ACRST-3434

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
551st Meeting - Open Session

RECEIVED
APR 15 2008

Docket Number: (n/a)

Location: Rockville, Maryland

[redacted]

Date: Thursday, April 10, 2008

Process Using ADAMS Template
ACRS/ACNW-005
SUNSI Review Complete

Work Order No.: NRC-2115

Pages 1-103

ORIGINAL

NEAL R. GROSS AND CO., INC.
Court Reporters and Transcribers
1323 Rhode Island Avenue, N.W.
Washington, D.C. 20005
(202) 234-4433

TR04

DISCLAIMER

UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

April 10, 2008

The contents of this transcript of the proceeding of the United States Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, taken on April 10, 2008, as reported herein, is a record of the discussions recorded at the meeting held on the above date.

This transcript has not been reviewed, corrected and edited and it may contain inaccuracies.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

+ + + + +

551TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

OPEN SESSION

+ + + + +

THURSDAY

APRIL 10, 2008

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

The Advisory Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., Dr. William J. Shack, Chairman, presiding.

COMMITTEE MEMBERS:

WILLIAM J. SHACK, Chairman

MARIO V. BONACA, Vice-Chair

SAID I. ABDEL-KHALIK, Member-at-Large

GEORGE E. APOSTOLAKIS, Member

J. SAM ARMIJO, Member

SANJOY BANERJEE, Member

DENNIS C. BLEY, Member

MICHAEL CORRADINI, Member

OTTO L. MAYNARD, Member

DANA A. POWERS, Member

JOHN D. SIEBER, Member

JOHN W. STETKAR, Member

GRAHAM B. WALLIS, Consultant

TABLE OF CONTENTS

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

8:31 A.M.

CHAIRMAN SHACK: The meeting will now come to order. This is the first day of the 551st meeting of the Advisory Committee on Reactor Safeguards. During today's meeting, the Committee will consider the following. Extended power uprate application for the Hope Creek Generating Station, proposed licensing strategy for the next Generation Nuclear Plant, the NGNP and preparation of ACRS reports.

The session on TWR Owners Group Topical Report WCAP-16793, Evaluation of Long Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluids scheduled to be held between 12:30 and 2:30 has been postponed to a future meeting at the request of the NRC staff.

A portion of this meeting related to the Hope Creek extended power uprate will be closed to protect information that is proprietary to General Electric, Hitachi and Continuing Dynamics, Incorporated. In addition, the session on the proposed licensing strategy for the next generation nuclear power plant will be completed closed to prevent disclosure of information, the premature disclosure of which would be likely to frustrate implementation of a proposed agency action. The meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Mr. Sam Duraiswamy is the designated federal official for the initial portion of the meeting.

We have received no written comments or requests for time to make oral statements from members of the public regarding today's session. A transcript of portions of the meeting is being kept. It is requested that speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so that they can be readily heard. We have representative of the State of New Jersey, Tennessee Valley Authority and NRC contractors on our bridge line listening to discussions related to Hope Creek extended power uprate.

To preclude interruption of the meeting, the phone line will be placed in a listen in mode during the presentations and the Committee discussion. I will begin with some items of current interest. I am happy to announce that Dr. Powers has received the Tommy Thompson Award for his outstanding contributions toward enhancing the safety of nuclear power plants and in particular towards an improved understanding of the phenomenology of severe accidents.

Congratulations, Dr. Powers.

(Applause)

Commissioner Kristine Svinicki has -- was sworn in on Friday, April 4th, 2008. She is filling the seat vacated by the former Commissioner Merrifield. Commissioner Svinicki's term will run until June 30, 2012. Ms. Sonary Chey, who has been with the ACRS staff for about five years is leaving on April 14th to join the Division of License Renewal in NRR. During her tenure on the ACRS staff, she has provided outstanding administrative support to the committee members and the staff in several areas, including preparing CD's for several ACRS full committee meetings and assisting in the preparation of PNT subcommittee meetings, agendas, meeting agendas and anticipated workload matrix. Her enthusiasm, dedication, professional attitude, hard work, attention to details and willingness to assist others are very much appreciated. Thank you and good luck in your new job.

Ms. Carol Brown, who has been with operation support for about two years is leaving on April 18th to join the staff of the University of Virginia in Charlottesville. During her tenure on the operation support staff she has provided outstanding support to the committee in several areas including processing travel vouchers for the members, issuing Federal Register notices for the ACRS meetings, finalizing summary reports and ACRS reports as well as providing administrative support in the preparation of ACRS reports during the meetings. Her professional attitude, dedication, hard work, attention to details, patience and willingness to assist others are very much appreciated. We thank her very much and wish her good luck in her new job.

I will also mention that my informulates (phonetic) and some of my colleagues have commented on my informal attire. It's not meant with any disrespect. I simply can't get an arm up to tie a necktie. So perhaps by next meeting, I will be back to --

Our first topic today will be the extended power uprate for the Hope Creek Generating Station and Said will be the member leading us through this.

DR. ABDEL-KHALIK: Thank you, Mr. Chairman. On March 20th and 21st of 2008, the ACRS Power Uprate Subcommittee heard presentations by and held discussions with the staff, the licensee and its contractors on a range of topics important to the safe operation of Hope Creek at EPU conditions. The subcommittee members had the opportunity to review the staff SER, the licensee's power uprate safety analysis report, staff requests for additional information and the specific topics presented at the meeting.

At the conclusion of the meeting, the general consensus of the subcommittee was that the Hope Creek EPU application is ready to be forwarded to the full committee for consideration at today's meeting. The subcommittee selected five topics to be highlighted in today's presentations. These are, probabilistic risk assessment, containment analysis, materials, fuel dependent analyses and methods and steam dryer and power ascension testing.

