additional information here, but recently our SRV leakage has been very good.

We had problems, I'll say in the mid-90's, with SRV leakage that was indicated by rising suppression pool temperatures, and the frequency at which we had to place suppression pool cooling in service, to maintain pool temperatures just at normal power.

That has not been the experience that I've seen in the power plant for the last 10-12 years, and I would say that we're not seeing it in tail pipe temperatures, or the suppression pool temperatures. George, do you have anything additional to add to that?

MR. INCH: No. I might add that when we stopped doing the steam flow surveillance tests, actually opening the SRVs and closing them, which challenges the receding, the SRV leakage has gone away as a problem. So it's been very effective.

MEMBER ABDEL-KHALIK: And post-outage testing of the SRVs, they meet the specs, as far as set point?

MR. INCH: Oh yes. I'm not prepared right now to go into any of those details, but yes. They change them out in accordance with a rotation plan. They are tested at offsite. I believe they're sent offsite and tested each outage. If you need more details, I would have to come back.

MEMBER ABDEL-KHALIK: I'm just trying to get just step by step here. So this emphasis on instrumentation is in primarily during the power upgrade, is in primarily concerns with regard to the steam dryers.

MR. AMWAY: It's primarily with the steam dryer, but it also includes balance of plant piping, because of the increased steam flows and feed flows, and we will be monitoring that vibration in those same increments on the way out.

MEMBER ABDEL-KHALIK: Is there any concern about increased leakage from the SRVs, as a result of the increased steam flow, and the potential acoustic coupling associated with the SRVs?

MR. INCH: No. We've looked at that fairly significantly with our instrumenting. The main steam lines with accelerometers in the vicinity of SRVs, to make sure that there's no coupling.

Analytically, we're not seeing any, haven't seen and are not predicting to see any issues with this --

MEMBER ABDEL-KHALIK: How would your instrumentation tell you whether or not you have increased leakage out of the SRVs, as a result of the --

MR. INCH: Well have, there's tailpipe instrumentation to tell us if it's leaking. They know.

MR. AMWAY: That's correct.

CHAIRMAN ARMIJO: And you have what kind of measurement?

MR. AMWAY: It's a temperature measurement, right on the tailpiping.

MEMBER ABDEL-KHALIK: Do you have a two-stage or a three-stage SRV?

MR. GOODNEY: Are you referring to Target Rock?

MEMBER ABDEL-KHALIK: Right.

MR. GOODNEY: No, we don't have Target Rock.

MEMBER ABDEL-KHALIK: You don't have Target Rock. So what kind of SRVs do you have?

MR. INCH: They're Dikkers.

MR. GOODNEY: Dikkers.

MEMBER ABDEL-KHALIK: And they're instrumented only by measuring temperature in the tailpipe?

MR. INCH: I believe so. I would have to verify that.

MEMBER ABDEL-KHALIK: Okay.

MR. AMWAY: As far as what we see in the control room, the temperature is our primary indicator, the tailpipe temperature. You know, they're also fitted with acoustic monitors that would tell you to actually lift it.

MR. INCH: That's what I think, yes.

MR. AMWAY: But for the leaking, it's just the thermocouples on the tailpipe.

MEMBER ABDEL-KHALIK: All right, thank you.

MR. AMWAY: You're welcome. I'm up to Slide 21 now, the power ascension testing schedule. Data collection in one percent intervals, data evaluation, two and a half percent intervals, and then the major testing plateaus at five percent intervals.

That five percent test plateau includes passive data collection, which includes the vibration monitoring, radiation monitoring and plant parameter monitoring. The active testing is associated with the stability of the various pressure control and feedwater level control systems.

We will perform data analysis of both the active and passive testing, and then that data will be reviewed by station management through the Plant Operations Review Committee, and then submitted to the NRC for review.

This next slide just shows an overview of the various tests that are performed at the power levels. Across the top of the slide, you'll see the percent for current license thermal power, and the intervals that we're doing the testing. Those power levels in red, that are red highlighted, are those associated with the five percent test plateaus at which data will be transmitted to the NRC for review.

Then all the X's in the box along the left column, you see the various tests that are performed, and the X's designate how often they're performed, at what power levels. Those indicated in the blue shading are those that also have one percent data collection requirements.

For power ascension testing acceptance criteria, there's two major levels. The Level 1 acceptance criteria is associated with a limit associated with plant safety.

If we reach a Level 1 criteria, we abort from the test. We reduce power level to a known safe condition, and that would be the power level at which the Level 1 criterion was verified met, and that we will use our corrective action program to evaluate the condition and determine required actions. Then we will repeat the testing, to verify Level 1 criterion is satisfied and document results.

Level 2 is the limit associated with plant or equipment performance that does not meet design expectations, but is not immediately adverse to plant safety. We will perform similar actions, in the terms of we will place the test on hold, and if needed, lower power, and then again use the corrective action program to determine the requirements.

In the Level 2, we may make a determination that the data is satisfactory and that we can continue testing. In either event, we will have to also document the results as a test exception. The final acceptance criterion that we may encounter following the start-up program includes things such as technical specification required surveillance tests.

They have their own acceptance criteria, based on tech spec limits, and if we reach one of those limits, we would follow the actions in the procedures and the plant technical specifications.

MEMBER ABDEL-KHALIK: Can you give me an example of a Level 2 acceptance criterion, and an example of a Level 1 acceptance criterion that you can immediately indicate with a test?

MR. AMWAY: Yes, I can. For a Level 1, for the control systems which are tuned, we will introduce step changes, for example, reactor pressure. We would expect that the system will respond in a quarter wave damping fashion, so that any oscillation is quickly dampened, then maintain steady control of the plant pressure.

If for some reason that we don't meet the quarter wave damping, but the oscillations are convergent, such that you reach a final steady value and pressure, that would be an example of not meeting Level 2. It doesn't meet the design expectation, that we should be able to meet the quarter wave damping.

If we did that same step change, and we got a divergent behavior in the oscillations, which means they did not dampen out and in fact got worse, then we would have actions in the procedure for how to deal with that. That would be a Level 1 criterion, and we would abort that test, to figure out why that happened.

MEMBER ABDEL-KHALIK: Okay.

DR. WALLIS: Are these limit curves evaluated directly by computer, or does someone have to look at this and look at that, and compare them?

MR. AMWAY: We will have both guidance in the procedure for what the operators can look at directly by plant instrumentation. But there will also be backup confirmatory database reviews of the parameters using computers.

DR. WALLIS: So there will be something set in place, so that when something unusual happens, it's right there on the computer or there's a warning or something?

MR. AMWAY: That is correct.

DR. WALLIS: You don't have to wait for someone to look at something?

MR. AMWAY: That's exactly right. If there's no further questions on the power ascension test program, I'd like to proceed on to the long-term stability, Option 3.

In this topical area, I'll discuss an NRR audit that was performed at Nine Mile 2 in support of our uprate. We'll discuss long term stability, Option 3, and under that topical area, we'll discuss the oscillation power range neutron monitor that's installed at Nine Mile 2, the OPRM settings, the backup stability protection.

We'll discuss the 2003 Nine Mile Point stability event, and conclude with the effects of extended power uprate on long-term stability solution. Under the ATWS stability, we'll discuss the Unit 2-specific ATWS mitigation design features, preparation for the simulator demonstration that was done in support of the NRR audit.

We'll discuss the MSIV closure with failure to scram and turbine trip with failure to scram events, and then we'll address the conclusions associated with ATWS stability.

NRR audit was performed at Nine Mile 2 in October of 2009. The purpose of that audit was to demonstrate procedure actions and operator response to ATWS transience, that EPU conditions conform to regulatory requirements.

The audit reviewed implementation of long-term stability, Option 3, and it also observed operator performance in the plant reference simulator for the two events I discussed, the MISV (ph) closure and turbine trips with failure to scram.

The audit included a review of related procedures and mitigation strategies, and that audit also concluded in 2009 the simulator was not ready yet to show comparison data for EPU versus current license thermal power data. That has since been completed and provided to the NRC.

In terms of a time line for the oscillation power range monitor, in 1998, Nine Mile 2 received Amendment No. 80, which allowed the installation of the system, and it ran in the unarmed condition while we evaluated the performance of the simulator and performed tuning, to make sure that it was set up for the Nine Mile 2-specific application.

In 2000, we received Amendment 92, which armed the system, to make the OPRM trips active. In 2002, we implemented a plant-specific DIVOM curve, as a result of GE Safety Communication 01-01. In 2003, we implemented further changes to filter frequency and period tolerance setting for GE Safety Communication 03-20, and that was as it related to the Nine Mile Point 2003 event.

For the OPRM settings, we have cycle-specific DIVOM analysis performed using a TRACG methodology. The cycle-specific amplitude set points are defined in the core operating limits report, and for extended power uprate, we have reduced the enabled region from 30 percent of rated thermal power to 26 percent of rated thermal power, and that's to maintain the level of protection the same for extended power uprate, as it is for current license thermal power.

For backup stability protection, the backup stability protection regions are determined using cycle-specific ODYSY decay ratio calculations, and the regions are defined on the plant's power to flow operating maps. The backup stability protection actions are defined in plant procedures, with routine reinforcement in the operator training program, and the BSP exit regions --

MEMBER ABDEL-KHALIK: So if you'll back to the pervious slide, please.

MR. AMWAY: Sure.

MEMBER ABDEL-KHALIK: The set point for recirculation dry flow less than 60 percent. This two percent uncertainty between the dry flow, in the calibration between dry flow and actual core flow. Which direction does that normally go? Does it push you inside the exclusion zone, or outside the exclusion zone?

MR. AMWAY: The way we actually have our procedures set up is we implement the enabled region at 62 percent. So it's active up to 62 percent core flow, which gives us margin to the 60 percent requirement.

MEMBER ABDEL-KHALIK: So even if there was --

MR. AMWAY: So even if it was in a non-conservative direction, we're still bounded the way we set the system parameters.

MEMBER ABDEL-KHALIK: By the way you set it up?

MR. AMWAY: That's correct.

MEMBER ABDEL-KHALIK: Okay, thank you.

MR. AMWAY: On Slide 32, I discuss the 2003 stability event. That event was initiated by a component failure that resulted in a high to low speed transfer of both reactor recirculation pumps. In that event, the period-based detection algorithm initiated an automatic scram, because of core-wide oscillations.

The reactor in the post-trip event review, we determined that the reactor was properly tripped by the period-based detection algorithm. However, we did see some unexpected confirmation count resets prior to the scram.

The post-review analysis determined that two parameter settings needed to be changed, to address the confirmation count resets, and those parameters changes had been implemented for a BWR owner's group recommendation.

MEMBER ABDEL-KHALIK: So the net effect of these resets was a delay in reactor trip?

MR. AMWAY: That's correct.

MEMBER ABDEL-KHALIK: And how much of a delay was that, time-wise?

MR. AMWAY: I would have to take that as an action to take a look. It was, I mean I was on the event review team that looked at that data.

I can tell you that the backup stability protection actions that the operators would normally take and look for in that event were to the point that the operators even saw any oscillatory behavior, when the period-based detection algorithms scrammed the reactor.

CHAIRMAN ARMIJO: But we're talking seconds, minutes, hours?

MR. AMWAY: Seconds.

CHAIRMAN ARMIJO: Seconds, okay.

MEMBER ABDEL-KHALIK: So you'll follow up on this, and let us know?

MR. AMWAY: I will follow up on the actual time delay between when we think the reactor should have --

MEMBER ABDEL-KHALIK: Should have tripped, and the time when it actually tripped.

MR. AMWAY: Correct, and when it actually did, but that is a period of seconds.

MEMBER ABDEL-KHALIK: Okay, thank you.

MR. AMWAY: You're welcome. Effects of extended power uprate on the long-term stability solution. There are no methods changes for extended power uprate. The maximum rod line remains the same, and that is the maximum extended load line limit analysis boundary.

The OPRM armed region maintains the same level of stability protection. Cycle-specific set point analysis captures core design variations. Option 3 long-term stability solution remains unchanged, and the Option 3 OPRM set points will be developed based on plant-specific DIVOM curves for the extended power uprate cycle-specific reload analysis.

That concludes the overview of the Option 3. We'll move on to the next topic area, the ATWS mitigation for Nine Mile 2. We'll start off with a review of the mitigation system design features at Nine Mile 2. We have a redundant reactivity control system that is there to protect against ATWS events and provide mitigation actions.

That system is initiated on RPV high

pressure of 1,065 psig. At time zero, once that system actuates, we get a backup scram method, which is the alternate rod insertion. At the same time, time zero, we get an automatic reactor recirc pump trip, to slow speed. Normally, the plant would be at high speed operation of the recirc pumps.

At 60 hertz operation, that would transfer to low speed at 15 hertz. At 25 seconds into the transient, which the IRCSS is initiated if reactor power remains above four percent, which means the ARI was ineffective at completing the scram, then we get an automatic feedwater runback.

That's going to drop reactor water level down to where we want it for ATWS mitigation. It's very effective at mitigating any instabilities that may occur during the ATWS transient. We also at 25 seconds receive an automatic reactor recirc trip to off, which would be zero speed.

If reactor power remains above four percent, at 98 seconds, we get automatic boron injection, and that's with both trains of standby liquid control.

MEMBER ABDEL-KHALIK: And where does the 98 come from?

MR. AMWAY: The 98 seconds is a timer

that's built into the redundant reactivity control system. In our accident analysis, we assume 120 seconds. So the 98 seconds bounds the 120 second analysis.

When we prepared for the simulator demonstration, that demonstration was performed in 2009. So it was before any operator-specific EPU training on EPU conditions. The crews were provided with a 10 or 15 minute brief, just to say this is what EPU is in terms of power levels, steam flow, feedwater flows, and I used an SRO for the demonstration that is not part of the extended power uprate team, to avoid biasing the operator response.

The purpose of the setup was to confirm the expectation that the current procedures that exist today, and the actions and action times, are sufficient to address the ATWS event at EPU conditions, post-EPU compared to current license thermal power.

The initial conditions that we set up prior to the demonstration. We establish a reactor power, a full EPU power level of 39.88 megawatts thermal and 99 percent core flow, which is consistent with the upper end of the MELLLA boundary. We establish a pressure pool temperature at 90 degree

Fahrenheit, and a suppression pool level at 199.5 feet. Service water temperature, 84 degrees Fahrenheit, and no control rod motion occurred during the scram.

These initial conditions are consistent with worse case conditions that could occur prior to the ATWS initiation, and it's also consistent with the design analysis inputs.

As a result of that demonstration, we confirmed that the operators are able to place both loops of suppression pool cooling in service in 404 seconds, which is well within the assumed action time of 1,080 seconds. We were able to achieve hot shutdown in 406 seconds, and we maintained peak suppression pool temperature below the heat capacity temperature limit, with a five degree margin.

It's also important to note that five degree margin is based on a pressure band of 800 to 1,000 pounds, which is the normal pressure that we would maintain post-ATWS, until we confirmed that we were in hot shutdown. There is alternate strategies available, that if the approach the heat capacity temperature limit, we would take manual action to reduce reactor pressure, to gain margin for the heat capacity temperature limit, and avoid the blowdown.

MEMBER ABDEL-KHALIK: Do you have a schematic of how your power flow map will change after this power uprate? I'm particularly interested in the upper right corner of the power flow map.

MR. AMWAY: I do have that. If it's okay, if you let me get through here, before I conclude this presentation I can bring up my backup slides and show you the power to flow map.

MEMBER ABDEL-KHALIK: Thank you.

MR. GUZMAN: This is Rich Guzman. After the break, we can actually get this laptop working, and we do have backup slides available. Particularly if it's something we need to go in closed session, we can also cover it during the closed session.

CHAIRMAN ARMIJO: Yes. Probably we'll finish this part of the presentation, then take a break, and so --

MR. AMWAY: I can review the two loop power flow map right at the end of this presentation, I've got a few more pages, and Joel, that's going to be my backup Slide No. 10, if you want to get that ready.

So continuing on with Slide No. 37, for the MISV closure with failure to scram, as I stated, the containment parameters remain well within design analysis, and when we evaluated the simulator response to compare critical parameter response, they closely matched the design analysis for that event.

For turbine trip with failure to scram, we set up the same initial conditions, which again conform for the worse case conditions expected, and are consistent with the Design analysis inputs for that event. The results of the turbine trip with failure to scram, we again demonstrated the operators can place both loops of suppression pool cooling in service at rated flow, in 425 seconds.

Again, that's well within the assumed action time of 1,080 seconds. We achieved hot shutdown at 465 seconds. We maintained a suppression pool temperature margin to heat capacity temperature limit of 19 degrees Fahrenheit. Containment parameters remained well within design limits, and again the plant reference simulator behavior, in terms of critical parameter response, closely matched the analysis.

MEMBER ABDEL-KHALIK: Well, what are you trying to prove by the fourth bullet?

MR. AMWAY: The fourth bullet being the containment parameters --

MEMBER ABDEL-KHALIK: Right, indicated by the simulator.

MR. AMWAY: Oh. It would be the fourth or the fifth bullet then?

MEMBER ABDEL-KHALIK: Either this slide or Slide 37. It's the same kind of information.

MR. AMWAY: Okay. The containment parameters, I'm speaking there in terms of the suppression pool, peak temperature, the containment pressures in both the dry well and the supp chamber, those are the parameters I'm discussing.

The design analysis assumes approximately six to seven psig for these events. That's largely driven by the expected suppression pool temperature response.

MEMBER ABDEL-KHALIK: But I was just trying to get to the point of what are you trying to -- let's go back to Slide 37, please. So if we look at the fourth bullet here, okay, what are you trying to prove with this statement?

MR. AMWAY: What I'm trying to prove is that the operator -- that we can meet the operator response times, and maintain the containment parameter within design assumptions, or design analysis for these events.

MEMBER ABDEL-KHALIK: Is this a reflection on the simulator model, or a reflection on the design analysis, or a reflection on the operator ability to respond to the event?

MR. AMWAY: It's based on the operator's ability to respond to the event.

MEMBER ABDEL-KHALIK: So this is not a statement regarding the fidelity of the simulator?

MR. AMWAY: No.

MR. INCH: The simulator's not an engineered model. We don't use it for design. We use it for operator training. It's been benchmarked to plant data, and transient data in accordance with the guidance. I think there's, you know, for simulator fidelity.

MEMBER ABDEL-KHALIK: And that's why I'm asking the questions, right. The simulator is not an engineering model. It's simply an empirical model that's fit to analysis and plant data. So what does this statement tell you, other than --

MR. AMWAY: What I was trying to demonstrate was the simulator was providing an accurate training tool to the operators in this event. So what I did was I looked at critical parameters. So I'll give you an example. For the boron initiation temperature, 110 degrees.

The analysis assumption assumes that we would reach that temperature limit of 110 degrees in 59 seconds. When I reviewed the simulator data, it achieved 110 degrees in 60 seconds.

MEMBER ABDEL-KHALIK: Sure, because --

MR. AMWAY: In other words, it's just a qualitative analysis to say the simulator's performing similarly to how we expect the plant to behave, based on our design analysis.

MEMBER ABDEL-KHALIK: But isn't that a circular argument? If the simulator is based on the analysis, wouldn't you expect it to perform according to what the analysis said it should do?

MR. AMWAY: I would, and I'm not trying to qualify the simulator by that, but just to make sure that we have the simulator modeled to match what the design analysis says.

CHAIRMAN ARMIJO: But basically, the operators didn't have to do anything different for this event at EPU than they would have done at current licensed thermal power?

MR. AMWAY: That is correct.

CHAIRMAN ARMIJO: And you demonstrated that.

MR. AMWAY: That's right.

CHAIRMAN ARMIJO: That's what I get out of this.

MR. AMWAY: And that's really what we were trying to demonstrate with this, that we can use the same EOPs, same EOP actions, same action times and mitigate the event.

CHAIRMAN ARMIJO: But if, just for this slide, the only thing that you might expect to have changed was the margin on the suppression pool temperature.

MEMBER SHACK: Well, if he actually, if the operators took 1,500 seconds rather than 404 seconds, then the other bullets wouldn't have followed.

CHAIRMAN ARMIJO: Sure.

MR. AMWAY: That's correct.

MEMBER ABDEL-KHALIK: Oh, I see.

MEMBER SHACK: I mean that's -- so it really is a test of the operator action, assuming that in fact the design analysis --

MEMBER ABDEL-KHALIK: Is valid.

MEMBER SHACK: Is valid. But you know, what else can you expect?

MEMBER ABDEL-KHALIK: In a sense, you know, if the response on the procedure matches the assumptions in the analysis, and since the --

MEMBER SHACK: Because you expect it to.

(Simultaneous speaking.)

MEMBER ABDEL-KHALIK: The simulator is sort of fit into what the analysis says, you would expect it --

MEMBER SHACK: But you want to make sure that in fact that operators can do what the analysis assumes they do, and they seem to have -- they do it with some margin.

MEMBER ABDEL-KHALIK: I think I understand.

MR. AMWAY: That brings me to Slide No. 40, the conclusions. The conclusion of the demonstration showed that the existing procedures, operator reaction times and strategies are effective in mitigating ATWS and ATWS instability events.

Nine Mile 2 features an ATWS recirc trip function, and as a result, the transient power levels are primarily based on the maximum control rod line, which is unchanged for extended power uprate, and that operators can perform actions in a timely manner, to bring the plant to safe shutdown.

CHAIRMAN ARMIJO: If you could bring up that backup slide and take a look at it before we take a break.

MR. INCH: We want to take a break first, to look at it.

CHAIRMAN ARMIJO: Well, you know, I'd like to wrap it up with this, because this is --

(Simultaneous speaking.)

MR. AMWAY: I have a hard copy here.

CHAIRMAN ARMIJO: Can't get it up on the screen conveniently?

MR. AMWAY: Well the staff presumably is going to address the same issue after the break, so we can do it either way.

CHAIRMAN ARMIJO: Let's get it done now. We're a little bit ahead of schedule. Just take a minute, to kind of freshen your mind.

(Off record discussion between panel members.)

CHAIRMAN ARMIJO: Yeah, why don't we do that? We'll reconvene at 10:00, give us a 15 minute break.

(Whereupon, a short recess was taken.)

CHAIRMAN ARMIJO: Okay, let's come back into session. I think now we'll address the question before the break on the power flow map.

MEMBER ABDEL-KHALIK: I think you

indicated that you will stay on MELLLA.

MR. AMWAY: That's correct.

MEMBER ABDEL-KHALIK: Could you show us where the boundary is before the power uprate, and where it's going to be after the power uprate?

MR. AMWAY: I can. This is our two loop power flow operating map, and you can see on here these are the backup stability protection regions that are defined on our map that I spoke of, in that section on stability. This line right here is the MELLLA boundary, okay.

So right now, the operating point, this is shown for extended power uprate, but our current licensed thermal power is at roughly 85 percent, which would be about right across in here, okay.

So the expanded domain is really above this line, up to in this triangular area right here is the EPU power level. Where we're permitted to operate is anywhere within the white regions or the green regions, okay.

So the difference between current license thermal power and extended power uprate is this domain that used to be our MELLLA domain. It's considerably shrunk by, there's actually two lines shown here. The Gulf 1 line that's indicated by this marker right here, is the 100 percent rod line.

So the 100 percent rod line is defined as the rod line at which if you get to 100 percent core flow, you should be at 100 percent of rated power. There is actually a thin boundary domain within those two lines. It's very small. It's on the order of one percent.

So we have used most of that MELLLA operating room to achieve the uprate power level, such that when we're at 100 percent of EPU power level, we're in this corner.

MEMBER ABDEL-KHALIK: So you don't have much --

MR. AMWAY: We don't have much operating room, which underscores the reasons why we're trying to restore the original design margin in the reactor recirc system, which will enable us to go into the green region here, which will give us the operational flexibility we need at 100 percent of rated power for, you know, to account for small pattern adjustments of the control rod system, and for, you know, depletion of the fuel cycle.

MEMBER ABDEL-KHALIK: But you can go the other way?

MR. AMWAY: I cannot go into this yellow

region.

MEMBER ABDEL-KHALIK: Right.

MR. AMWAY: I can only go, this is my boundary to the left, and I can go all the way to the green boundary on the right.

MEMBER ABDEL-KHALIK: Five percent.

MR. AMWAY: It's roughly about five percent.

MEMBER ABDEL-KHALIK: How much will you gain by this improvement in the jet pumps?

MR. AMWAY: Right now, and I'll ask George to back me up a little bit on it, but we were going to -- we would not be able to achieve EPU power level with the existing condition of the jet pumps.

We would maximize our core flow, and we would not be at an operating point consistent with 100 percent EPU power level. George, did you want to add anything else?

MR. INCH: No, that's correct.

MEMBER ABDEL-KHALIK: So currently on this map, where is your maximum core flow, given the fouled condition of the jet pumps?

MR. INCH: Approximately 97 percent core flow.

MR. AMWAY: So that would be --

MR. INCH: At our current power level.

MR. AMWAY: --about right in this region. The flow is across the bottom, so you'd be measuring flow vertically, and they're in five percent increments. So this is 100 percent core flow here. This would be 95 percent core flow at this point.

MEMBER ABDEL-KHALIK: So is there a condition being proposed, that the power uprate be limited pending demonstrated performance of the refurbished jet pumps?

MR. GUZMAN: This is Rich Guzman. We do not have a proposed license condition on that at this time. But certainly we'll take that in development of our final safety evaluation. But I will certainly talk to the staff regarding that matter, and update our safety evaluation as needed, to address that matter.

MR. INCH: But we definitely don't believe there is any need for any license condition. I mean that's a limitation of --

MEMBER ABDEL-KHALIK: But you're assuming that when you do that, you'll get the right flow, to allow you to go to, you know, 120 percent of the original license thermal power.

MR. INCH: If, for example, if the jet

pumps don't perform as designed, then we'll have a shortfall in core flow, and we may not be able to complete the test program up to 120. But more than likely what will happen is that we won't be able to use the full increase core flow domain. But we'll be able to get to 120 percent.

Along the MELLLA line, 120 percent power is at 99 percent core flow. So and if we go to maximize dry flow on the system, we'll be able to get there. So we'll be able to achieve 120 percent, especially with the clean jet pumps.

MEMBER ABDEL-KHALIK: Okay. The point is currently, the way the plant is, you can't get the core flow required to allow you to remain within the power flow map at 120 percent power.

MR. INCH: I'd state it a little bit differently.

MEMBER ABDEL-KHALIK: Okay.

MR. INCH: We'll be within the power flow map.

MEMBER ABDEL-KHALIK: Okay.

MR. INCH: We may not be able to --

MEMBER ABDEL-KHALIK: To reach 120 percent.

(Simultaneous speaking.)

MR. INCH: --full power. But at all times, we're within the licensing envelope.

MEMBER ABDEL-KHALIK: Sorry, okay. That's fine.

MR. INCH: So that's why I was saying --

MEMBER ABDEL-KHALIK: Well, wouldn't it be reasonable then? I mean in other words, achievement of 120 percent power and remaining within the power flow map is contingent upon your ability to improve the jet pump performance?

MR. INCH: Well, it's really an operational flexibility issue, and Phil --

MR. AMWAY: You can operate with a small core flow window.

MEMBER ABDEL-KHALIK: But you can't even get there now.

MR. INCH: Well, you can get there without a flow window. Even with fouled jet pumps. You know with clean jet pumps, we're going to be able to get to 99 percent core flow. We'll be able to get to the full 105 percent core flow window.

There's no reason to anticipate any reason why we wouldn't, because the design analysis and all the original start-up testing supports that with a clean jet pump, we'll get there, you know.

Even with the higher DPs associated with EPU, all the numbers say we'll be able to get to 104 percent rated core flow. So we'll be within the power flow map the whole time.

MEMBER ABDEL-KHALIK: I guess this is really not the place to argue it, but you know, my feeling is that without demonstrated performance of the new jet pumps, it's --

MEMBER SHACK: That's sort of his problem. He has to stay within the power flow map.

MR. INCH: Right.

MR. AMWAY: I'm going to stay within the power flow map.

MEMBER ABDEL-KHALIK: Regardless.

MR. AMWAY: I've got nothing that tells me I can deviate. The way I see it, I mean we're taking out the existing inlet mixers, replacing them with the same type of inlet mixer that I have today. The reason why I'm doing that is to restore the original design margin of what the system was designed to do from day one when the plant was built.

It's not really a change, in terms of a different type of jet pump that would have different flow characteristics. It has all the same flow characteristics of the jet pumps today.

CHAIRMAN ARMIJO: And you'll demonstrate that in your power ascension test program.

MR. AMWAY: Right.

CHAIRMAN ARMIJO: So it's a flexibility issue, really.

MR. AMWAY: Yes.

MEMBER ABDEL-KHALIK: We'll just argue this point. We'll think about it. Thank you.

MR. AMWAY: Are there any other questions on the power flow map?

CHAIRMAN ARMIJO: Just keep it around.

(Laughter.)

CHAIRMAN ARMIJO: All right. I think we're going to go now to Peter.

MR. INCH: Oh, I do have an answer to the materials question, the 316 stainless.

CHAIRMAN ARMIJO: Oh yes.

MR. INCH: It's not the nuclear grade, but it is low carbon.

CHAIRMAN ARMIJO: Right.

MR. INCH: The carbon level is .02.

CHAIRMAN ARMIJO: Probably .03, isn't it?

MR. INCH: .023, I believe. It's a low carbon.

CHAIRMAN ARMIJO: It's a low carbon,

right. Okay. It is --

MEMBER SHACK: The spec on low carbon is .03, so he's well within that.

CHAIRMAN ARMIJO: 03, yeah. Okay, thank you.

MEMBER REMPE: Maybe it's a closed session question, but are there other plants that have used these new jet pumps with this new coating and had great experience? Or would you rather talk about it later?

MR. INCH: I think we need to discuss it later on that.

MEMBER REMPE: Okay.

MR. AMWAY: Thank you.

MR. GUZMAN: Good morning. At this time, the NRC staff will be presenting the Nine Mile Point 2 EPU ATWS instability review, specifically covering the audited areas that they covered, which the licensee did mention earlier. This presentation will be followed with an open session version of the fuel methods discussion by Dr. Yarsky, and then at that point, we'll go to closed session.

All right. So with that, I'm going to turn it over to Dr. Huang, to introduce his team, and go with the first slide.

DR. HUANG: Yes. I'm Tai Huang, from Reactor System Branch, along with Dr. Jose March-Leuba, who will present the subcommittee member on the staff evaluation on the Nine Mile Point 2 EPU.

There are two portion of the review. One is, you know, the submittal of available documents on their Option 3, long-term stability solution implementation, and second one would be the staff audit on their simulator, to verify whether their operating reactor, operating procedural to the training of their operator are adequate.

So that current long-term stability implementation, according to the staff evaluation, it's adequate for EPU. They satisfy the 10 CFR Part 50 design criteria 10 reactor design, and 12, suppression of the reactor power oscillation.

So level of protection in EPU is similar to the current licensed thermal power, and as well as the staff audit goes, we conclude that the Nine Mile Point 2 operator show good understanding of stability in ATWS issue for EPU, in staff observations of operators' action in the simulator support customary, assume a 120 second delay, assume for calculation.

You know, in their run, I mean the license run, you see they're using less than 120 seconds there. Nine Mile Point 2 EOP adequate for EPU, as the staff evaluation in SER shows. As we go on for that generic, you know, on that power flow map here, what the different from the curling (ph) thermal power to the EPU, you can see, you know, that in the power flow diagram here, you know, curling thermal power is right, corner is right here.

MEMBER SHACK: You need to do it on the mouse, I think.

DR. HUANG: Yeah, okay, on the mouse.

CHAIRMAN ARMIJO: Use the mouse.

DR. HUANG: Yeah, okay, and then EPU be extended out on that same narrow line to the EPU corner. You see the power flow map just is shorter. You see that EPU corner over there, all right. Then that there's no like end point are the same, is the same, after, you know, that reactor trip.

The end point would be the end point, following the pump trip, right here on the corner. So EPU and curling thermal power condition and not that would be the same point there. So that try to make EPU does not change the end point after the recirculation pump trip. So that diagram show this.

Next slide. Now there are two parts. One is Option 3, long-term stability implementation on the stability issue. You know the story on that, in the licensee's presentation. This is just summarized. They install since 1998 an arm since 2000, and plant has a very good experience on this Option 3.