Of these topics the subcommittee views the discussion on the steam dryer and the power ascension testing to be most important inasmuch as the licensee does not intend to replace, modify or directly instrument the steam dryer prior to or after granting of this license amendment. Instead, the licensee will rely on strain measurements on the main steam lines along with an analytical model to infer the loading on the steam dryer and hence calculate the state of stress at EPU conditions.

Since this is the last topic on the agenda, it is my hope that the discussions on the other four topics would proceed in a timely fashion. This would allow the Committee sufficient time to hear from the staff and the licensee on the acceptability of the steam dryer integrity and analysis methodology at the proposed EPU condition.

We have received a request for a teleconference from several individuals including the representative of the State of New Jersey, Mr. Jerry Humphries. There are several bridge numbers and passwords available, depending on whether the session is open or closed. Closed sessions will be announced by the designated federal official. Any caller who wishes to listen in on the closed session must have clearance from the licensee and/or the owner of the proprietary information. The correct bridge numbers were provided to participants in advance.

Attendees who are required to leave during the closed session can call 301-415-7365 to obtain a status report as to when they can rejoin the meeting. We will now proceed with the meeting and I call upon Mr. Tim McGinty of NRR to start the meeting.

MR. McGINTY: Good morning. I am Tim McGinty. I'm the Deputy Director for the Division of Operating Reactor Licensing in the Office of Nuclear Reactor Regulation. Consistent with Said's opening remarks, I'm going to keep mine brief to stay within the scheduled time. On behalf of NRR, I'd like to take the public opportunity to thank the ACRS for accommodating our schedule and reviewing the steam dryer portion on a short turnaround. The staff greatly appreciates the ACRS members' efforts in this regard.

I believe over the next three hours you'll hear the results of a very thorough US Nuclear Regulatory Commission staff review of the application submitted by Public Service Enterprise Group Nuclear Limited Liability Corporation or PSEG. Our purpose this morning is to convince you that the proposed Hope Creek extended power uprate provides reasonable assurance that the health and safety of the public will not be endangered. After three hours of hearing presentations from the staff and PSEG, we hope that you will agree and will recommend that the proposed Hope Creek EPU amendment be issued and reflect this is your letter report.

At this point, I'd like to turn over the discussion to my Senior Project Manager, John G. Lamb who will introduce the discussions.

MR. LAMB: Good morning. My name is John Lamb. I am the Senior Project Manager assigned to the

Hope Creek extended power uprate EPU. Before I give you this morning's agenda, I'm going to go over a brief overview of the background of the Hope Creek. I will quickly present that background information.

Hope Creek is located in the Lower Alloways Creek Township, Salem County of the State of New Jersey, which is approximately 70 miles southeast of Trenton, New Jersey. Hope Creek is a boiling water reactor that's a BWR4 and it has a Mark 1 containment. On July 25th, 1986, the NRC licensed Hope Creek for full power operation at 3,293 megawatts thermal. Hope Creek was granted a measurement uncertainty recapture, MUR, power uprate of 1.4 percent in Amendment Number 131 dated July 30th, 2001.

The MUR changes were based on the installation of a CE Nuclear Power LLC cross flow ultrasonic flow measurement system and its ability to achieve increased accuracy in measuring feedwater flow. This MUR increased power from the original licensed thermal power level of 3,293 megawatts thermal to the current license power level of 3,339 megawatts thermal.

The ACRS did not review the MUR as --

DR. WALLIS: Is that a typo or something that 3293, is that -- no, I'm sorry, that's okay. Go ahead. I'm confused, not you.

MR. LAMB: Sure. The MUR increased the original license power from 3293 megawatts thermal to the current license power of 3,339 megawatts thermal. The ACRS did not review this MUR as is custom for MURs. The proposed EPU would increase the maximum authorized thermal power from the current power level of 3,339 megawatts thermal to 3,840 megawatts thermal. This represents an approximate 15 percent increase from the current license thermal power.

Now I'd like to briefly go over today's agenda topics. The ACRS Subcommittee requested presentations for this morning to concentrate on the following topics, materials, containment, probabilistic risk assessment, PRA, fuel methods and steam dryer. As you can see, we have a great deal to cover in a short period of time. PSEG will cover short presentations on containment and PRA and the staff will provide a short presentation on materials, containment and PRA. Then you will hear presentations from PSEG and the staff on fuel methods.

Finally, you will hear steam dryer presentations in open and closed sessions from PSEG and that staff. Now, I'd like to turn it over to Mr. Paul Davison, the PSEG Engineering Director at Hope

Creek Generating Station.

MR. DAVIDSON: This is Paul Davison, PSEG. The order of our presentation begins with PRA actually and so the first presenter would be Ed Burns.

MR. BURNS: Good morning. My name is Ed Burns. I am the Hope Creek Risk Management Team Technical Leader. I'm responsible for the Hope Creek PRA development, implementation and its application. Thank you for this opportunity to discuss the effects of the EPU implementation on the Hope Creek risk profile.

DR. ABDEL-KHALIK: We can go ahead. We have copies of your presentation until they work out the details.

MR. BURNS: Great. The first slide is a background summary. We were able to provide the Hope Creek EPU risk profile information during our ACRS Subcommittee presentation on March 21^{st}. We identified that the EPU submittal is based on a deterministic evaluation of licensing criteria --

DR. WALLIS: This isn't the picture of Hope Creek on here?

FEMALE PARTICIPANT: No, we were trying to swap it to this.

DR. WALLIS: Go ahead, we've got the

slides. We have it, go ahead.

MR. BURNS: We identified that the EPU submittal is based on a deterministic evaluation of licensing criteria and is not a risk informed submittal. Nevertheless, we've provided a risk perspective regarding the effect of EPU implementation. That presentation included a discussion of PRA scope and quality, quantitative results of the internal events, a qualitative assessment of external events and concluding that the risk change resulting from EPU implementation is very small.

The ACRS subcommittee requested additional detail regarding the disposition of individual fire and seismic accident sequences effected by EPU implementation, therefore, this presentation focuses on the subcommittee's request. The next slide summarizes the risk evaluation methods used for these analysis to meet the subcommittee's request.