According to that information, 2003 Nine Mile Point 2 event was detected and the scram activated. So that mean that OPRM on the Option 3 system is working. But the lesson learned from that, that the owners group, they come out with adjustment of parameter setting, so that that's already done for this plant.

So there's no impact expected for EPU. Option 3 and DIVOM methodology are applicable to this plant. Now ATWS, the second part on the ATWS instability, that the Nine Mile Point 2 has implemented latest EPZ and SAGs. So early level reduction in boron injection are accomplished through automated ATWS action. If high pressure is detected with power grid at four percent, then there's automatic flow runback, automatic boron injection.

At Nine Mile Point, we had excellent ATWS response, because they have a select system injections through high pressure core spray system on the top. So they don't need to worry about at the bottom up there. So they don't have that problem.

So 100 percent water driving feedwater, yeah, for this Nine Mile Point 2. So EOPs are reviewed every cycle, but are not affected significantly by EPU, because boron is injected in a high pressure core spray system, and there is no need to define a hot shutdown boron weight, you know, because from the top down.

So EPU does not affect heat capacity thermal limits slightly. It's by one degree, according to the analysis. So that's the only difference right there, right.

Now staff, second part. The staff has audited, and the purpose of that when staff review the performance, OPRM, there are two parts. OPRM Solution 3 system in the simulator, and staff reviewed the ATWS performance in the simulator as three events.

One is turbine trip ATWS from the MELLLA corner. MELLLA corner was simulated on stable observation in the slides. MELLLA corner will be the, you know, it's back in the slides, the MELLLA corner.

CHAIRMAN ARMIJO: The upper right-hand corner.

DR. HUANG: I understand that, okay. Mainstream oscillation valve, oscillation case ATWS from the MELLLA corner, and also from EPU conditions.

So we compare to that, and we can show on later slide what the difference between EPU and curling thermal power condition.

Nine Mile Point 2, you know, submit additional information, because at the time the staff audit at the plant, simulator not up to the EPU conditions for the ATWS. So they ran the results and show additional information to the staff. So we show that, you know, in a later presentation on the simulator. Now turn over this to Jose, Dr. March-Leuba on the simulator portion.

DR. MARCH-LEUBA: I'm Jose March-Leuba from Oak Ridge National Laboratory, an NRR consultant, and the recent discussion this morning about what is the purpose of doing simulator calculations of ATWS. Let me reemphasize your conclusion, that it is to review the operator actions.

You can ask Dr. Yarsky, who has presentation 20 minutes from now, how long it takes to run an ATWS calculation with engineering code, and he'll tell you several days, if not weeks, of CPU time.

This is with multiple restarts and multiple stops and backtracks, when the computer didn't do what it was supposed to, and five engineers looking at the results, to see what feedwater strategy you should have been using.

So we have two types of scenarios. I mean one is a very accurate code trace, for example, of Type G, that very accurately models the conservation of energy, mass and momentum and does everything well. On the other side, we have a pretty good simulator model, but it has a human in the loop, as the real operator doing the real control system on real time.

The first 120 seconds go in that time real fast. So it is not abundantly clear to anybody looking at it that with the 120 seconds, operator will be able to do anything. So the purpose of this audit that we performed was to go and in the real simulator with real operator, to see what they're supposed to do. My goal, just to give you a visual, is do we need Superman in the control room to do everything that we're asking these operators to do?

The conclusions after watching this is indeed, we don't need Superman in there. The operators are really well-trained, they're very professional, and if I were to show you a video of the operators handling an ATWS, and operators handling a control room in motion, you will not see the difference if you didn't have audio with it.

The operators just walked to the panel. They're not running from panel to panel. There's one operator in charge of level control; there's another operator in charge of the control rows, and they're doing their job. Indeed, the timing turned out to be -- the 120 seconds turned out to be very realistic. So that was the purpose of --

MEMBER ABDEL-KHALIK: During these observations, the operators knew in advance what event they're going to be responding to?

DR. MARCH-LEUBA: Not always. They knew they were going to be doing an ATWS, and I was going to point out that you do all these runs in sequence. So by the time you do the third simulation, they already know the procedures by heart, okay. So there is a variability from time to time, but not always.

We do go there and we kind of moved operators into oh, why don't you run this case now for us, and they didn't know it in advance. We really, we didn't do it on purpose, but we have extra time. We said well, let's run now assumption that the control room's not going at all, or let's run an assumption that there's a leak doesn't come in at all.

So they, we do change. We put some variability in the system. But even if they were to know it in advance, they have the procedures in big panels and there's a flow chart, and the first thing there the senior operator does is go get the right flow chart, put it on top of the table, and he's just following it.

He gets his marker, that's done. We enter in this branch, that's done. So they're well-trained. They're well-trained in advance. It's maybe knowing what training scene they're going to get reduces their anxiety a little bit, but I don't think it changes the results.

So the real difference was probably adrenalin. I realize the adrenalin will be flowing a little differently, and they might be doing things a little faster. But that's where training comes on. You do faster the right procedure, and they do follow procedure.

No operator goes and touches any panel unless the senior operator from behind says "it's time to do EOP 3G," and gives the order. So here we have very small description, because I don't want to show the details. Two MSRV closures, a cooling seal, cooling licensing thermal power at EPU. The very first thing that happens, you have this kickout here, which this is the MSRV closure.

You have a big pressure transient, and they have this on both. Right after that, you trip the pumps, and the moment you trip the pumps, both CLTP and EPU become the same condition. Then you see approximately the same thing happens.

There are some difference out here, and this is not due to the initial condition. It's what the operator did differently in these two runs. This run was done before, and this has to do with maintaining level once you reach the fuel, and he did it better on the second run. But there is no significant difference between the two.

We'll go to the next slide here. Again, this is not the engineering simulator, but it's pretty good. It does concern mass and energy and momentum, and we see it in Nine Mile Point, the peak capacity temperature limit, which is 140 degrees F, or 139, is not even reached for an MSRV closure. The maximum temperature in the suppression pool is 130 degrees F.

This is in part because Nine Mile Point 2 is a really great ATWS plant. I mean if God forbid we don't have an ATWS, but we're allowed to have an ATWS.

(Laughter.)

DR. MARCH-LEUBA: Because everything, everything is right in that plant. I mean everything is done automatically. The boron is injected in the top of the core, so it is no issue with remixing, and there's plenty of margin to everything.

CHAIRMAN ARMIJO: A slight reduction in margin is just the result of having more heat to get rid of, or is that an artifact of the --

DR. MARCH-LEUBA: It's an artifact. In principle, there should be no difference in the HCTL, heat capacity temperature limit, between the two.

CHAIRMAN ARMIJO: The actual.

DR. MARCH-LEUBA: Oh, you're talking about the --

CHAIRMAN ARMIJO: The actual could be higher, wouldn't it? You're getting rid of more.

DR. MARCH-LEUBA: You're not getting rid of more heat. That's the point. After you trip the pumps, you are at the same power than you were before, or an approximation. Now you do have a different core, you have a different coefficient. So you end up having the slightly different numbers, one, two, three percent difference.

CHAIRMAN ARMIJO: Yes, okay.

DR. MARCH-LEUBA: The difference between EPU and OLTP is in decay heat. There, you have 20 percent more decay heat. But as long as you don't go into a one week extended outage, decay heat is removed. It's before your scram and you are still putting 50, 60 percent power into your containment. That's what you have in ATWS.

So in summary, the staff found the EPU operation acceptable from a stability point of view, because the long-term solution, which is Solution 3, provides exactly the same level of protection under EPU than under the coolant power. Therefore, the OPRM scram and the OPRM procedures satisfy the GDC, general design criteria 10 and 12, which is the criteria that we have to satisfy.

On the ATWS scenarios, really the ATWS stability is not affected significantly by EPU event, and it's because after you trip the pumps, you are in exactly the same condition. I mean that satisfy all our acceptance criteria, which are three criteria, if you remember.

They are the core coolability, meaning you don't destroy your fuel and put it in the bottom of the vessel; you maintain vessel integrity; and you maintain containment integrity, and the containment integrity has to do with the suppression pool temperature we were talking about before.

Just to emphasize that Nine Mile Point 2 has an excellent ATWS pro forma design, and I wish all the plants were like it. I mean it has automatic trips, so we really don't even have to worry about the operator doing the right thing. The control system will do it for them. They inject the boron on the top of the core, so there's no mixing problems, and the feedwater pumps are 100 percent motor-driven, meaning that there is no issue with how much availability of inventory to maintain level in the vessel during ATWS, and that's the end of our presentation.

CHAIRMAN ARMIJO: Any questions? I suspect one. Thank you.

MEMBER ABDEL-KHALIK: What is the last bullet?

DR. MARCH-LEUBA: If you do not have motor-driven feedwater pumps, which many plants don't; they use steam-driven, the moment you close the MSRV, then you don't have steam for those pumps and you don't have feedwater, and you rely on other ACCS systems, which are not as large.

If you were to increase, in some of these plants, if you were to increase the power significantly, you will not have enough. HPCI will not be sufficient to maintain level. Here, you have 100 percent feedwater available. You don't have any problems.

MEMBER ABDEL-KHALIK: Okay, thank you.

CHAIRMAN ARMIJO: Thank you. Let's keep going. I guess we're getting into Peter Yarsky's presentation.

DR. YARSKY: Hello. I'm Dr. Peter Yarsky from the staff. I'm a member of the Office of Research, and I'm going to be talking about the applicability of the interim methods to the Nine Mile Point 2 extended power uprate LER. The basis for our methods review was the safety evaluation for the interim methods license and topical report, NEDC-33173P.

In the course of our review, we have confirmed that the EPU LER is fully consistent with all of the conditions and limitations in the staff's SE for the IMLTR. The IMLTR specifies 24 different conditions and limitations. In the Nine Mile Point 2 EPU application, no supplements to the IMLTR are referenced.

The Appendix A to the power uprate safety analysis report provides the disposition of each of the conditions limitations, and in the course of our review, we found that all 24 conditions limitations were acceptably met. In the course of our review, we conducted one regulatory audit pertaining to the IMLTR.

Then the audit had to do with initially guiding LPRM calibration interval. The frequency with which the LPRMs are calibrated, that's sometimes referred to as the LPRM update, affects core monitor accuracy to predict core power distribution. In the Nine Mile Point 2 technical specifications, the LPRM calibration interval is specified in units of effective full power hours. So at EPU conditions, the equivalent exposure interval between LPRM calibration intervals would increase along with the thermal power by approximately 15 percent.

We asked RAI SMPB-1, which was the only RAI coming from the methods review, to address LPRM calibration interval, and the outcome of that RAI was that the staff conducted an audit at GEH, to confirm that the power distribution uncertainties were acceptable for this longer exposure interval.

MEMBER ABDEL-KHALIK: Was this issue raised at the original stretch uprate?

DR. YARSKY: I'm not familiar with the stretch power uprate review.

MEMBER ABDEL-KHALIK: But it would have been equally applicable, wouldn't it?

DR. YARSKY: The extension of the interval, yes, would have been applicable, but not equally applicable --

(Simultaneous speaking.)

MEMBER ABDEL-KHALIK: Well, I mean essentially the same issue.

DR. YARSKY: Yes. I personally became first familiar with this topic during the review of the Monticello EPU, and the conclusion and resolution of that issue was different in Monticello than for Nine Mile Point 2.

MEMBER ABDEL-KHALIK: Okay.

DR. YARSKY: Yes. The Subcommittee was briefed on this issue during a generic review related to interim methods, I believe, in August.

CHAIRMAN ARMIJO: This summer.

DR. YARSKY: It is June, in June. So it's the same topic, just applied on a plant-specific basis.

CHAIRMAN ARMIJO: Peter, if the uncertainties hadn't been acceptable, wouldn't the solution be pretty straightforward? You just recalibrate?

DR. YARSKY: The solution would have been straightforward. It could easily have been an adjustment to the LPRM calibration level. That has been done by other licensees seeking power uprate.

CHAIRMAN ARMIJO: Okay.

DR. YARSKY: That's all I have. So --

CHAIRMAN ARMIJO: That's it. Okay. Well then I think we're ready to go into closed session, and first I'd like the staff and the applicant to confirm that the right people are here, and that nobody's on the bridge line that shouldn't be on the bridge line.

(Whereupon, at 10:29 a.m., the meeting was adjourned to closed session.)

O P E N S E S S I O N

10:59 a.m.

CHAIRMAN ARMIJO: Ready to go.

MR. AMWAY: Okay. Before we begin the material mechanical civil discussion, I wanted to respond to the open question on the quantifying the scram delay back in the Stability section, where I presented the 2003 stability event for Nine Mile Point 2, and the question of that was what kind of time delay do we have from the onset of oscillation to where we should have scrammed to when we actually scrammed the reactor.

The answer to that question is 15 to 20 seconds.

CHAIRMAN ARMIJO: That's the delta between the ideal and the somewhat delayed because of the resets?

MR. AMWAY: The time delay from when the OPRM should have scrammed the reactor, based on confirmation counts, and when it actually did, that total delta is 15 to 20 seconds.

CHAIRMAN ARMIJO: Okay, and with the corrections and the updates, that has disappeared where you expected?

MR. AMWAY: That's correct. That would

eliminate that 15 to 20 second delay.

MEMBER SHACK: And you're not going to run an experiment to verify that?

MR. AMWAY: That is also correct.

CHAIRMAN ARMIJO: And we don't expect you to.

MR. INCH: My name is George Inch. I'm the physical engineer for mechanical structural for the power uprate of Unit 2. I'm going to be going through the reactor pressure vessel internal materials issues, and related flow-induced vibration evaluations.

So for the internals, the EPU evaluations included the effect effluence, the effect of flow-induced vibration, structural effects that are non flow-induced vibration-related, and the impact of EPU on the current material condition with regard to intergranular stress corrosion cracking, and irradiation-assisted stress corrosion cracking.

The accepted threshold for effluence, where irradiation-assisted stress corrosion becomes a significant factor in the growth rate of an existing IGSCC flaw and potential IASCC that's accepted in the BWR vessel internals program is 5E to the 20 neutrons per centimeter squared.

The existing components that are expected to exceed that threshold in the current license term, and that's expected, anticipated in the vessel internal program scope are the top guide, the shroud and the core plate.

CHAIRMAN ARMIJO: George, you know, that threshold is a pretty fuzzy threshold. It's not a hard line.

MR. INCH: Right.

CHAIRMAN ARMIJO: So what other components are close to that 5 times 10 to the 20th? You know, these were, this is the same list as pre-EPU.

MR. INCH: Yes.

CHAIRMAN ARMIJO: So as you go up 20 percent more in flux, I would expect more components come into this population, and others get closer.

MR. INCH: Additional components really don't come into the mix. I mean the effluence level goes up, but the threshold, that threshold really isn't exceeded by any additional components.

CHAIRMAN ARMIJO: So okay.

MR. INCH: So that the --

MEMBER ABDEL-KHALIK: What's the next -- I mean the question --

(Simultaneous speaking.)

MR. INCH: It would be some jet pump components that are in the core region. Everything that's going to --

(Simultaneous speaking.)

MR. INCH: And, you know, these are the core region components and then, you know, what's outside of the reactor, I mean the core shroud would be the jet pump components. But because of the size of the annular region, you get significant attenuation. So those components are, you know, don't approach --

CHAIRMAN ARMIJO: None of the guide tubes, whether it's drives or instrumentation.

MR. INCH: All the instrumentation in the core is above this, just as a matter of course. So and the guide tubes are all below the core plate, and so you get significant attenuation as you go down.

CHAIRMAN ARMIJO: Okay. So these are still the same components you worried about pre-EPU?

MR. INCH: Yes.

MEMBER ABDEL-KHALIK: What's the condition of your shroud now?

MR. INCH: The Nine Mile Point 2 core shroud has IGSCC cracking associated with the belt line welds. The H-4 weld and H-5 weld have OD IGSCC cracking. Cracks are approximately 70 percent of the circumference. They're relatively shallow, as it's a two inch shroud, and the cracks are less than half of an inch in depth.

They were first identified in the baseline inspections performed in the 90's. I believe it was in '97 where they were identified. We've ultrasonically inspected those multiple times, at least four times.

Since we implemented hydrogen water chemistry and noble metals, we haven't seen measurable growth that we consider to be real growth. With UT, there's always variation. So you never match it up within the uncertainty of the deployment tools and the UT devices. But the condition has been stable for at least ten years.

CHAIRMAN ARMIJO: Are these shrouds clamped? Have you put any of these --

MR. INCH: There's no tie rods.

CHAIRMAN ARMIJO: There's no tie rods, so it's just as-built, and you're monitoring and testing the cracks?

MR. INCH: Yes. The flaw evaluation for the core shroud has been updated for the power uprate higher loads for differential pressures. It's, that flaw evaluation still shows that the normal upset event is the controlling event for the core shroud. It's not the faulted event.

The vertical welds, all the vertical welds are clean. There's no cracking on the vertical welds. Very minor cracking on other locations, less than ten percent, very typical. So the location of the indications on the core shroud are consistent with the understanding in the fabrication process. So it was the final weld built --

CHAIRMAN ARMIJO: After you, on the shroud, since you've inspected it a lot, after you implemented the hydrogen water chemistry and the noble metals, have you found any new cracks that hadn't been there pre-hydrogen?

MR. INCH: Not that we consider -- we don't consider them new. The UTs have evolved over the past ten years. I'm always seeing, you know, I get a scan and the percent cracking is essentially the same. But we get a little additional coverage at a location. There's a lot of starts and stops, but there's been no significant change in with the new cracking.

CHAIRMAN ARMIJO: Okay, just a couple of other things. On the top guide and the core plate, have you, do you have any -- can you inspect them well, and do you have cracks, IGSCC, in those components?

MR. INCH: The top guide, we have the inspection can be performed of the top guide grid beams. That was inspections that were recently added to the VIP program, approximately --

MEMBER SHACK: These are enhanced VT1?

MR. INCH: These are enhanced VT1, where we clear the cell and they have a cleaning process, to get the enhanced visual capability. We've done, completed an initial deployment of this new tool in 2010, and that worked quite well. So we haven't established that Unit 2A baseline yet on the top guide, but we --

CHAIRMAN ARMIJO: Have you seen anything that looks like a crack?

MR. INCH: No, no. We've looked at two cells. We've done standard refueling inspections. We've looked at two cells with the enhanced VT1, and we've done the standard VT inspections that would normally detect any significant structural issues. But we haven't completed the five percent baseline that is recommended as planned for implementation over the next --

MEMBER SHACK: Now these are one of these milled top guides, right? Solid, or do you have the interlocking beam kind of thing?

MR. INCH: This is a BWR-5, so it's not the BWR-6 top guide.

MEMBER SHACK: The 6 is the one that's milled out?

MR. INCH: I believe so.

CHAIRMAN ARMIJO: So these are interlocking, welded?

MR. INCH: I'll verify that, but I'm pretty sure these are interlocking designs.

MEMBER SHACK: So there's lots of corners to look at.

CHAIRMAN ARMIJO: Okay so -- yeah, that's the problem. Now the shroud is UT inspectible, but what about the core plate?

MR. INCH: Core plate at Unit 2, the only inspection requirements for core plate are for the bolting, and it's part of the program. The evaluation that we have right now is a generic evaluation for the inspectability of the bolting, and so we have in place, as pretty much all the, most of the BWRs do, an interim analyses that shows that the bolting will retain its integrity through the 40-year term.

The VIP is working on alternatives to the inspection recommendations, so --

CHAIRMAN ARMIJO: But basically, they're not inspectible?

MR. INCH: Not currently, that's correct.

CHAIRMAN ARMIJO: Okay. So you're really relying on the analysis and the mitigation afforded by the water chemistry?

MR. INCH: That's correct.

CHAIRMAN ARMIJO: Okay.

MR. INCH: So the effect of effluence is not insignificant on the core shroud. It's a 40 to 60 percent increase at peak locations in the core barrel there, and you know, that's because we're loading higher batch fractions and the higher power bundles are getting closer. So there is a, it does increase effluence.

We have taken that fluence out through the 60-year term, looked at the peak fluence. We stay within the currently accepted range, where hydrogen water chemistry will remain effective.

MEMBER SHACK: What is that end of life fluence?

MR. INCH: Let me get back to you. It's less than 10 to the 22, I know that.

MEMBER SHACK: It's cold comfort, but --

MR. INCH: It's above the threshold by which radiation-assisted crack growth rate acceleration is expected to occur. We also have reduced ductibility of the materials.

CHAIRMAN ARMIJO: Sure.

MR. INCH: So we're within, we're applying the VIP guidance.

MEMBER SHACK: The normal guide for effectiveness of hydrogen water chemistry, somewhere around three times ten to the 21, and it's sounds like you're probably pushing that.

MR. INCH: Towards the end of the 60 years. Yes, we'll be pushing that number. But let me verify, get you a good number on that. But so that covers the slide, I think.

CHAIRMAN ARMIJO: Not really. Your last bullet, I just, I think it's -- I have to argue with that statement, because your actions are much better than the words on this chart.

The fluence does everything. It does nothing good for you except make power. Your radiolysis rate goes up in proportion to the power uprates. So that means the water chemistry gets more aggressive.

But you're compensating that by increasing your hydrogen input rate by the same ratio.

MR. INCH: Yes.

CHAIRMAN ARMIJO: So you are addressing that. Radiation hardening is going to push things in the wrong direction. So I looked at your various documents, and you're doing everything that I think can be done, that addresses the mechanism of this stress corrosion cracking, either IASCC or IGSCC.

But I just take exception to that statement, that it doesn't represent a significant increase in potential, because I think it really does, and your actions indicate that you kind of think so too. So I don't know where that statement came from. But maybe you want to get rid of it in the full Committee.

MR. INCH: We don't need to debate that, except -- the flow-induced vibration of the internal, Nine Mile's well in the pack of the GE operating experience for the flow rates that were taken in the Unit 2. The components that are really impacted are the shroud head separator assembly, because you've got the higher steam flows coming up through it.

The jet pumps to a lesser extent. As I said, it's a 1.9 percent effect there versus an 18 percent on a steam flow.

DR. WALLIS: So the effect is really a V-squared effect? Is that what it does?

MR. INCH: It's a turbulent -- that's how it's evaluated, yes. For those internals, it's a velocity squared due to turbulent loading.

DR. WALLIS: Resonance or anything?

MR. INCH: No. They do a separation evaluation to any vortex setting, and that's the standard procedure that GE's used. The peak stresses for that shroud head remain less than 5,000. GE uses a 10,000 psi criteria. So the internals really are robustly made. There's significant margin to any FIV issues.

The top head region where you have the higher steam flow, those velocities remain very low, where you have the spray nozzles and the head or the head vent lines. So those stay below two feet per second, and the cross-flow configuration has been taken into account, and there's large margins there.

So the conclusion is, you know, pretty clear that there's no impact or detrimental effects on any of the internals due to potential for the FIV.

Structural effects. You know, the higher power levels, you do have higher internal pressure differences, and there's some small temperature changes. All the analyses are done consistent with the original design bases. For Unit 2, there's really no structural change has been made to the internals that need to be considered. As I said, the thermohydraulic changes are fairly straightforward.

With the pressure differences and the temperature changes, there's a little bit change in the carry-under fractions. The way GE does these analyses is with scaling factors to the original design, and with that, you know, for example the core plate and core shroud goes from 11,000 to about 14,000, and that's with primary membrane bending is limiting, with an allowable of 21,450 psi.

So the shroud head bolt. The limiting component there is the T bolt and bearing stress, and that goes from 8,000 to 13,500, with an allowable of 18,000. Now that shroud head bolt analyses is taking credit for the reduced number of shroud head bolts than we actually currently have installed. So it's a conservative evaluation.

So all the usage factors really didn't have for the internals, they didn't have a significant change. The only one of note was really the shroud went from .43 to .507. That's primarily due to the slightly larger temperature variation on heat up and cool-downs. That's not a high cycle fatigue.

So all the internal components are fully qualified, and as I mentioned previously, the core shroud flaw evaluations that we work have been updated to reflect the higher pressure differences, the higher fluences, reflective of the power uprate condition.

We covered pretty well in the questions the, you know, what's been done for IGSCC and IASCC. It's procedurally controlled. The program that's been implemented has always considered, you know, aging effects and the higher fluence level. So the selection of the components and the intervals that are selected aren't impacted by the higher fluence levels. They're still fairly conservative intervals for all the components.

Like for the shroud, it's a maximum of ten years, even with hydrogen water chemistry and ultrasonic inspections. We talked about the hydrogen water chemistry and noble metals. There is an increase in the hydrogen, just to keep, maintain the three to one molar ratio in the downcomer.

CHAIRMAN ARMIJO: How do you monitor the molar ratio water chemistry program?

MR. INCH: At Unit 2, the implementation

of the hydrogen water chemistry currently uses molar ratio. We are evaluation the recently-issued staff approved BWRVIP-62-Alpha guidance, where the staff has allowed, on an interim basis, to use molar ratio. But they want to see electrochemical potential monitoring being performed, to credit the hydrogen water chemistry. We haven't implemented that at Unit 2 at this time.

CHAIRMAN ARMIJO: Yeah. Instrumentation is tough. It's not necessarily survivable.

MR. INCH: Yes. That's a very challenging request from the staff right now.

CHAIRMAN ARMIJO: Yes, yes. But molar ratio gives you good indication that it's working.

MR. INCH: Yes.

CHAIRMAN ARMIJO: You have, also you have online noble metal capability, so you don't have to do this during an outage.

MR. INCH: Yes. We implemented, we were one of the first plants to implement online. We did it in 2008, and it's done on a yearly basis, and it works. It's about two, two and a half weeks every year done, at least 90 days after the new fuel is installed, and it's working well for us. It's much easier to do and less of a plant transient than the offline process.

MEMBER SHACK: Thank you, and you don't have to -- I mean this goes on forever. You don't ever have to do an offline noble metal injection again?

MR. INCH: Yes. That's --

MEMBER SHACK: That's the goal.

MR. INCH: That's the qualification of the process, yes, that you don't have to ever do an offline application. Yes, the details of the process are, you know, I don't think we probably need to get into. But it's a different particle size, much finer. It's engineered to penetrate deeper into cracks, and so --

CHAIRMAN ARMIJO: So this is platinum that you're added or not palladium?

MR. INCH: Yes. I believe with online, they eliminated the, I think it was rhodium that they had in there. But it's only platinum. So it's a different cocktail that they're using. But it's fundamentally the same.

CHAIRMAN ARMIJO: Okay.

MR. INCH: I think we already talked about the control blade cracking. One of the impacts of power uprate is scaling of the reactor pressure vessel, you know, nozzles, and one of the -- so the scale factors were applied to the design bases fatigue usage, accounting for both 40 and 60 years in design bases.

When we get our license renewal for 60 years at Unit 2, it was identified that the feedwater nozzle and another location in feedwater had the potential to exceed one in the license renewal term. At that time, we took a hard look at, you know, what's the best way to approach this, and you know, design bases fatigue usage calculations are usually very conservative, and they take up a design cycle, and a number of design cycles.

So when we looked at it, it was clear to us that we could optimize, you know, get a more realistic usage factor by actually more accurately trending. So we committed for the locations that were predicted to be above one, to implement a fatigue monitoring program, such that long before we would approach one, we would be predicting it and could plan any appropriate actions.

Those remain the case for the power uprate conditions. The scaling of the -- well, before I go to that, the one location that we did select for fatigue monitoring using stress-based monitoring was the feedwater nozzle location. That's not unusual for the BWRs. Where that location is part of power uprate, we did a fatigue, a refined fatigue usage calc.

MEMBER SHACK: Oh yeah. I guess what caught my eye was computing usage with FatiguePro, which has generally not been received well, unless there's a new version of FatiguePro that eliminates the one stress factor kind of approach.

MR. INCH: The numbers I'm showing here, this is an important clarification; I'm on Slide 62, these are based on design, not FatiguePro. So the numbers we're quoting here for the EPU 40-year CUF are a refined design basis usage for the 40 year term, not keyed to FatiguePro.

But as you can see, the standard multiplication factor for license renewal is a 1.5 factor on, you know, for the additional 20 years. Even with the refined usage, we would still predict the stainless steel clad portion of the feedwater nozzle safe end to be above one.

So we're using FatiguePro right now. It was first, the software was first installed in 2008, as a simplified way and a more accurate way to count cycle. We are doing the stress-based monitoring of this location, and the current FatiguePro software does use the simplified greens (ph) function. So the RIS is applicable.

The EPU scaling is relatively small. It's a six percent to 15 percent change, and --

MEMBER SHACK: Now does that include an environmental factor?

MR. INCH: No.

MEMBER SHACK: What would happen if I put in an environmental factor?

MR. INCH: This usage factor is a design bases usage factor. The power, the license renewal provisions have evaluations for environmental effects in the license renewal term. So the FatiguePro monitoring does include environmental usage, and I believe there was an environmental usage evaluation done.

I would have to get back to you on this particular nozzle, on how environmental fatigue affects these numbers.

MEMBER SHACK: If you could go back to the next bullet, it says there's still a discussion going on, I guess.

MR. INCH: Well, yes. You know, the whole industry is working to address the RIS. There is a --

Structural Integrity is working on a FatiguePro update that would address the full NB-32 fatigue methodology, and to include environmental fatigue usage in the rules. You know, right now, we're not required in using this to manage below one in the 40 year term.

So we enter the license renewal term in 2026 at Unit 2, so there's quite a bit of time to get this right. We are committed to implementing, you know, fatigue monitoring, and as part of that, the RIS requires us to evaluate that, you know, before we're actually crediting it for usage below one. So that is what our current status us.

CHAIRMAN ARMIJO: George, could you go back to Slide 62?

MR. INCH: Yes.

CHAIRMAN ARMIJO: I don't understand how the cumulative usage factor at current license thermal power is higher than that at EPU. Am I reading this thing wrong, or --

MR. INCH: Well, that's the refined calculation.

CHAIRMAN ARMIJO: But they're two different calculations?

MR. INCH: Yes. The original calc had a, for multiple events, had --

CHAIRMAN ARMIJO: Well, you would say that was a crude calculation?

MR. INCH: It was conservative, and you know, there's a way --

MEMBER SHACK: Unconcerned. It was less than one. That was all you needed.

MR. INCH: It's the way they did the calcs. If you were less than one, you were done. Everybody knew they were conservative, so --

CHAIRMAN ARMIJO: Okay. So now this is a refined?

MR. INCH: Yes.

CHAIRMAN ARMIJO: The EPU is a refined? Okay.

MR. INCH: Now when I say "refined," what they did is they went back and looked at each particular event, and then for each event, there was a thermal FEA, where they looked at what the actual cycling on the stress --

CHAIRMAN ARMIJO: It was detailed.

MR. INCH: It was a detailed accumulation for each event.

MEMBER SHACK: Now it's got four significant figures rather than three?

CHAIRMAN ARMIJO: Yeah, right. It's

really good.

(Simultaneous speaking.)

CHAIRMAN ARMIJO: So it's an analysis of -- so you did cycle by cycle you did?

MR. INCH: Yes. There's, to get these numbers, they started from a natural baseline, and then they refined each event, and then refined each cycle and what the usage for each cycle is.

CHAIRMAN ARMIJO: And then they added them up?

MR. INCH: And then added them up.

CHAIRMAN ARMIJO: Okay, got it.

DR. WALLIS: What is this EPU scaling factor small mean? What's that?

MR. INCH: You know, relative to what does EPU do to the usage factor?

DR. WALLIS: But does it mean that without the refined calculation, it would actually increase the CUF above one?

MR. INCH: I'm not sure I understand here.

DR. WALLIS: Well, to me, it implies that the EPU increases things by 6 percent to 15 percent.

MEMBER SHACK: Right.

DR. WALLIS: So if you use the old CUF

calculation, you get something closer to one?

MR. INCH: That's right.

DR. WALLIS: Okay. So you need to refine the calculation?

MR. INCH: Yes sir.

DR. WALLIS: Thank you.

CHAIRMAN ARMIJO: This scaling, it's a stress, scaling on stress?

MR. INCH: Yes.

CHAIRMAN ARMIJO: Certainly not cycles.

MR. INCH: Right. It's a scaling on stress.

CHAIRMAN ARMIJO: Okay, okay.

MR. INCH: That's all I have.