We identified plant configuration and procedural changes due to EPU. We used updated PRA models for internal events consistent with the ASME PRA standard. We used available IPEEE fire and seismic PRA models updated to incorporate the internal event success criteria and we identified the PRA elements effected by the changes. Those changes are reflected in the PRA principally in the crew response characterization, the success criteria, initiating event frequencies and a number of challenges to systems.

We incorporated hardware and procedure changes in the PRA model. We used realistic success criteria limits from the revised PRA and finally, we compared the results with the Reg. Guide 1174 acceptance guideline. As requested by the subcommittee, the next slide identifies that the quantified fire risk evaluation uses available information recognizing that the quantitative results may be conservatively biased.

Resources available include a fire scoping analysis available from the IPEEE. This IPEEE model is not yet updated as part of the Hope Creek available PRA tools. In particular, there are conservatisms which bias the results of the fire scoping study and those include initiating event frequencies, the fire modeling, the fire suppression reliability assessment and the human error rates.

However, from a deterministic standpoint, the EPU evaluation recognizes that there is no increase in combustible loading and no new fire initiating events or increased fire frequency. However, there are some potential changes in the time available for crew response. Therefore, the quantitative fire PRA model calculations include the following; an examination of the critical fire scenarios from the IPEEE, searches for risk contributions that may be effected by EPU, including loss of equipment or access to equipment which lead to CDF directly, regardless of EPU implementation, and in addition, we determined that sequences are related to loss of decay heat removal scenarios where changes in HEPs are small or negligible because of the long times available for response and recovery.

The fire analysis results in changes to the risk profile due to EPU principally related to changes in the allowed operator action times. The next slide gives the fire quantitative results. The dominant EPU effect is related to reduced time available for manual actions. The quantification of the 16 fire core damage scenarios resulted in a change in CDF of $7E^{-8}$ due to this reduced time available for crew response. We also tried to bound the residual fire induced CDF, five percent of the total CDF, and we conservatively used the worst case effect of a decrease in allowable time for crew action and applied it to the full five percent of the residual fire CDF from the IPEEE and that led to a change in CDF of $3E^{-8}$ per year.

Therefore, the total risk change due to fire induced CDF is quantified at $1E^{-7}$ per year. The conclusion is that the EPU has a very small impact on the fire risk profile despite this conservatively biased fire scoping analysis.

DR. BLEY: Ed, how much was the timing reduced, operator response?

MR. BURNS: In the most restrictive case it was from 33 minutes to 27 minutes.

DR. STETKAR: Ed can I interrupt you just for a second. You focused on the operator response times. Did you change the -- the only other effect that I could see in the EPU was the change in success criteria for number of SRVs open for depressurization. They went from one out of 14 to two out of 14, I think. The only reason I ask about that is that a lot of the fire induced initiating events were loss of offsite power and MSIB closure which would tend to challenge those success criteria a little bit more.

MR. BURNS: Right.

DR. STETKAR: Did you change that in the IPEEE models or whatever you used for this requantification?

MR. BURNS: We actually inserted the 60 -- there are 60 fire initiating events, 6-0 as part of the 16 different compartments that were evaluated and we actually put those 60 initiating events into the latest PRA model --

DR. STETKAR: Oh, okay, great.

MR. BURNS: -- to fail the equipment that were identified.

DR. STETKAR: Great, thank you. Thank you.

MR. BURNS: For the seismic induced sequences, the qualitative seismic risk evaluation identified that there were -- there is a seismic PRA scoping model available from the IPEEE and that 89 percent of the contributors are hardware failures leading directly to core damage and no change in risk profile results from the EPU implementation. The EPU effects qualitatively assessed include the following; no change in seismic qualification for systems structures or confluence, no significant change in equipment mountings or anchorages, no new seismic vulnerabilities were identified.

The dominant contributors to the risk are related to equipment failures with no operator actions credited. The next slide summarizes the quantitative results for the bounding seismic calculation performed to support the subcommittee's request. The dominant contributors to the seismic induced risk spectrum of 89 percent of the seismic sequences result in CDF due to direct hardware failure where no operator response could be credited and therefore, no change in CDF results for these sequences.

One dominant sequence, Sequence SDS-26 includes credit for crew action and is treated below. The single dominant seismic sequence, SDS-26 and the residual contributors represent 10.4 percent of the seismic sequences and they may involve some crew failure actions that lead to core damage. So Sequence Number 3, SDS-26 at $2E^{-7}$ per year contributed a delta, a change in CDF of $1.4E^{-8}$ due to changes in the time available for manual actions.

We also conservatively assume that all other residual seismic sequences approximately five percent, have an impact associated with reduced time available for crew response assuming the worst case change in HEP observed in the internal events evaluation and that resulted in a change in CDF of 1.3 times $10E^{-8}$.

In the next slide is the aggregation of

the contributors by hazard.

DR. POWERS: I was a little confused by the last line of the slide.

MR. BURNS: Sorry.

DR. POWERS: What are these --

MR. BURNS: I'm sorry.

DR. POWERS: You've got a .051 times a .072.

MR. BURNS: The $3.6E^{-6}$ per year is the total CDF evaluated for seismic risk from the IPEEE. The .051 is the five percent of the residual. So I accounted for 95 percent of the seismic CDF in the other topics that we discussed, so this is the five percent that remains. And we determined that if we use the worst case operator action timing and assume that that applied to all of those residual five percent, that that would result in a change in CDF of .07, a conditional change of .07, so the product of the three leads to the absolute change of CDF of $1.3E^{-8}$. So I'm just looking at the residual -- there are a number of sequences that are in that lower five percent and we applied the worst case operator action chain, the effect of the worst case operator action chain on those five percent.

DR. POWERS: I was a little bit at a loss at what the .072 is.