CHAIRMAN ARMIJO: Well, we have a dilemma here. We could take a quick look here. The staff, let me ask the staff. Could they get their presentation done in half an hour?

MR. GUZMAN: Actually, the assigned presenter is not here.

CHAIRMAN ARMIJO: Well, that answered my question. I think we'll take an early lunch, and we'll be back at 11:30, unless somebody's got an objection to that. At 12:30, I'm sorry.

(Simultaneous speaking.)

CHAIRMAN ARMIJO: Okay. We're going to take a lunch. Be back at 12:30. Thank you, Bill.

(Whereupon, at 11:31 a.m., a luncheon recess was taken.)

A F T E R N O O N   S E S S I O N

12:57 p.m.

CHAIRMAN ARMIJO: Okay, gentlemen. We're going to reconvene. For those who don't know, that jackhammer is actually outside. It's not in the building, so structurally the building is sound, I think.

MEMBER SHACK: Of course, there's resonance.

CHAIRMAN ARMIJO: Right. So we'll go ahead. Rich, you put your group is up.

MR. GUZMAN: Good afternoon again. My name is Rich Guzman. Before we transition over to the staff giving their presentations on the materials and mechanicals and civil engineering reviews, the licensee requested to give some clarifications from the morning meeting. I thought this would be a good time to provide that, before we start delving into the materials and steam dryer discussions.

MR. WENGLOSKI: Good afternoon. Phil Wengloski, Constellation Energy Nuclear Group. I just wanted to clarify my response. It was brought to my attention on the control blade cracking issue, that my answer could have been taken one of two different ways.

The response I intended to give, which was consistent with George's Slide 60, is that we do not have the control blade models, marathon models as susceptible to the cracking. We may have other marathon models present in the core, but not the models that are susceptible to cracking.

CHAIRMAN ARMIJO: Is yours a C lattice plant?

MR. WENGLOSKI: That's correct.

CHAIRMAN ARMIJO: Yes, but I saw the comment about C lattice plants having susceptibility to this problem, and I think that's very -- I don't want to (papers shuffling). Anyway, it doesn't make any sense to me. But you do have marathon blades in a C lattice plant?

MR. WENGLOSKI: Correct.

CHAIRMAN ARMIJO: And GE has told you that's not susceptible?

MR. WENGLOSKI: Right. The models that we're using are not susceptible to cracking.

CHAIRMAN ARMIJO: Time well tell, but okay. But you do have marathon blades?

MR. WENGLOSKI: Correct.

CHAIRMAN ARMIJO: Okay.

MR. WENGLOSKI: Any further questions?

CHAIRMAN ARMIJO: No.

MR. WENGLOSKI: Thank you for that.

MR. INCH: Yes, George Inch. The peak effluence were approaching 22 EFPY (ph), probably out two years. So it's approximately 2E to 21 each location on the H4 well, projected out at EPU effluences until the end of 40 years would be 2E to the 21. Then it goes to 4E to the 21 at that H4 location.

CHAIRMAN ARMIJO: That's at 60 year or 54 --

MR. INCH: The 54 EFPY in 60 years, yes. So it looks like we're just below at the 60 year mark --

MEMBER SHACK: Well, people have various opinions.

CHAIRMAN ARMIJO: About how long it will last, how long it will work.

MR. INCH: The top guide clearly is above that at these locations.

MR. GUZMAN: Okay. With that, we'll go ahead and start our presentation. Pat Purtscher will be giving his brief on the materials engineering review. Pat, you want to get started?

MR. PURTSCHER: Okay. We looked for the

reactor vessel embrittlement, and there are several factors that we always consider. The EPU is going to increase the total fluence in the vessel, so we're going to check initially the material surveillance program.

That's the fundamental program we use to monitor. We build with the limiting materials in the plants. At Nine Mile, they are enrolled in the BWR VIP integrated surveillance program.

As part of that program, they're not a host plant. They're limiting materials are characterized by capsules that are in other plants, BWR plants. So the change in effluence for Nine Mile doesn't directly affect the capsules that will be used to characterize the limiting beltline materials.

They do still have two capsules in their reactor vessel that are being irradiated, but there's no current plans to use those at this point. They're backup capsules. Some of the other factors that we look at are all related to Appendix G requirements. The PT limits, the upper shelf energy projections for all the materials in the beltline, and then there's an inspection exemption that's been granted for the circ weld on the vessel.

So we examine that to see if that, how

that changes with the increased fluence. In all cases, they have passed, they still meet these requirements from Appendix G, with significant margins remaining. So there really is no concern from the staff, based on this increased fluence due to the EPU.

Next slide. We're now going to look at the internals and the core support materials. Again, due to the -- now that the fluences are higher on the internals than they're on the reactor vessel itself, and as they mentioned in their presentation, now the top guide, the shroud and the core plate are all exceeding what we take to be the threshold for radiation-assisted stress corrosion cracking.

So now we consider them to be susceptible materials once they get above that threshold. To address that, they have instituted, you know, using BWRVIP-62, that's been characterized as a Category 3 plant, they're using noble metal additions to mitigate the possibility of stress corrosion cracking. This is following all the EPRI guidelines.

So they are following all the industry standards, and the staff sees no issues related to the increased fluence.

MEMBER SHACK: Well, there was sort of a discussion this morning that as a Category 3B plant, where they're only looking at molar ratio, you guys take a somewhat skeptical view of that, and you're asking them to do ECP measurements.

MR. PURTSCHER: Well, but they did do one time measurement of EPC when they instituted noble metal online additions. So they did it once, and kind of validated the secondary parameters that they're monitoring.

So as long as they monitored, as long as they checked that once, and there have been no major changes to the noding and the environment, we feel that's enough justification. So with that one time measurement, to validate it.

So really that's, to the vessels and the internals, that's really the summary. Just to say it again, the EPU has a minimal effect on the embrittlement issues, the upper shelf values, the PT limits and the surveillance program. These three internal components we've talked about, that exceeded the threshold for IASCC, are being managed by BWRVIP documents that have been accepted the staff.

So this, since there should be no problem associated with the increased fluence related to the EPU. So we're satisfied with their submittal. Okay. Any questions?

MR. TSIRIGOTIS: My name is Alexander Tsirigotis. I work in the Mechanical and Civil Engineering Branch, which reviews the EPU impact on the structural integrity of the system structural components.

Mainly, the pressure retaining components and the supports, the reactor pressure vessels and supports, the control mechanisms, reactor situation pumps and supports, reactor pressure vessel internals and core support --, and the seismic and dynamic qualifications of the mechanical and electrical equipment.

The approach to evaluate the Nine Mile Point 2 EPU impact on the structural integrity of the -- follows the guidance which is provided in the staff-approved Z topical report entitled "Constant Pressure Power Uprate," and it's licensing report NEDC-33004P-A.

This is commonly referred to at the BWR EPU as the CLTR. The CLTR also refers into two other Z topical reports, the ELTR-1 and ELTR-2, which provide more detail on the generic guidelines and generic evaluations.

These two topical reports, together with associated NRC staff position paper on safety evaluation, incorporated in the topical reports, have been applied for all BWR extended power uprate submittals, since the NRC review and acceptance or endorsement for --.

The staff approved the CLTR is for constant pressure power uprates, commonly referred to as CPPUs. With a power increase up to 20 percent from the plant's 100 percent original thermal power, and with a minimum and maximum steam and feedwater flow increases up to about 24 percent.

The CPPU approach assumes that the maximum reactor pressure dome remains unchanged from the licensed power level, and the dome temperature is also unchanged. The Nine Mile Point 2 proposed EPU does not change the current plant maximum normal operating reactor dome pressure, and it increases the original thermal power by 20 percent, with a maximum steam and feedwater flow increases of nearly 24 percent. Therefore, we found that it meets the limitations of the topical reports.

In addition to the main steam and feedwater piping, which are the main systems that are affected by the EPU due to its increase in the flows, other piping systems that are mostly affected by the EPU due to increased system temperatures and pressures within those systems are the extraction steam, the feedwater heater vents and drains, the moisture separator heater vents and drains, and the auxiliary condensate.

The licensee's evaluation of flow-induced vibration levels for piping follows the vibration acceptance criteria found in ASME OM-S/G Part 3, which provides requirements for pre-operational and initial start-up vibration testing of nuclear power plant pipe and systems.

The OM-S/G Part 3 provides monitoring requirements, acceptance criteria, and it includes equations for calculating the vibratory alternating stress for Class 1 and Class 2 and 3 piping, and for thickness for Class 1 piping. It also contains guidance and visual inspection methods, displacement methods and vibrational deflectional values for various pipe sizes and spans.

The structural evaluations for the system structures and components under EPU conditions employ the current plant design base methodology and acceptance criteria. The structural evaluations also met design basis code and record allowable values. That's why we found there is reasonable assurance that the plant SSAs (ph) important to safety as structurally adequate to perform the internal design functions under the EPU conditions.

MEMBER SHACK: But you can't really be sure they're not going to need any modifications to the pipe supports until you run that FIV test, right? I mean they could well need to do something.

MR. TSIRIGOTIS: The SIV test?

MEMBER SHACK: The FIV test.

MR. TSIRIGOTIS: Oh yes. You are right about that. During the -- so far, the evaluations that they have done, they have found out that they don't need any piping modifications or any support modifications or additions. During the start-up testing, they had a plan in place which they will monitor the vibration levels, and if there is a need for any modifications through the corrective code, they will provide corrective actions to do that work.

CHAIRMAN ARMIJO: But that will indicate that their analysis wasn't really that good --

(Simultaneous speaking.)

MR. TSIRIGOTIS: They have already done -- I understand what you're saying. They have already done walk-downs to establish the baseline. From those walk-downs, they have identified whether there is an issue with the vibration levels that the pipe's supposed to see.

CHAIRMAN ARMIJO: That's where they'll put instrumentation.

MR. TSIRIGOTIS: Right, right. They have instrumentation. They have -- there are locations which have been gauged, strain gauges, and --

CHAIRMAN ARMIJO: Accelerometers.

MR. TSIRIGOTIS: Accelerometers where needed. The OM-S/G Part 3 is being applied in just about every power uprate, and it's during the initial start-up also for the plants.

MEMBER SHACK: What's been the experience? I mean if they found they need to add supports, or the analysis have been generally satisfactory?

MR. TSIRIGOTIS: So far from what we've seen in the power uprates from the walk-downs, they haven't identified, from what I know at least, they haven't identified an issue where they needed to add something, mainly because when they established the baseline, they project that baseline to the EPU flows, with velocity square, which is customary to do so. If they find an issue, then they take a corrective action.

DR. BONACA: I have a question regarding environmental qualification.

MR. TSIRIGOTIS: Uh-huh.

DR. BONACA: There is a statement in the SER that says that, you know, for inside containment, licensee noted that post local conditions, radiation levels will increase above the levels using the current EQ program. Then it says the NRC staff reviewed the increase EQ evaluation and confirmed that the increase should not affect the qualification of the equipment. What would be the basis for that?

MR. TSIRIGOTIS: I'm sorry. I can't hear you very well. Are you reading from the SER, from the staff SER?

DR. BONACA: Yes, yes, page 57.

MR. TSIRIGOTIS: 57. That's not my page.

(Simultaneous speaking.)

DR. BONACA: Yeah, inside containment.

MR. TSIRIGOTIS: Is that Section 2.2.5, seismic and dynamic qualifications of mechanical and electrical equipment?

DR. BONACA: It must be, yes. Do you have the page?

(Off record discussion.)

MR. TSIRIGOTIS: I don't see that. That's not in my evaluation. That's the same part. I think it might be in the electrical part. Anyway, it's in the SER.

(Simultaneous speaking.)

DR. BONACA: -- stand by. Clearly, the radiation field must be higher. Why is --

(Simultaneous speaking.)

MR. TSIRIGOTIS: (reading to self)

MR. GUZMAN: What we can do is we'll take that -- I just need to get back to the safety evaluation, and find out the technical staff --

DR. BONACA: What's the basis for it, yes.

MR. TSIRIGOTIS: This is not my writing. I will find out whose review this falls under, and we'll get back to you.

DR. BONACA: Okay, I appreciate it. Thanks.

CHAIRMAN ARMIJO: Okay. Any other questions from the committee?

(No response.)

CHAIRMAN ARMIJO: All right. I think we're ready to move on to the next topic.

MR. GUZMAN: The next topic is intended to be in closed session.

CHAIRMAN ARMIJO: So we're going to go into closed session. Again, remind everyone here, it should only be folks from Nine Mile and their consultants, and make sure the bridge line is closed.

(Whereupon, at 1:15 p.m., the meeting adjourned to closed session.)

O P E N S E S S I O N

4:35 p.m.

CHAIRMAN ARMIJO: If not, we're going to come out of closed session and do a public session, and at this point, I'll ask if there are any comments for members of the public, either on the bridge line or open the door so they can come in to this room. Is there anyone on the bridge line who would like to make a comment? If so, please identify yourself.

(No response.)

CHAIRMAN ARMIJO: Maybe the bridge line isn't open yet. Peter will -- is it open now?

DR. YARSKY: The bridge line is open.

CHAIRMAN ARMIJO: Okay. The bridge line's open. Is there anyone, a member of the public, who would like to make a comment concerning this review? If so, please identify yourself.

(No response.)

CHAIRMAN ARMIJO: Okay. How about someone in here, n this meeting room?

(No response.)

CHAIRMAN ARMIJO: Okay. I'm going to take that as there's no comment from the public. At this point, I'd like to turn it over, just as far as Subcommittee discussion. I'd just like to go around the table, and see if there's any added points that the members would like to make, and then after that, maybe try and give the staff and licensee some guidance on the full Committee meeting, because this obviously all has to get done in two hours. So that's the challenge. Joy?

MEMBER REMPE: Some education. During the discussion, this last topic, there was a mention of an upcoming audit. Could you provide a little more background? I think Stephen was the one who mentioned it and was talking to people in the crowd. What exactly is going to happen here? Is there going to be an end to end audit or what exactly is it?

DR. SHAH: This is Vik Shah. I think we have to plan it out before we finalize what kind of audit we'll be doing. I think we are going to review the (breaking up) 194, and during that review, we will be having an audit. I think we will have a (breaking up).

MEMBER REMPE: I'm having trouble understanding you.

CHAIRMAN ARMIJO: It's breaking up.

DR. BASAVARAJU: We have already submitted a topical report.

MEMBER REMPE: Who submitted it? Say it

again. Who submitted the topical?

DR. BASAVARAJU: BWR Vessel Internals program. They submitted a topical report, BWR 194.

MEMBER REMPE: Okay.

DR. BASAVARAJU: And it just came in for review, and that is the one which will summarize and give all the steam dryer evaluation, the ACM and the structural evaluation. So during that review, we were planning to have an audit, but we have not still identified the times or extent. So that's --

MEMBER REMPE: Thank you.

MEMBER BANERJEE: What is the time scale? I don't want to go out of turn, because this is new information. What is the time scale for review of the topical and I assume it will come to us as well?

DR. BASAVARAJU: Yes.

DR. SHAH: It's about two years, right Pani?

DR. BASAVARAJU: Yes. It's a topical report. Because this is an important topical report, we may accelerate it. But the typical topical report reviews, NRC's time is for two years.

CHAIRMAN ARMIJO: Okay. So it won't help us for Nine Mile. Bill, nothing. Said? No, any comments.

DR. BONACA: No. Well, a little input. I thought in general that this was a good application. I felt it was thorough and the steam dryer issue, there are a number of questions which were raised today. I think that I feel pretty comfortable with what I saw.

CHAIRMAN ARMIJO: Thank you.

DR. BONACA: Anyway, I will make comments to you.

CHAIRMAN ARMIJO: Yes, okay, in your report. Mr. Wallis?

DR. WALLIS: Well, I thought on most issues, Nine Mile Point people did a very good job. I'm still working on the steam dryer. I'm still puzzled, because I'm told that this small scale test was only used to establish resonance, and yet I read the report, the objective was to develop a bump-up factor relating (coughing) to those anticipated at EPU, to use in the acoustic circuit model.

I mean the whole thing says, the whole purpose of the report is to develop numbers to put into a model. I'm really puzzled by this assertion that none of that was the case. I don't understand that, and I'm still working on the numbers. I have learned some things which have been very helpful about some of the issues I raise, and I thank the participants for doing that for me.

CHAIRMAN ARMIJO: Sanjoy.