MR. BURNS: The .072 is the conditional probability of a change in CDF associated with the operator action impact, a reduction in the reliability of the manual action for the actions that would be taken to successfully prevent a core damage.

DR. POWERS: So a 7.2 percent less reliable, that's 7.2?

MR. BURNS: Right, correct. So this slide is the aggregation of the contributors by hazard. Normally, it is not prudent to combine these contributors because they are based on significantly different bases, realistic for internal events and conservatively biased for fire and seismic. Nevertheless, even using these conservatively biased results from the IPEEE leads to a very small risk change compared to the Reg Guide 1174 acceptance guidelines, specifically at placed in Region 3.

Finally, in conclusion, the Hope Creek risk profile as effected by EPU implementation is appropriately characterized for first internal events consistent with the ASME PRA standard and secondly fire and seismic hazards using the IPEEE scoping study insights. The quantified risk impact is a small percentage of the current plant risk and the change in

CDF risk metric is a very small risk change per Reg Guide 1174 acceptance guidelines. Thank you for this opportunity to describe the risk perspective of EPU implementation at Hope Creek.

CHAIRMAN SHACK: Are there any questions for Mr. Burns?

DR. POWERS: Yeah, let me understand a little better. You made a 17 -- a 15 percent, I'll just count this 15 percent increase in the operating power. What's the percentage increase in the risk?

MR. BURNS: The percentage increase in risk as measured by CDF, seven percent.

DR. POWERS: It's seven percent.

MR. BURNS: Yes, sir.

DR. POWERS: Why isn't it exactly 15 percent?

MR. BURNS: Because the seven percent of the internal events, sorry, that doesn't count the external even analysis. It's because there are a large -- the dominant contributor to the risk is associated with the operator actions in very short time frames. However, the accident sequences that lead to core damage include both sequences that occur over very short periods of time and sequences that occur over very long periods of time.

So the sequences that occur over a longer period of time are not effected by the small changes in operator response, tiny, and therefore, those sequences, when integrated over that whole spectrum, result in lower than a 16 percent change. And it would be also wrong for me to infer that the change in operator action is directly propor -- operator action probability is directly proportional to the change in timing also.

DR. WALLIS: So since Dr. Kress isn't here, I should say that -- point out the risk is frequency times consequence and you have changed the frequency by seven percent. You've changed the consequence by 15 percent, so the change is risk is really something like 22 percent and it's kind of unfortunate that the agency uses the term "risk" to mean frequency and identically with risk which is not really the right way to look at this. Unfortunately CDF becomes called risk which is really frequency times consequence.

DR. CORRADINI: Is that the case, though? I'm not sure -- it's just directly multiplicative, isn't it?

DR. POWERS: No.

MR. BURNS: You're correct, we do not

calculate ex plant consequences as part of this analysis. We do calculate --

DR. WALLIS: Inventory of radioactive quantities is increased --

MR. BURNS: Yes, yes.

DR. WALLIS: -- by 15 percent.

MR. BURNS: Yes, correct, yes, absolutely.

DR. ABDEL-KHALIK: Thank you, Mr. Burns.

DR. WALLIS: I guess you really go into the detail of what effects the consequences are on the calculation and it become a very complicated one.

DR. POWERS: The question of what is small tends to come --

DR. ABDEL-KHALIK: Mr. Davison will now present the licensee's containment analysis.

MR. DAVISON: Thank you and good morning. I'm Paul Davison. I'm the Hope Creek Engineering Director. I'm also the sponsor for the EPU project at the site as well as a power ascension test director during power ascension. Slide 11 commences the presentation regarding the containment analysis that we performed at Hope Creek for EPU conditions.

We utilized the LAMB and M3CPT analysis codes to develop the short-term containment response which is dominated by the initial blow-down flow rate.

The results are influenced by the higher decay heat but are minimal due to the nature of the constant pressure power uprate approach. We then used SUPERHEX analysis codes to develop the long-term containment response. This was impacted due to the increased decay heat associated with the uprate.

All analysis were performed at or above 102 percent of the EPU power level of 38/40 megawatts thermal. The ANS/ANSI 5.1 methodology with two signal uncertainty was utilized to provide a more realistic containment response. This differs from our current Hope Creek UPSAR analysis which is based on the Maywitt (phonetic) decay heat methodology. The analysis did credit passive heat sinks including the drywell and torres metal shells and the containment vent piping.

Our submittal demonstrates that all the containment parameters remain below their respective design limits. In fact, on page 13, it shows these results. This table compares the containment analysis results for the analyzed parameters including the peak drywell pressure and temperature, the peak bulk suppression pool water temperature and the peak suppression pool air space pressure and temperatures.

When compared to the design limits,

positive margin is demonstrated. Therefore, the design basis accident LOCA containment performance results are well below any design limits.

Turning to slide 14, I'll cover the ESS pump net positive suction --

DR. ARMIJO: Before you leave that chart, now you're 218 degree design limit for the pool temperature, you changed that from -- it had been a lower number, maybe 212 or something.

MR. DAVISON: Correct.

DR. ARMIJO: Why is that justified? Exactly, you know, what did you do to get -- to justify changing the design limit?

MR. DAVISON: That number -- the design limit of 218 was actually picked arbitrarily to encompass both the worst case temperatures of the pool during LOCA and during the loss of power events. That was picked as a number that bounded and exceeded those and that's what we analyzed to.

DR. POWERS: I think what he's trying to understand is --

DR. ARMIJO: Why is that okay?

DR. POWERS: -- what creates that limit?

MR. DAVISON: I could invite Skip Denny up or Ted DelGaizo to comment on that.

MR. DelGAIZO: Yes, good morning, I'm Ted DelGaizo, Mainline Engineering and I'm a mechanical engineer on the EPU project. The -- there are several limits. I mean, there are limits on the piping and on the tora structure and components which are much higher than these two trials. They're in the 300s, 300, 310, numbers of that order. And so the controlling factor was really MPSH and also the pumps themselves. There were some pump seal issues. We had to go back to the vendor to -- because they were originally qualified at 212 and so to go to 218, we want back to the pump vendors to make sure the pumps were fine at 218. And then the key result was the MPSH calculation itself to show that we had sufficient MPSH at 218 and having done that, that becomes the new design limit for pool temperature, for bulk pool temperature.

DR. ARMIJO: So it's really, you'd have adequate performance of your pumps.

MR. DAVISON: Correct.

DR. ARMIJO: Even if the pool temperature was 218.

MR. DAVISON: Yes.

DR. ARMIJO: Okay, thank you.

MR. DAVISON: On Slide 14, I'll cover the

ECCS net positive suction head analysis assumptions and the conclusions that we have adequate pressure available without crediting containment over-pressure. Through shore bounding analysis conditions is utilized for determining the available net positive suction head. The assumptions for the analysis include the 218 degrees we just discussed for the suppression pool water temperature and 14.7 psia --

DR. WALLIS: It's a bit bizarre because water would boil at 218 degrees. It would be super-heated water, wouldn't it, at 14.7 psia.

MR. DAVISON: That is correct.

DR. WALLIS: So you're using a regulatory assumption which is physically unrealistic.

MR. DAVISON: Reg Guide 1.1, that is correct. Additionally, the torus water levels assume to be at the tech spec minimum level of 71 feel one-half inch. For the required positive suction head we also had the ECCS pumps assumed to be at the maximum tested flow rates.

Hope Creek has several design features that provide margins in net positive suction head requirements. We utilize horizontally mounted stack disc strainers that are located seven feet below the minimum tech spec allowed torus water level. The three and a half foot diameter strainers with the significant submergence when coupled with the low ECCS pump strainer approach velocities prevent vortexing from occurring. The ECCS pumps are vertically mounted, deep well, canned pumps located greater than 17 feet below the minimum tech spec allowed torus water level. The combination of these bounding assumptions and design features results in the containment analysis concluding that the available net positive suction head margin is conservatively determined to be 1.7 feet for the RHR pumps and 1.2 feet for the core spray pumps, therefore, adequate net positive suction head is provided without crediting containment over-pressure. This concludes my presentation. Any questions.

DR. ABDEL-KHALIK: Thank you, Mr. Davison. Are there questions for Mr. Davison? Thank you. We will now hear from the staff on three subjects, materials, PRA and containment analysis.

MR. LAMB: Okay, I am John Lamb. I have with me Matt Mitchell, the Branch Chief of the Vessels and Internals Integrity Branch of the Division of Component Integrity and NRR. The first subject we're going to cover is materials. The second subject is containment, which I have Rich Lobel here, that's a new engineer and then the last topic will be PRA which I have Donnie Harrison here for.

Okay, the ACRS subcommittee requested a very short presentation on materials based on the observation that Hope Creek is the only US facility to have a reactor pressure vessel constructed by Hitachi. Although the manufacturer of the Hope Creek reactor pressure vessel is unique within the US fleet, the materials of construction, for example, A508 forgings and A533 grade B plates, and the fabrication processes for example, shielded metal arc and sub-arc welding, used by Hitachi were consistent with those used to construct other US reactor pressure vessels.

As noted on the slide, the staff concluded that continued implementation of the boiling water reactor vessel and internals project integrated surveillance program would support Hope Creek's compliance with the requirements of 10 CFR Part 50, Appendix H, reactor vessel surveillance program requirements, the existing Hope Creek reactor vessel pressure temperature limits remain valid for the 32 effected full power years of operation and reactor vessel upper shelf energy analysis is acceptable with all reactor vessel beltline materials remaining above the 50-foot pound screening criteria of 10 CFR Part 50

Appendix G, Fracture Toughness Requirements. That concludes the material section unless there's any questions. I have with me Rich Lobel, the Senior Engineer in the Containment and Ventilation Branch of the Division of Safety Systems in NRR. Rich has 33 years of experience at the NRC.

NRC staff performed a thorough and complete containment analysis review in accordance with the review standard for extended power uprates. PSEG used NRC approved General Electric analysis and methods and I'm going to turn it over to Rich Lobel to explain his review.

MR. LOBEL: Good morning. The staff's review the containment systems portion of the EPU application was straightforward. The licensee performed the necessary analyses using approved methods and the results were within the acceptance criteria. The several new assumptions in the analysis were reasonable. They included crediting heat sinks in the calculations and crediting a jet deflector in the sump compartment calculations and a new decay heat model, new to Hope Creek.

CHAIRMAN SHACK: Isn't it conventional in these design basis things not to credit the passive heat structures?

MR. LOBEL: Not for the long term.

CHAIRMAN SHACK: Not for the long term.

MR. LOBEL: They've been credited in most of the BRW analyses. Hope Creek was the exception really in not doing it before now.

DR. CORRADINI: So just to understand, when you say long term, there is the blow-down phase and then everything after that is what you determine to be long-term?

MR. LOBEL: Right, the short term is for calculating the peak pressure and temperature --

DR. CORRADINI: Right.

MR. LOBEL: -- in the long term.

DR. CORRADINI: And that you cannot account for; is that right?

MR. LOBEL: Right, it doesn't play much of an effect there.

DR. CORRADINI: Right, but nonetheless it's not in there.

MR. LOBEL: Right, right.

DR. CORRADINI: And so this was -- I'm sorry, I didn't mean to interrupt you. I'm sorry.

MR. LOBEL: And the long term is the calculation of the suppression pool temperature.

DR. CORRADINI: And that, historically,

with other analyses have credited the heat sinks in some fashion based on some sort of accepted procedure

MR. LOBEL: Right.

DR. CORRADINI: Okay, thank you.