MEMBER BANERJEE: Nothing more than I said already.

CHAIRMAN ARMIJO: Nothing more. Jack?

MEMBER SIEBER: I have no comments or questions at this time.

CHAIRMAN ARMIJO: Okay, all right. Well, my view is I think the Nine Mile people and staff are very well prepared for this. I think we can beat the steam dryer to death, but and we obviously can't, won't be able to spend that much time at the full committee meeting.

So that between the staff and the Nine Mile, I think we really need one good presentation, without any repetition at all. So you're going to have to sort that out. I think the plant's in -- the impression I got from the presentation, the plant's in very good shape for EPU.

I think the work you've done on the materials, and on the various upgrades and modifications, that goes a long way to making me feel pretty comfortable --

MEMBER SHACK: And they use that good

barrier clad --

CHAIRMAN ARMIJO: And they use my cladding, so that's great.

(Laughter.)

CHAIRMAN ARMIJO: But I'm still surprised at how conservative the fuel design is. It's got more capability, but that's okay. But overall, I think you're well-prepared. I think our problem will be to manage the time, so that you get, the full Committee gets a good feel for the entire plant, and that we don't let the steam dryer dominate everything.

So that's going to be hard to do, but I think since so many of us have heard this presentation, and as soon as we get Mr. Wallis' report and Mario's reports, we probably can sort out our questions and focus down. But overall, I think you're well-prepared. We'll work with Rich.

MR. GUZMAN: I was hoping I could close out one quick action item, and I wanted to make sure I responded to Dr. Bonaca's question and concern on part of the safety evaluation, and I'll just restate it. The section is part of the Electroengineering review, under Environmental Qualifications for Electrical Equipment.

The statement said that "The NRC staff

reviewed the licensee's environmental qualification evaluation, and confirmed that the increase should not affect the qualification of the EQ equipment located inside containment." The question is why? Because they should not. Does it or does it not?

The staff recognizes the obscurity in that wording, and so it was in error, and we should or we will correct that, and --

MEMBER SHACK: No, I think the question was that the statement was made that it exceeded the environmental qualification, and then the statement followed that --

MR. GUZMAN: Okay.

MEMBER SHACK: It's still all right.

MR. GUZMAN: Still all right, okay.

MEMBER SHACK: It was that first line that was the killer.

MR. GUZMAN: Okay, yes. So right. So the preceding statement says the licensee noted that the radiation levels would increase above the levels used in the current EQ program.

MEMBER SHACK: Right.

MR. GUZMAN: And NRC staff review confirmed that these increases should not affect the qualifications. So I guess the question is why, and if you go further in the safety evaluation, I mean recognizing that the "should not" was not the appropriate words. It should have said, for lack of a better word, "would not" or "will not."

But further down in the safety evaluation, it does go into the, you know, that the staff reviewed the licensee's assessment of the effects of the proposed EPU on EQ equipment, and ultimately we reviewed it against 10 C.F.R. 5049, which is the electrical equipment qualification.

MEMBER SHACK: Maybe you should give something like "Despite this step."

MR. GUZMAN: Right. So we recognize that the wording needed to be tightened up. It certainly should have been more definitive and explain it. So we will make sure on the final --

DR. BONACA: This is an example of the way that the information is provided. There were two other, three other in the SER, I believe goes to the outside containment portion. There is a statement that simply says that's okay, and the question is why is it okay?

I mean it's counterintuitive that if you have a higher radiation field, it doesn't make any difference. There has to be some reason why, and I mean the qualification exceeded the value. Therefore, there was margin. But something should be said.

MR. GUZMAN: And as you know, I mean the intent is to give you the best product that we can for the draft safety evaluation. But in parallel, what we tried to is we actually send the draft safety evaluation to the licensee. They provide some comments to us and we will incorporate those comments, as well as another round of quality check by the staff, to strengthen the product, which would be the final safety evaluation.

So there will be some changes, and we will note that one, as well as the other ones that you did note.

DR. BONACA: Okay, thank you.

CHAIRMAN ARMIJO: Okay. Well with that, I think I'd again like to thank Nine Mile Point and the staff. Good presentations. Good discussion, and with that we're going to adjourn the meeting. Thank you.

(Whereupon, at 4:46 p.m., the meeting was adjourned.)

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UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

ACRS Subcommittee on Power Uprates

NRC Staff Review
Nine Mile Point, Unit 2
Extended Power Uprate

October 5, 2011

Opening Remarks

Louise Lund

Deputy Director

Division of Operating Reactor Licensing

Office of Nuclear Reactor Regulation

Opening Remarks

- NRC staff effort
  - Pre-application review and public meetings
  - Acceptance Review
  - Requests for additional information
- Challenging review areas included:
  - Steam dryer stress analysis
  - Thermal Hydraulic Design: Stability / ATWS-Stability
  - Interim Methods: Applicability of GE Methods to Expanded Operating Domains
- Draft SE – no open technical issues

Introduction

Rich Guzman

Senior Project Manager

Division of Operating Reactor Licensing

Office of Nuclear Reactor Regulation

Introduction

- Objective
- Background
  - NMPNS EPU Application – May 27, 2009
  - 3467 to 3988 MWt, 15 % increase (521 MWt)
  - 20 % increase above original licensed thermal power
- EPU Review Schedule
  - Followed RS-001
  - No linked licensing actions under review
  - Supplemental responses to NRC staff RAIs
- EPU Implementation

Topics for Subcommittee

- NMPNS EPU Overview
- Anticipated Transient without Scram and Stability
- Fuel Methods - IMLTR
- Materials and Mechanical & Civil Engineering
- Steam Dryer Analysis
- Review of open items / Conclusions

Nine Mile Point, Unit 2
Extended Power Uprate
ACRS Subcommittee Meeting

Materials Engineering

Patrick Purtscher
Vessel & Internals Integrity Branch
October 5, 2011

Reactor Vessel Embrittlement

- EPU increases total fluence on RV
- RV Material Surveillance Program,

  Uses BWRVIP ISP, but not a host plant, still has 2 capsules in RV
- Meets Appendix G requirements for

  P-T limits, USE projections, circ weld inspection exemption, significant margins remain

Internals and Core Support Materials

- EPU increases total fluence on RV Internals
- Top guide, shroud, and core plate all exceed IASCC threshold for susceptibility
- BWRVIP-62, Category 3b plant – uses NMCA for mitigation of SCC, follows EPRI guidelines for effectiveness

Conclusion

- EPU has minimal impact on RV embrittlement issues
- Three RVI components exceed threshold for IASCC, but adequately managed;

  Core plate – BWRVIP-25-A

  Top guide – BWRVIP-26-A

  Shroud – BWRVIP-76-A

QUESTIONS

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Nine Mile Point, Unit 2
Extended Power Uprate
ACRS Subcommittee Meeting

Mechanical & Civil Engineering Review

Alexander Tsirigotis
Mechanical & Civil Engineering Branch
October 5, 2011

Review Scope

- EPU impact on structural integrity of systems, structures, and components (SSCs):
  - Pressure-retaining components and their supports
  - RPV and supports
  - Control rod drive mechanisms
  - Reactor recirculation pumps and supports
  - RPV internals and core supports
  - Seismic and dynamic qualification of mechanical and electrical equipment.

Review Results

- Piping systems that are mainly affected from the EPU:
  - Main steam, condensate, feedwater, extraction steam and heater vents and drains.
  - Evaluation for FIV levels of piping in accordance with the ASME OM –S/G Part 3.
  - There are no modifications to piping and pipe supports that are required due to EPU.
- Structural evaluations of SSCs at EPU conditions employed current plant design basis methodology and acceptance criteria.
- Structural evaluations met design basis code allowable values.

Conclusion

- Reasonable assurance that plant SSCs important to safety are structurally adequate to perform intended design functions under EPU conditions.

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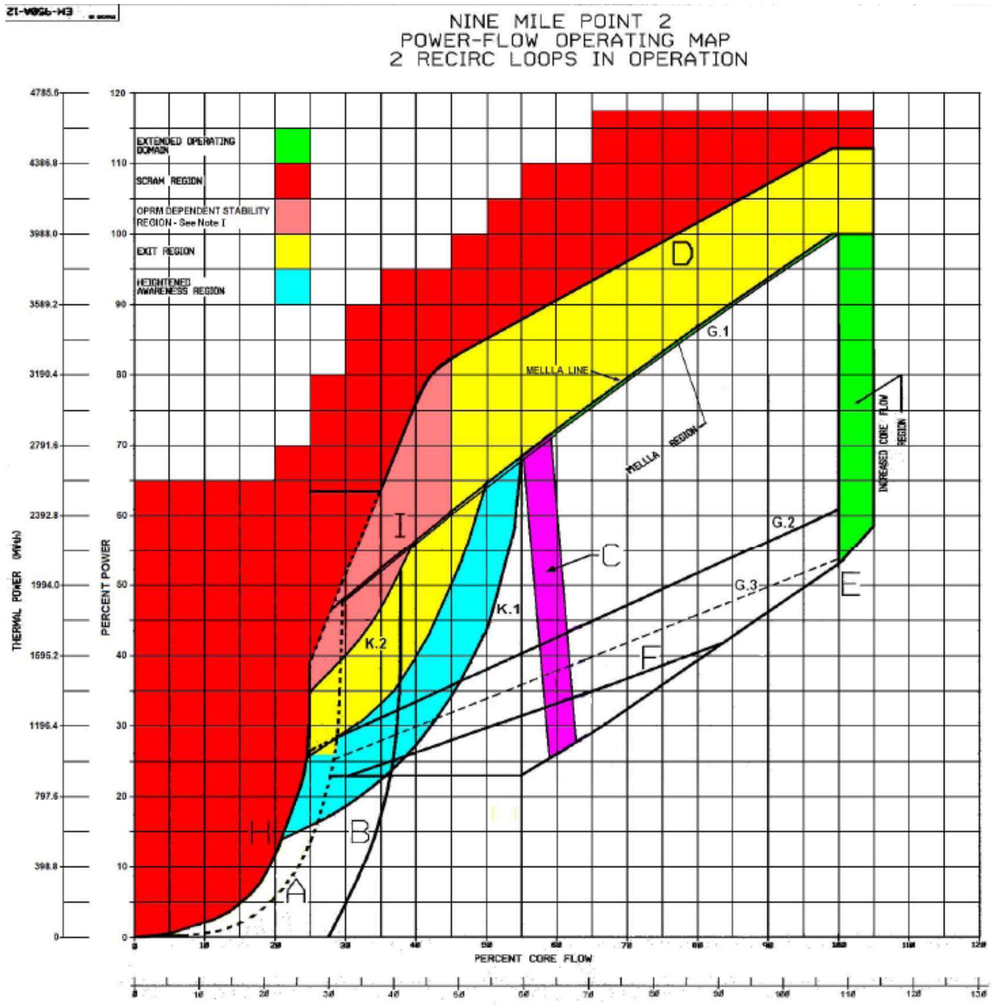
QUESTIONS

Power / Flow Operating Map



A. ORIGINAL ESTIMATED NATURAL CIRCULATION, LOW RECIRC PUMP SPEED BOTH FCV'S MIN. POS. LINE (SEE NOTE H)
B. LOW RECIRC PUMP SPEED BOTH FCV'S MAX. POS
C. EXPECTED POWER/FLOW RANGE AT RATED RECIRC PUMP SPEED, BOTH FCV's AT UPSHIFT POSITION TYPICAL FOR POWER ASCENSION
D. FLOW-BIASED THERMAL POWER SCRAM
E. CAVITATION INTERLOCKS
F. MINIMUM RODLINE FOR TRANSFER TO RATED RECIRC PUMP SPEED (47%)
G. RODLINES 1. 100% 2. 61% 3. 54%
H. LOWER BOUND OF NATURAL CIRCULATION, LOW RECIRC PUMP SPEED BOTH FCV'S MIN. POS.
I. THIS REGION IS AN EXIT REGION WITH ≥3 OPRMs OPERABLE, OTHERWISE (<3 OPRMs OPERABLE) THIS REGION IS A SCRAM REGION (SEE NOTE K)(TECH SPEC 3.3.1.1).

% ROD LINE = PERCENT POWER / (.22191 + (8.9714 X 10^-3) PERCENT CORE FLOW - (0.1905 X 10^-5) PERCENT CORE FLOW^2)

J. THE ROD LINES SHOWN ARE BASED ON CALCULATIONS THAT ARE NOT CYCLE SPECIFIC AND THAT ACTUAL PLANT RESPONSE MAY NOT FOLLOW THE ROD LINE EXACTLY DURING A REACTOR RECIRCULATION FLOW CHANGE.
K. ALTERNATE METHODS FOR BACKUP STABILITY PROTECTION MEASURES FOR LESS THAN 3 OPRM's OPERABLE (TECH SPEC 3.3.1.1)
K.1 BACKUP STABILITY PROTECTION IMPLEMENTATION LINE
K.2 BACKUP STABILITY PROTECTION SCRAM LINE (SEE NOTE I)

THIS IS A COLOR CODED DRAWING

SAFETY RELATED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ACRS Subcommittee Presentation
Nine Mile Point 2
Extended Power Uprate

October 5, 2011


Nine Mile Point 2
Extended Power Uprate

NMP2 EPU Overview

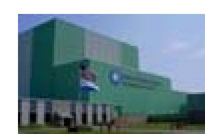
Sam Belcher

SVP-Site Operations

NMP2 EPU Agenda

— Overview	Sam Belcher
— Plant Modifications	Dale Goodney
— Power Ascension Testing	Phil Amway
— Anticipated Transient Without Scram (ATWS) and Stability	Phil Amway
— CLOSED SESSION - Fuel Methods (IMLTR)	Phil Wengloski
— Material, Mechanical/Civil Engineering Topics	George Inch
— CLOSED SESSION - Steam Dryer Analysis	George Inch



NMP2 EPU Overview

- GE BWR 5
- Mark II Containment
- Thermal Power
  - Original License Thermal Power (OLTP) 3323 MWth
  - Current License Thermal Power (CLTP) 3467 MWth
    - Stretch Uprate 104.3% (1995)
  - EPU Thermal Power (120% OLTP) 3988 MWth
    - Implement 2nd Quarter 2012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NMP2 EPU Overview (cont'd)

- Attributes of the NMP2 constant pressure power uprate:
  - NMP2 is not requesting any Containment Accident Pressure (CAP) credit to support ECCS NPSH
  - No new fuel introduction; the current core and the EPU core are composed entirely of GE 14 fuel
  - The Alternative Source Term for accident radiological consequences was previously implemented using the EPU power level as a base assumption
- NMP2 has implemented Maximum Extended Load Line Limit Analysis (MELLLA) expanded operating domain



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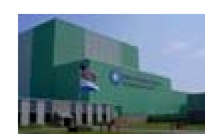

NMP2 EPU Overview (cont'd)

- The NYISO has reviewed and approved the EPU full power output - no grid modifications are necessary
- The first two phases of EPU modifications have been installed
- The third and final phase of modifications needed to support EPU operation will be complete by 2nd Quarter of 2012




Nine Mile Point 2
Extended Power Uprate

Plant Modifications

Dale Goodney

EPU Lead Project Engineer

Plant Modifications

- General Approach
- Plant Parameters
- Installation Timeline
- Major Modifications
- NMP2 Plant Improvements


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Plant Modifications - General Approach

- Engineering studies were performed to evaluate structures, systems and components to determine the plant's ability to operate at EPU conditions
  - Analyzed effects of increase in steam flow, feedwater flow, reactor power and electrical output
  - Evaluations were based on NEDC-33004P-A, "Licensing Topical Report Constant Pressure Power Uprate," Revision 4 (CLTR)
  - Analyses are based on the target power level of 120% OLTP
  - Operating Experience was evaluated and applied



Plant Modifications - General Approach (cont'd)

- Design and operating margins were identified and evaluated for both NSSS and BOP systems
- Over 20 physical plant modifications were described in the License Amendment Request
  - Restore Material Condition
  - Instrumentation for data collection and analysis
  - Upgrades to restore design and operating margin at EPU conditions
- Installation began in 2007 and will continue through 2012 Refueling Outage


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Plant Modifications - Plant Parameter Changes

Parameter	CLTP (104.3% OLTP)	EPU (120% OLTP)
Thermal Power (MWth)	3467	3988
Reactor Pressure (psia)	1035	1035
Rated Steam Flow (Mlb/hr)	15.002	17.636
Rated Feedwater Flow (Mlb/hr)	14.970	17.604
Generator Output (Mwe)	1211	1369
Feedwater Temperature (°F)	425.1	440.5



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Plant Modifications - Installation Timeline

2007 and 2008	2010 and 2011	2011 through 2012 Refueling
• Main Steam (MS) Line Vibration Monitoring Strain Gages • Partial Bypass Around the Condensate Demineralizers	• Replace 3rd Point Feedwater Heater • Replace Feedwater Heater Drain Pumps and Motors • Install Piping Vibration Monitoring • Install Shielding for Equipment Qualification	• Upgrade Feedwater Pumps and Gear Sets • Replace Feedwater Pump Motor Cables • Recirculation Runback Initiation and Runback Rate • Replace High Pressure Turbine • Replace Low Pressure Turbine Cross Around Relief Valves • Replace Low Pressure Turbine Atmospheric Relief Diaphragms • Steam Dryer Modifications • Feedwater Heater Rerate • Generator Isolated Phase Bus Duct Cooling Improvements • Improve Main Transformer Cooling • Instrument Replacement and Scaling • Improve Turbine Building HVAC • Turbine Building Closed Loop Cooling Enhancements • Extraction Steam Expansion Joint Replacement • Main Steam/Feedwater Pipe Supports


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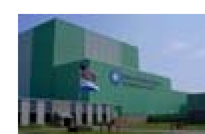

Major Plant Modifications

- Condensate and Feedwater
  - Feedwater Pump Upgrades
  - Heater Drain Pumps and Motors
  - Reactor Recirculation Runback
- Steam Path
  - High Pressure Turbine
  - Cross-Around Relief Valves
  - Moisture Separator Reheater and 5th/6th Point Feedwater Heater Requalification
  - Steam Dryer



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Major Plant Modifications (cont'd)

- Electrical/I&C
  - Isophase Bus Duct Cooling
  - Main Transformer Cooling
  - Technical Specification Instrument Setpoints
  - BOP Instrument Rescaling and Setpoints
- Auxiliary Support Systems
  - Turbine Building HVAC
  - Turbine Building Closed Loop Cooling



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NMP2 Plant Improvements

- NMP2 has implemented, or is planning to implement prior to EPU, a number of upgrades to restore margin, improve equipment reliability and reduce risk. Examples are:
  - Replaced Third Point Feedwater Heaters in 2010
  - Increased Standby Liquid Control Relief Valve Margin in 2010
  - Performed Cooling Tower Upgrades in 2008 and 2010
  - New Feedwater Pump Seals in 2012
  - Replace Jet Pump Inlet Mixers in 2012
  - Several PRA-related risk reduction improvements. Since 2008, Core Damage Frequency (CDF) has been reduced by 78%


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Nine Mile Point 2
Extended Power Uprate

Power Ascension Testing

Phil Amway
EPU Operations Lead - SRO

Power Ascension Testing

- Preparation
- Approach
- Schedule
- Test Plan
- Acceptance Criteria and Actions



Power Ascension Testing - Preparation

- Test Objective Development
  - Satisfactory Equipment Performance
  - Careful, Monitored Approach to EPU Power
  - Meet Established Requirements
- Roles & Responsibility Development
- Industry Benchmarking
- Test Plan and Implementing Test Procedure Development
- Power Ascension Test Training

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Power Ascension Testing – Approach

- Similar to approach used in other EPUs – Incremental Testing
- Baseline data at 75%, 90%, 95% and 100% CLTP
- Greater than 100% CLTP
  - Data acquisition performed in incremental steps of 1% and 2.5%
  - Active Testing and NRC Data Review at incremental steps of 5%




Power Ascension Testing – Approach (cont'd)

- No Large Transient Testing
  - Industry OE indicates that plants will continue to respond to transients as designed following EPU implementation
  - Plant specific OE at 104.3% OLTP (Generator Load Reject and MSIV Closure)
  - NMP2 has previously performed Large Transient Testing and documented results
  - Plant operators will be trained on large transient events in the simulator
  - Analytical methods and training facilities adequately simulate large transient events



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Power Ascension Testing – Schedule

- Data collection – 1% intervals
- Data evaluation – 2.5% intervals
- EPU Major Testing Plateau – 5% intervals
  - Passive data collection (e.g. vibration, radiation monitoring, plant parameter monitoring)
  - Active control system stability dynamic testing
    - Pressure regulator step test
    - Feedwater level control step test
  - Data Analysis
  - Plant Management (PORC) Review
  - NRC Review


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EPU Power Ascension Testing – Test Plan
Major Testing

Test #	Description	70% CLTP up to 85% CLTP as evaluated	75% CLTP	80% CLTP	85% CLTP	90% CLTP	95% CLTP	100% CLTP	102.5% CLTP	105% CLTP	107.5% CLTP	110% CLTP	112.5% CLTP	115% CLTP	115% CLTP (@ point "D" 100% EPU power and 99% flow)	115% CLTP (@ point "F" 100% EPU power and 105% flow)	First controlled shutdown after EPU implementation
1A	Chemical and Radiochemical Water Chemistry and Gaseous Effluent Data Collection							X		X		X		X			
1B	Steam Dryer/Separator Moisture Carryover							X		X		X			X	X	
2	Radiation Measurement Data Collection							X		X		X		X			
10	IRM Overlap																X
11	LPRM Calibration	X												X			
12A	APRM Calibration/Functional Test							X						X			
12B	APRM Calibration to Heat Balance					X		X		X		X		X			
18	TIP Probe Uncertainty													X			
19	Core Performance					X	X	X	X	X	X	X	X	X			
22A	Pressure Regulator Testing					X		X		X		X		X			
23A	Feedwater Level Control Testing					X		X		X		X		X			
23B	Feedwater Flow Data Collection					X		X		X		X		X			
23C	Max Feedwater Runout Data Collection					X		X		X		X		X			
24A	Turbine Valve Surviellance Margin Test	X															
24B	MSIV Partial Closure Surveillance Margin Test							X		X		X		X			
35	Recirc Drive Flow Gain Adjustment Calibration							X						X			
74	Offgas System Performance					X		X		X		X		X			
75	Drywell Cooling				X	X		X		X		X		X			
100A	Piping VibrationTesting: MS, FW, and other key systems > 100% CLTP: ▪ Data Collection Every 1% ▪ Data Evaluation Every 2.5%		X			X	X	X	X	X	X	X	X	X			
100B	Dryer/Separator Stress Analysis > 100% CLTP: ▪ Data Collection Every 1% ▪ Data Evaluation Every 2.5%		X			X	X	X	X	X	X	X	X	X			
101	Key Parameter Monitoring					X		X	X	X	X	X	X	X			
102	Reactor Recirculation Performance Data			X		X		X		X		X		X			


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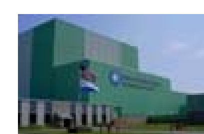

Power Ascension Testing – Acceptance Criteria and Actions

- **Level 1 Acceptance Criteria:** A limit associated with plant safety
- If Level 1 criterion is not met:
  - Abort the test
  - Reduce power to last known safe condition
  - Use the Corrective Action Program to evaluate the condition and to determine and implement required actions
  - Repeat testing to verify that the Level 1 criterion is satisfied
  - Document problem resolution




Power Ascension Testing – Acceptance Criteria and Actions (cont'd)

- **Level 2 Acceptance Criteria:** A limit associated with plant or equipment performance that does not meet design expectations but is not immediately adverse to plant safety
- If Level 2 criterion is not met:
  - Place the test on hold and confirm the plant is in a safe condition
  - Use the Corrective Action Program to evaluate the condition and to determine and implement required actions
  - Repeat testing to verify the Level 2 criterion is satisfied unless the as-found condition is determined satisfactory
  - Document problem resolution


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Power Ascension Testing – Acceptance Criteria and Actions (cont'd)

- **Other Acceptance Criteria:** A limit associated with plant surveillance requirements, plant operating procedures, rounds or alarm responses
- When this criteria is not met, plant procedures will be followed



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Nine Mile Point 2
Extended Power Uprate

Long Term Stability Solution Option III and ATWS – Stability Events

Phil Amway

EPU Operations Lead - SRO

Long Term Stability Solution – Option III/ATWS

- NRR Audit at Nine Mile Point 2
- Long Term Stability Solution – Option III
  - Oscillation Power Range Neutron Monitor (OPRM)
  - OPRM Settings
  - Backup Stability Protection (BSP)
  - 2003 NMP2 Stability Event
  - Effects of EPU on the Long Term Stability Solution
- Impact of EPU on ATWS – Stability Events
  - NMP2 ATWS Mitigation Design Features
  - Preparation for Simulator Demonstration
  - MSIV Closure with Failure to Scram
  - Turbine Trip with Failure to Scram
  - Conclusions




NRR Audit at NMP2

- Performed October 28, 2009 to demonstrate procedure actions and operator response to ATWS transients at EPU conditions conform to regulatory requirements
- Reviewed implementation of Long Term Stability Solution – Option III
- Observed operator performance in plant reference simulator
  - MSIV Closure with Failure to Scram
  - Turbine Trip with Failure to Scram
- Included review of related procedures and mitigation strategies
- Requested follow up information when plant reference simulator was modified to provide ATWS - Stability transient response data


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Oscillation Power Range Monitor (OPRM)

- 1998 – NUMAC Power Range Neutron Monitor OPRM hardware installed (Amendment 80) - system tuning performed for plant specific settings
- 2000 – Reactor Protection System (RPS) OPRM trips armed (Amendment 92)
- 2002 – Implemented Plant Specific Delta CPR Over Initial CPR Versus Oscillation Magnitude (DIVOM) curve per GE Safety Communication 01-01, Stability Setpoint Calculation using Generic DIVOM Curve
- 2003 – Implemented filter frequency and period tolerance settings per GE Safety Communication 03-20, Stability Option III Period Based Detection Algorithm Allowable Settings


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OPRM Settings

- Cycle specific DIVOM analysis is performed using TRACG methodology
- Cycle specific amplitude setpoint is defined in the Core Operating Limits Report
- OPRM trips will be enabled $\geq$26% RTP and $\leq$60% recirculation drive flow to maintain the same enabled region in terms of MWth power




Backup Stability Protection (BSP)

- BSP regions are determined using cycle specific ODYSY decay ratio calculations
- BSP regions are defined on plant power/flow operating maps
- Operator actions are defined in plant procedures with routine training reinforcement
- BSP exit region procedures are enforced at all times



2003 NMP2 Stability Event

- Component failure resulted in high to low speed transfer of both Reactor Recirculation pumps
- OPRM Period Based Detection Algorithm (PBDA) initiated an automatic reactor scram because of core wide oscillations
- The reactor was properly tripped by the PBDA
- Unexpected Confirmation Count (CC) resets occurred prior to the scram
- Post scram analysis determined that two parameter settings needed to be changed to address CC resets
- Parameter setting changes have been implemented per BWROG recommendations


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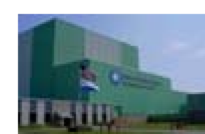

Effects of EPU on Long Term Stability Solution

- No methods changes for EPU
- Maximum rod line remains the same (MELLLA boundary)
- OPRM armed region maintains the same level of stability protection
- Cycle specific setpoint analysis will capture core design variations
- Option III long term solution remains unchanged
- Option III OPRM setpoints will be developed based on plant specific DIVOM curves for the EPU cycle specific reload analysis


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NMP2 ATWS Mitigation Design Features

- High RPV pressure initiates ATWS systems

  T=0 seconds

  – Automatic Alternate Rod Insertion (ARI)

  – Automatic Reactor Recirculation Pump Trip (RPT) to slow speed

  T=25 seconds and power >4%

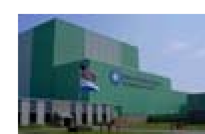
  – Automatic Feedwater Runback

  – Automatic Reactor RPT to off

  T=98 seconds and power >4%

  – Automatic Boron Injection

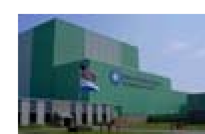


Preparation for Simulator Demonstration

- Simulator demonstration performed prior to operator training for EPU conditions
- Operating crew was provided with a briefing on EPU power level, steam and feedwater flows
- An SRO other than the EPU Operations Lead participated in the demonstration to avoid biasing operator response
- Simulator demonstration confirmed that current procedures and strategies successfully mitigate ATWS events



MSIV Closure with Failure to Scram

- 3988 MWth at 99% core flow (MELLLA boundary)
- Maximum Suppression Pool temperature 90°F
- Minimum Suppression Pool level 199.5 feet
- Maximum Service Water temperature 84°F
- No Control Rod Motion
- The above worst case conditions are consistent with design analysis inputs

MSIV Closure with Failure to Scram (cont'd)

- Both loops of Suppression Pool cooling in service at rated flow in 404 seconds vs action time of 1080 seconds
- Hot shutdown (<0.1% power) achieved in 406 seconds
- Peak Suppression Pool temperature remains below Heat Capacity Temperature Limit with 5°F margin
- Containment parameters remain well within design analysis
- Plant reference simulator critical parameter response closely matched the design analysis for this event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Turbine Trip with Failure to Scram

- 3988 MWth at 99% core flow (MELLLA boundary)
- Maximum Suppression Pool temperature 90°F
- Minimum Suppression Pool level 199.5 feet
- Maximum Service Water temperature 84°F
- No Control Rod Motion
- The above worst case conditions are consistent with design analysis inputs



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Turbine Trip with Failure to Scram (cont'd)

- Both loops of Suppression Pool cooling in service at rated flow in 425 seconds vs. action time of 1080 seconds
- Hot shutdown (<0.1% power) achieved in 465 seconds
- Peak Suppression Pool temperature remains below Heat Capacity Temperature Limit with 19°F margin
- Containment parameters remain within design analysis
- Plant reference simulator critical parameter response closely matched the design analysis for this event


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Conclusions

- Existing procedures, operator action times and strategies are effective in mitigating ATWS and ATWS instability transients
- NMP2 features an ATWS RPT function. As a result, transient power levels are primarily based on the maximum control rod line which is unchanged for EPU
- Operators can perform actions in a timely manner to bring the plant to safe shutdown


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Nine Mile Point 2
Extended Power Uprate

Material, Mechanical/Civil
Engineering Topics

George Inch
Principal Engineer - Mechanical/Structural Lead

RPV Internals

- Fluence
- Flow Induced Vibration (FIV)
- Structural Effects (Non-FIV)
- Intergranular Stress Corrosion Cracking (IGSCC) and Irradiation Assisted Stress Corrosion Cracking (IASCC)



RPV Internals – Fluence

- Irradiation assisted stress corrosion cracking (IASCC) fluence threshold is 5 E20 n/cm^2
- The following components exceeded the IASCC fluence threshold:
  - Top Guide (BWRVIP-26-A)
  - Shroud (BWRVIP-76-A)
  - Core Plate (BWRVIP-25-A)
- Continued implementation of the current program in accordance with the BWRVIP recommendations assures the prompt identification of any degradation of reactor vessel internal components
- NMP2 Utilizes Hydrogen Water Chemistry and Noble Metals
- Reactor vessel water chemistry conditions maintained consistent with the EPRI and established industry guidelines
- Peak fluence increase does not represent a significant increase in the potential for IASCC

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RPV Internals – Flow Induced Vibration

- Vibration levels for EPU were estimated by extrapolating vibration data from prototype plant or similar plants and on GEH BWR operating experience
- The following components were evaluated: a) shroud head and separator assembly; b) jet pumps; c) core delta P line; d) guide rods; e) in-core guide tubes and control rod guide tubes; f) jet pump sensing lines; g) feedwater sparger; h) fuel assembly, top guide, and core plate; i) RPV top head spare instrument nozzle; j) RPV top head vent nozzle; k) RPV head spray pipe and head spray nozzle; l) core spray piping
- Results show that continuous operation at EPU conditions does not result in any detrimental effects on the safety-related reactor internal components


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RPV Internals – Structural Effects (Non-FIV)