MR. LOBEL: Okay, the major changes effecting the containment due to extended power uprate are the increase in decay heat and a slight change in the reactor coolant sub-cooling that effects the mass and energy release from the vessel to the containment. No credit was taken for accident pressure and computing available NPSHs as you just heard. I'll come back to that. The staff requested additional information in several areas and the licensee's responses were clear, detailed and acceptable.

There's considerable margin between the dry well and wet well design pressures, as you've just seen. The licensee considered the effects, the extent of power uprate on the hydrodynamic loads as a result of the vessel blow-down and they were in acceptable limits. The licensee changed the method of calculating the mass and energy release into the containment. The new method has been used in other extended power uprates and is approved in the power uprate topical reports.

It consists of calculating the blow-down with the LAMB code rather than the M3CPT code. The licensee considered the effect on the EPU of hydrodynamic loads including pool swell, vent thrust, condensation, oscillations and chugging. All were within their respective limits. And because the reactor pressure remained unchanged, there was no change in the containment loads due to SRV discharge.

I -- what I have is really kind of a repeat of what the licensee just said on the MPSH. They aren't taking credit. They assumed a suppression pool temperature greater than what was calculated, about five degrees greater than what was calculated is the peak temperature. The LOCA is the -- in this case for Hope Creek is the peak suppression pool temperature event as opposed to other presentations we've given on other plants. The Appendix R atlas and station blackout have lower suppression pool temperatures for Hope Creek. And as was discussed also, the methods of calculating the head loss and debris source and that kind of thing are consistent with NRC approved methods.

DR. ABDEL-KHALIK: Thank you, Mr. Lobel.

MR. LAMB: Okay, next Donnie Harrison is going to talk about PRA.

MR. HARRISON: On this topic, I'm really

going to focus on changes that were made to the SECY evaluation of the staff based on the comments from the subcommittee. This first slide just tells you the intro that the Hope Creek application is not risk informed. We don't directly evaluate against the reg guide acceptance guidelines. We actually use the Standard Review Plan Appendix D, the SRP 19.2 for our guidance in the review standard, which is consistent with that guidance to determine if there is special circumstances that would make us question adequate protection at the plant.

The next slide. In addressing the subcommittee's comments, I believe Dr. Wallis, you pointed out that we had an error in -- there you are, an error in the -- I'm used to you over here -- in one place and you were correct. We actually made a change in the percentage to get that corrected. There is also some subcommittee comments on the fire and seismic approach that the staff use to estimate a quantitative CDF.

We went back, looked at the licensee's submittal, their RAI response to a question in this area and replaced that quantitative estimation by the staff with a qualitative observation that's based on that information that was docketed. Through that we rewrote those sections to eliminate the quantitative discussion and insert the qualitative discussion related to this.

We also made related changes through the rest of the chapter to reflect the changes that were made above in the quantitative sections.

DR. STETKAR: It sounds like you didn't have the benefit from the numerical information that we saw in the licensee's presentation.

MR. HARRISON: Correct. For this purpose the licensee, I believe, is, if you will, being a good citizen. They're bringing information to the ACRS that was discussed at the subcommittee. It was not submitted to the staff for being docketed in consideration for the license amendment, nor would it be necessary. Again, this is not a risk informed submittal. If it was risk informed, we would have pursued that information to support this application.

With those changes, I just want to note with the last thing on this page that with the revision to the SE, this revision is consistent with how the staff has conducted these reviews previously so there was a comment that if you will beg the question of, you know, were we doing these type of estimations before and the answer to that, on this particular area, is no, we were not. So this is consistent now. And just to conclude on this topic, the -- oh, I also want to point out one other thing. The information that was provided to you was revised and provided prior to actually getting concurrence from the management branch chief.

He's caught another typo that we inserted with our insert, so we're correcting and expedential, so win some lose some. So with that correction, you'll see a revised input but nothing really changes in our conclusion. The licensee has adequately modeled and addressed the risk impacts on the EPU. The EPU doesn't create the special circumstances. That's the overall conclusion of the staff. Are there any other questions on this?

MR. LAMB: Okay, to summarize the short presentations on materials, containment and risk, the staff concluded that the reactor pressure vessel meets the NRC regulations. All containment parameters remain below the design limits and the risks are acceptable because Regulatory Guide 1.174 Risk Acceptance Guidelines are met.

Now, we're going to turn our focus to the two areas where the majority of the ACRS subcommittee discussion time was spent, fuels and steam dryer. I'm now going to turn it over to Don Notigan, the PSEG Nuclear Fuel Manager to kick off the fuels presentations.

MR. NOTIGAN: Good morning. My name is Don Notigan. I am the Nuclear Fuels Manager at PSEG Nuclear. I have responsibility for design of the fuel, managing changes to core designs and the reload safety analysis for the Hope Creek Generating Station. Today I will be presenting the slides which cover the fuel methods and analyses done to support the safe operation of the fuel during the Hope Creek 115 extended power uprate.

Also I have with me Francis Safin, who is the Safety Analysis Engineer for EPU.

MR. SAFIN: Good morning.

MR. NOTIGAN: Our presentation will cover these four areas for the fuel response to EPU. I'll present the core loading map and fuel placement for EPU operation. I'll highlight the fuel performance and core design for EPU. I'll summarize the safety analysis results for Hope Creek's EPU and I'll present our conclusion statements about the fuel response to Hope Creeks extended power uprate.

This is the Hope Creek EPU core loading map. Cycle 15 has a combination of two fuel designs and both designs are the 10 by 10 lattice. Cycle 15 is the third consecutive reload of GE-14 fuel and there are some remaining co-resident fuel from Westinghouse identified as the SVEA 96 plus fuel. This slide illustrates the core loading and placement of those two fuel designs in the Hope Creek Core. The white color cells are the GE fuel, GE-14 and the blue color cells are the SVEA 96 plus fuel cells.

There are 548 GE fuel assemblies representing 72 percent of the core. And there are 218 of the SVEA fuel assemblies representing 28 percent of the core. An important observation of the EPU core loading map for Hope Creek CPU is that the SVEA fuel is primarily placed all around the periphery of the core and the remainder is placed in low bundle power locations in the core.