- Evaluations/stress reconciliation was performed consistent with the Design Basis Analysis
- Original configurations of the internal components utilized, unless a component had undergone permanent structural modification
- Effects of thermal-hydraulic changes due to EPU were evaluated
- EPU loads compared to those in the existing design basis analysis
- For increases in load, linearly scaled the critical/governing stresses based on increase in loads – compare resulting stresses against the allowable stress limits
- All stresses and fatigue usage factors are within the design basis ASME code allowable values
- RPV internal components demonstrated to be structurally qualified for operation at EPU conditions




RPV Internals – IGSCC and IASCC

- Procedurally controlled program consistent with BWRVIP issued documents
- Components inspected include: core spray piping and spargers; core shroud and core shroud support; jet pumps and associated components; top guide; lower plenum; vessel inner diameter attachment welds; instrumentation penetrations; steam dryer drain channel welds; and feedwater spargers
- Program assures prompt identifications of any degradation
- Hydrogen water chemistry and noble metal applications to mitigate the potential for IGSCC and IASCC
- Recent Control Blade Cracking OE
  - Not applicable to GEH Marathon “C” lattice models
  - GEH concluded no lifetime reduction for “C” lattice


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Fatigue Monitoring Program

- NMP2 implemented FatiguePro for fatigue monitoring in 2008 independent of EPU
  - Automated event tracking and usage based on cycle counting for most event
    - Assumed design basis event severity, records actual event severity
  - Stress based monitoring of FW nozzle location to improve the accuracy of usage


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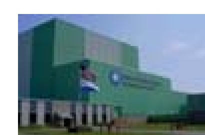
Fatigue Monitoring Program (cont'd)

- The EPU evaluation performed refined fatigue usage calculations for the FW nozzle
  - Reduced usage from original design basis
- FW nozzle high usage defined by off-normal rapid cycling events occurring during partial loss of feedwater heating and hot standby operation
  - EPU scaling factor small (between 6% and 15%)
  - Stress based fatigue monitoring anticipated to demonstrate usage less than 1.0 for 60 years

	CLTP 40 year CUF	EPU 40 year CUF
Carbon Steel Safe End	0.965	0.6537
Stainless Steel Clad	0.916	0.8299



Fatigue Monitoring Program (cont'd)

- FatiguePro implemented at NMP2 uses a single stress term for stress based monitoring
  - Simplified Greens function
  - RIS-2008-30 is applicable to NMP2
- NMP2 is following industry developments to reconcile RIS-2008-30 issue
  - FatiguePro 4
    - ASME Code Sub article NB-3200 fatigue analysis methodology
    - Environmental Fatigue rules (NUREG/CR-5704/6583/6909)
  - NMP2 is considering alternative confirmatory analyses as proposed by RIS-2008-30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NMP2 EPU
ATWS & Stability

Dr. Tai L. Huang (NRR/ADES/DSS/SRXB)
Dr. Jose March-Leuba (ORNL)

ACRS Subcommittee Meeting
October 5, 2011

Staff SER

- Staff has completed an SER with positive findings based on the review of available documents and a staff audit
  - Current LTS implementation (Sol III) is adequate for EPU
    - Satisfies GDC 10 & 12
    - Level of protection in EPU is similar to CLTP
  - Staff audit concluded that
    - NMP2 operators show good understanding of stability and ATWS issues for EPU.
    - Staff observations of operators' action in the simulator support the customary 120 s delay assumed for safety calculations
    - NMP2 EOPs are adequate for EPU

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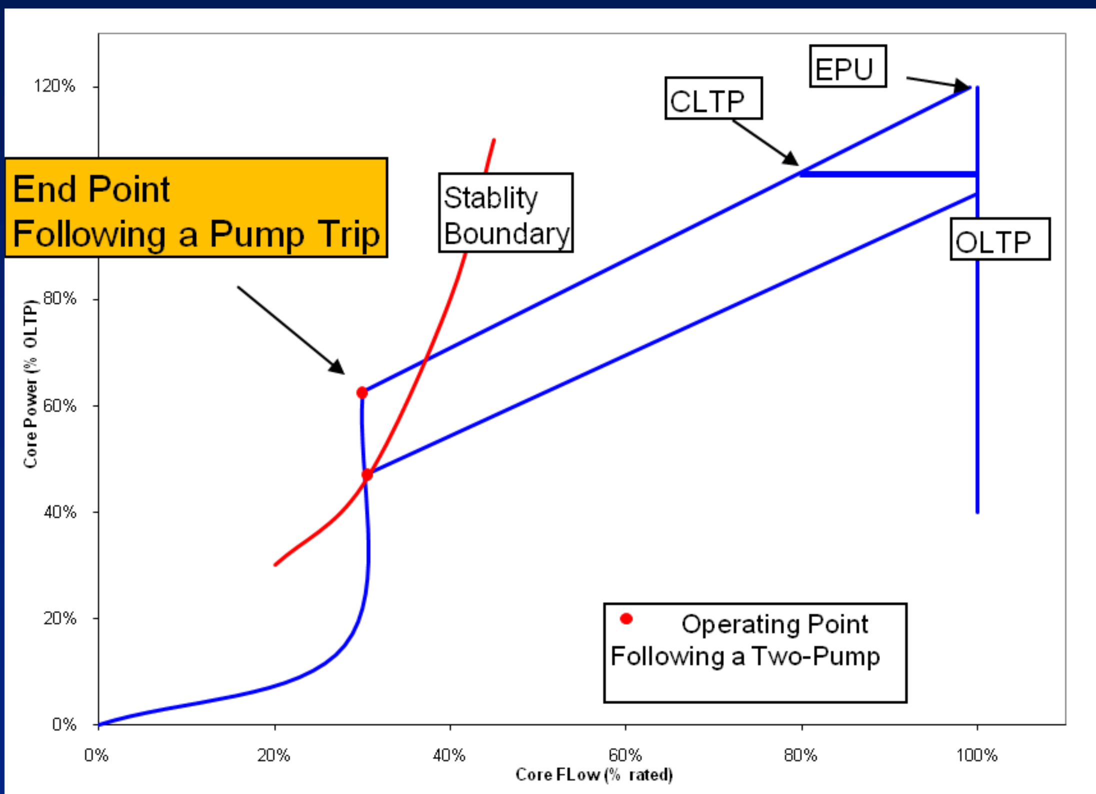
EPU Does Not Change the End Point After The Recirculation Pump Trip



- End Point is the same for CLTP and EPU because it is defined by
  - Natural Circulation
  - Subcooling (lower pressure of FW heating-steam)
- Stability characteristics of end point are similar

Stability

- LTS Option III installed since 1998, and armed since 2000
- Plant has good experience with Option III
  - 2003 NMP2 event was detected and scram actuated
    - very low amplitude oscillations, which kept on resetting the OPRM confirmation counts
  - Lessons learned (parameter settings) implemented at NMP2 per BWROG recommendations
- No impact expected for EPU
  - Option III and DIVOM methodology are applicable

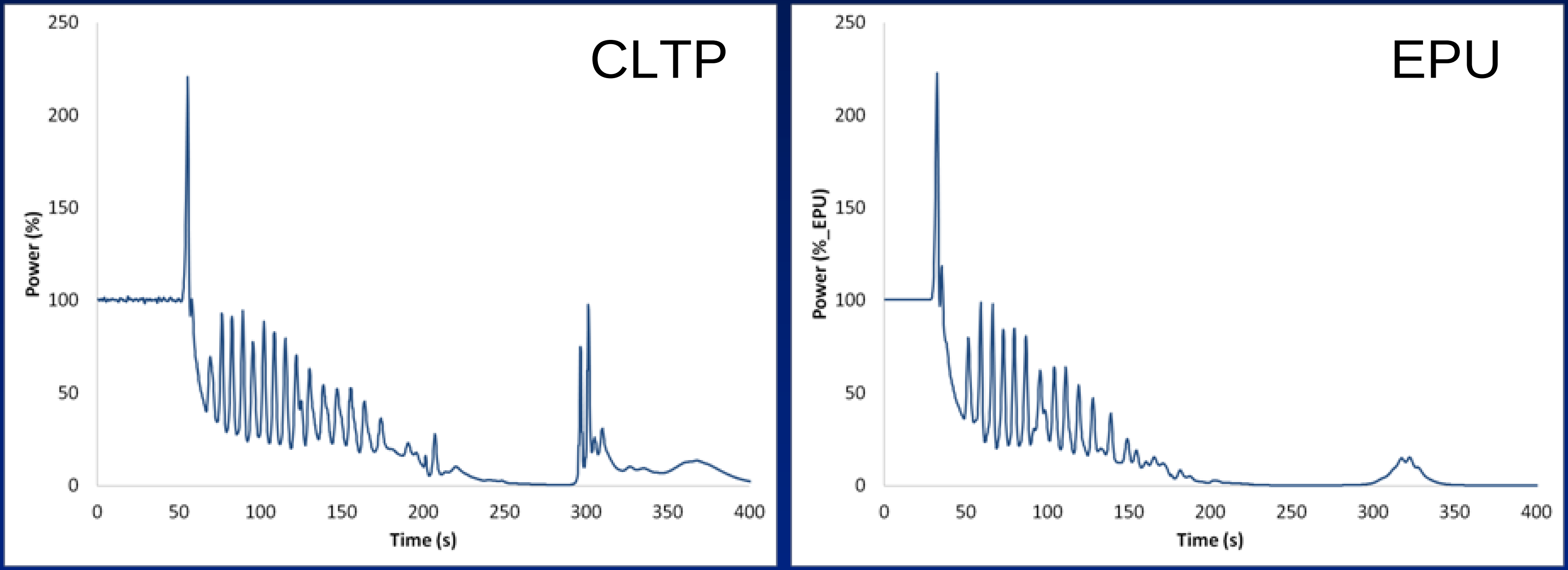
ATWS-Instability

- NMP2 has implemented latest EPG/SAGs
  - Early level reduction & boron injection are accomplished through automated ATWS actions if high pressure is detected with power >4%:
    - Automatic flow runback
    - Automatic boron injection
- NMP2 has excellent ATWS response:
  - SLC injection through HPCS (early shutdown)
  - 100% motor driven FW (sufficient HP inject capacity)
  - EOPs are reviewed every cycle, but are not affected significantly by EPU because boron is injected in HPCS and there is no need to define a HSBW.
    - EPU does affect HCTL slightly (from 140°F to 139°F)

Staff Audit

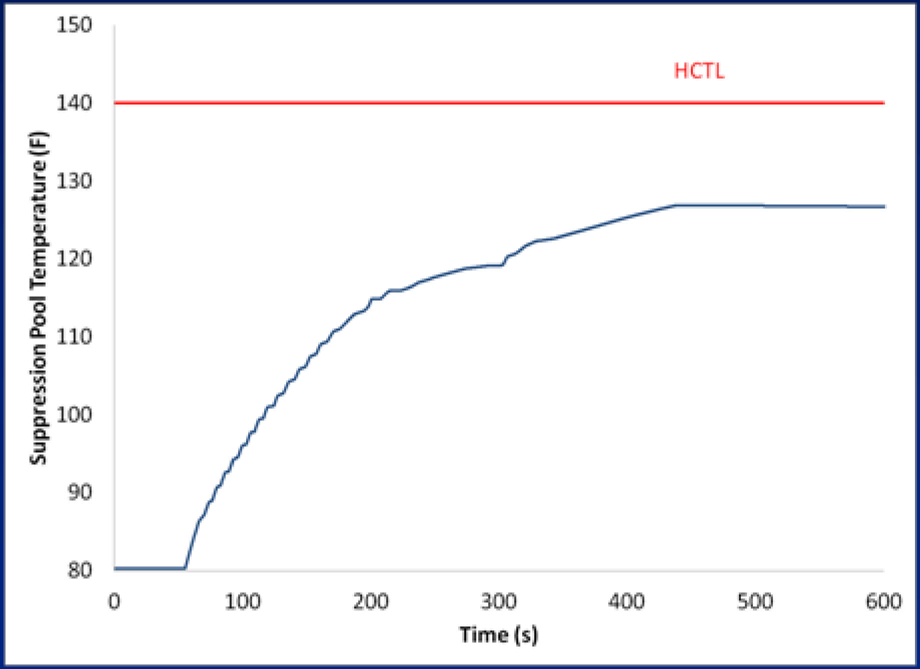
- Staff reviewed the performance of the OPRM Solution III system in the simulator
- Staff reviewed ATWS performance in the simulator (3 different scenarios)
  - Turbine Trip ATWS From The MELLLA Corner with simulated unstable oscillations
  - MSIV Isolation ATWS from MELLLA corner
  - MSIV Isolation ATWS from EPU conditions
- NMPNS submitted additional information with the simulator ATWS results

Simulator shows similar response at EPU and CLTP

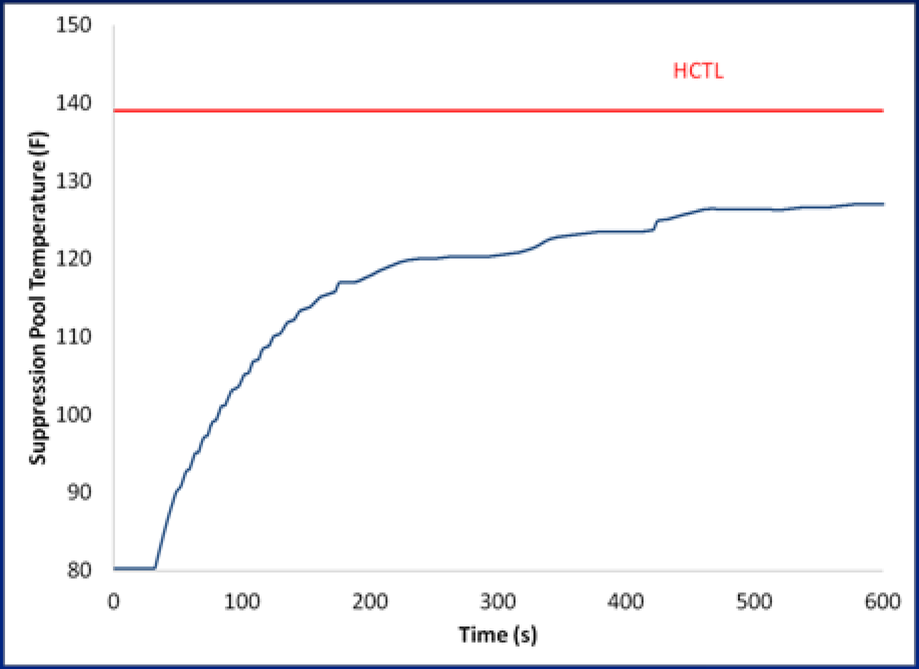


Simulator shows margin to emergency depressurization

CLTP



EPU



Summary

- EPU operation is acceptable from stability point of view
  - Installed LTS (Sol III) provides similar level of protection under EPU and CLTP
  - OPRM scram satisfies GDC 10 and 12
- ATWS and ATWS-Stability not affected significantly by EPU
  - Satisfies ATWS Acceptance Criteria (10CFR 50.62)
  - NMP2 has excellent ATWS performance design
    - Automatic trips
    - Upper plenum boron injection
    - 100% motor-driven FW pumps

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Nine Mile Point Unit No. 2 Extended Power Uprate ACRS Subcommittee Meeting

Interim Methods Applicability of GE Methods to Expanded Operating Domains

Dr. Peter Yarsky
RES/DSA/RSAB

Pani – can you please print out 2 colored copies of the attached and give them to me later today.
Thanks,
Rich

Methods Review Basis

- Review based on approved LTR NEDC-33173P “Applicability of GE Methods to Expanded Operating Domains,” (the IMLTR)
- Staff confirmed that the EPU LAR is fully consistent with the conditions and limitations specified in the staff’s SE for the IMLTR

Staff Review Items

- IMLTR: 24 Conditions and Limitations
  - No Supplements to the IMLTR referenced in the NMP2 EPU LAR
  - PUSAR Appendix A dispositions each condition and limitation
  - All 24 conditions and limitations acceptably met
  - Staff conducted one regulatory audit pertaining to the IMLTR

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Staff Review Items

- LPRM Calibration Interval
  - LPRM update affects core monitor accuracy to predict power distribution
  - Interval is 1,000 EFPH
  - Post EPU, exposure interval between calibrations would increase 15 percent
  - Staff audited GEH data to confirm that power distribution uncertainties were acceptable for longer exposure interval

Conclusions

- Methods application acceptable because all staff SE conditions and limitations on the IMLTR are met