Points for EPU core design at Hope Creek, all the fuel assemblies in the core have PCI resistant design with barrier liner clad and all the fuel assemblies have integrated debris filters. The SVEA fuel has a low reactivity profile and is loaded in non-limiting core locations. SVEA fuel will operate with maximum bundle powers below pre-EPU levels.

Although 28 percent of the core is SVEA fuel, it delivers less than 20 percent of the EPU power. The GE-14 fuel delivers 81 percent of our EPU power. All EPU core design calculations and reload safety evaluations are completed.

Points for safety analysis, all thermal limits were met with margins remaining for both GE and SVEA fuel. The SVEA fuel does not contribute to setting the EPU core safety limit minimum critical power ratio. Key safety analysis parameters will remain consistent with those from the EPU reference plant operating experience base. All applicable limitations, conditions and adders from the NRC approved licensing topical report, NEDC-33173P were fully incorporated into our EPU safety analysis.

And lastly, our conclusions. We applied NRC approved GE nuclear analysis methods for Hope Creek's EPU. All EPU cycle specific core design calculations and reload evaluations are completed. The EPU results incorporated all applicable limitations, conditions and adders from the approved licensing topical report, NEDC-33173P. The SVEA fuel is non-limiting in EPU core operation.

Hope Creek fuel performance is consistent with EPU reference plant operating experience base for the key parameters important to safety. Based on these conclusion statements, safe operation of the fuel is confirmed for the Hope Creek 115 extended power uprate. This ends my presentation for fuels pending questions.

DR. MAYNARD: The GE nuclear analysis methods, are they applicable to non-GE fuels?

MR. NOTIGAN: The GE -- yes, the GE nuclear analysis methods were qualified for SVEA fuel and we supplied the results of an analysis the benchmarked the TGBLA and PANACEA nuclear analysis codes to the Westinghouse fuel to the staff for review and we qualified the limitations and adders as well for Hope Creek's EPU.

DR. SIEBER: You seem to have made a special effort to do power impact of the SVEA fuel compared to the General Electric Fuel but to my knowledge, there's nothing wrong with the SVEA fuel that would cause you to do that. That was just a tactical decision?

MR. NOTIGAN: Yes, sir.

DR. SIEBERT: Okay.

MR. NOTIGAN: If you'd like, I can share that trend of the bundle power for SVEA fuel if that's important.

DR. SIEBERT: Have you had any fuel failures in the SVEA fuel?

MR. NOTIGAN: Yes, Hope Creek has had some recent fuel failures with the SVEA fuel.

DR. SIEBERT: Could you describe what those failures were and why -- what you think caused them and --

MR. NOTIGAN: Yes, sir. On the previous cycle prior to EPU, there were three identified failures of the SVEA fuel.

DR. SIEBERT: Do we have this slide in our package?

MR. NOTIGAN: No, sir, this is a backup slide.

DR. SIEBERT: You'll provide us with --

MR. NOTIGAN: We will leave the backup slides at the end of our conclusion. (sic)

DR. SIEBERT: Thanks.

MR. NOTIGAN: This lists the fuel ID numbers of those three fuel assemblies SVEA design. One was identified as debris related. The other two were manufacturing related. In addition, we had previous cycles where we had in Cycle 11 and 12, three failures of the SVEA fuel as well. One was --

DR. SIEBERT: Go ahead.

MR. NOTIGAN: One was related to manufacturing and two were debris related. So in total six failures with the SVEA fuel.

DR. SIEBERT: Okay, when you say manufacturing related, could you describe the failure other than -- a little more extensively, please?

MR. NOTIGAN: Okay, with regard to the manufacturing related failures.

DR. SIEBERT: Right.

MR. NOTIGAN: It's really a process of elimination, going through what can cause a defect in the core. So after you've gone through operating history for PCI type related failures, you then go through a process of elimination of other likely causes. We've been able to eliminate all of the causes with the exception of manufacturing. Therefore, in manufacturing causal area it takes precedence for what's left. And you can look at things related to the pellet manufacturing or flaws on the cladding surface. That would be our manufacturing related causes.

DR. SIEBERT: Now, you've taken steps in the design of this core to minimize the peaks and valleys it appears to me, on other words, to flatten the core and you seem to have gone beyond the minimum level of doing that. Is that your ordinary design philosophy or is that just for the next couple of upcoming cores?

MR. NOTIGAN: I would say that the answer to that question is those are operating philosophy and design philosophy.

DR. SIEBERT: From a regulatory standpoint it makes no difference as long as you meet the criteria, but I was curious as to what your design philosophy was.

MR. NOTIGAN: Yes, our philosophy is to maintain design margins that we established at the beginning.

DR. SIEBERT: Okay.

VICE CHAIR BONACA: Did you shadow the defective rods?

MR. NOTIGAN: Could you repeat that, please?

VICE CHAIR BONACA: Did you shadow the defective rods for continued operation?

MR. NOTIGAN: Yes, sir. When we discover that we have a suspected location for a fuel defect, our procedures have us do power suppression testing to locate the cell that may contain the defects in the core. And after we determine which cell location is likely to have the defect, we then insert control rods to the full insertion point to depress and suppress the power in that cell and that shadows the defect.

VICE CHAIR BONACA: You must be upset to your burning of the core. It must be a significant impact.

MR. NOTIGAN: In a cycle where you have a suppressed rod, yes. It causes, you know, spacial differences across the core and asymmetrical type operation because of the inserted control rod.

DR. WALLIS: Well, failure is a dramatic word. I think it would be useful if you described for us or any public who might be listening or read the transcript what you mean by failure. It's not something like failure of the brakes on a car. It's a defect of some sort. Maybe you could explain what you mean by this. It's not as if it's a dramatic event. It's some sort of glitch or something.

MR. NOTIGAN: Yes, sir, I'll explain.

DR. WALLIS: Why don't you explain what it is?

MR. NOTIGAN: The use of the term failure just relates to the fact that there's been a perforation of the rod cladding that allows fission products to possibly escape into the cooling system. It does not mean to insinuate that there's a failure of the fuel or that there's any catastrophic type --

DR. WALLIS: So if you compared it with a tire on a car, I mean, a tire which blows out, is a failure but the sort of failures you're talking about is a pinhole that loses maybe the pressure over a month or something like that. It's a very tiny event compared with a failure in the normal sort of context that people talk about.

MR. NOTIGAN: Yes, in fact, in Cycle 14, one of the defects was so small it was almost undetectable.

DR. WALLIS: Thank you.

DR. SIEBERT: Do you also do that steady state and transient safety analysis, Appendix K type analysis in your group?

MR. NOTIGAN: No, our group, we perform the fuel designs and the core design management and then we perform the core follow for the cycle of interest for operating. We then do the designs and the fuel for the upcoming cycle. We have safety analysis which reviews and accepts and participates in the --

DR. SIEBERT: Make sure everything fits in.

MR. NOTIGAN: Independently reviews, yes.

DR. SIEBERT: Yeah, now, I noticed on

Slide 12, which is the containment analysis, the standard for the TKE curve looks to be different than the Appendix K curve. Is that correct?

MR. NOTIGAN: I'll have to --

DR. SIEBERT: ANSI 5481, 1979 I think the current Appendix K is an earlier version, is that correct?

MR. NOTIGAN: I'm going to have to ask Skip Denny of GE to address that question. Skip.

MR. DENNY: This is Skip Denny of GE Hitachi. Could you repeat the question first?

DR. SIEBERT: It seemed to me that here was -- a standard Appendix K TKE curve is different than the one shown on Slide 12 of applicant's --

MR. DENNY: Yes, sir, this is long-term TKE curve. It's the ANSI 5.1. The short-term analysis would use ANSI 5, 1972, I think it is, with 1020.

DR. SIEBERT: Yeah, everybody else would like to switch to this.

MR. DENNY: Right, but this is used just for the long-term analysis.

DR. SIEBERT: But it's legitimate to use two different TKE curves for the two different analyses.

MR. DENNY: Exactly.

DR. SIEBERT: Thank you.

MR. NOTIGAN: Thank you, Skip.

DR. ABDEL-KHALIK: Any other questions for Mr. Notigan? If not, thank you.

MR. NAKANISHI: Good morning, my name is Tony Nakanishi and I'm with reactor systems and NRR and I'll be discussing the fuel methodology review for Hope Creek EPU application. I did want to acknowledge Dr. Peter Yarsky for his contributions as well as part of this review and towards the end of the presentation, I'll also be summarizing the staff review of the safety analysis as well.

So the purpose of the staff review was to insure that the GE fuel methodology being applied for Hope Creek was applicable at the projected EPU operating conditions. The scope of the review was limited to topics that were included in the generic GE topical report, NEDC-33173P and I'll discuss that -- I'll summarize that topical report in a subsequent slide. And in addition, staff provided an additional review to insure that GE methods are applicable to the co-resident SVEA 96 fuel for this particular EPU site.

NEDC 33173P is a generic topical report by GE which addressed the EPU impact on the GE methodology and it was submitted to the staff to address the effects of EPU on M+ applications. Now, I did want to clarify that Hope Creek at this time is not applying for M+ just EPU.

If you recall, staff had an opportunity to come before the committee and discuss this topical report in detail and the committee concurred with the staff conclusions along with any limitations that were imposed. There was an ACRS letter that was submitted on June 22nd, basically concurring with the staff assessment of that topical report. And Hope Creek referenced 33173P as you saw in the licensee presentation and obviously, that referencing of this topical report influenced how the staff reviewed the Hope Creek application.

So the approach was to insure that the plant specific application process specified on the staff SE on 33173 were applicable and they included the topical report limitations as well as insuring that the key core parameters were within the operating experience. And as mentioned earlier, the staff provided additional review in terms of applicability to the co-residents of AFU.

So staff finds that Hope Creek complies with all applicable limitations and conditions, any compensatory measure as specified in 33173P and associated staff safety evaluation. In terms of applicability of the GE methods to SVEA 96, the bundles, SVEA 96 bundles are operating well within the EPU operating experience and as predominantly in the range of pre-EPU conditions.

As you saw in the licensee presentation, SVEA 96 bundles are loaded in a manner that would not be contributing to the limits.

DR. WALLIS: So they're within operating experience. When you analyze accidents, do they turn out to play any role?

MR. NAKANISHI: The cycle specific analysis would evaluate both GE 14 and SVEA fuel and in terms of the transient analysis that's done on a cycle specific basis and they are basically designed to insure that the operating limits are met for both fuels.

DR. SIEBERT: It was my impression that the reason why, if you look at the loading diagram, the SVEA fuel was loaded in low power locations was that, I just guessed this, General Electric didn't have all the details of the mechanical and nuclear design of the fuel and so as a precaution, they put the Westinghouse fuel in locations where the duty would not be high as opposed to saying, "I have done a specific rod-by-rod analysis of this fuel and the condition that it's in after it's been through a couple of cycles.

MR. NAKANISHI: Correct, GE rods have --

DR. SIEBERT: You just don't have the detail, right?

MR. NAKANISHI: Right, GE would have a lot more information of their own fuel and so there's -- this is a conservative approach that they're taking. I will add that the licensee provided additional information to insure that for this particular manner of operation, the GE's neutronics code sweep adequately models the co-resident fuel.

DR. SIEBERT: Yeah, I would just point out that it looked to me like there was a lot of forethought put into this by whoever come up with the core design, actually since you aren't driving the twice used fuel very hard, you're actually spending dollars for neutrons and to get this conservative design, the fuel cost may go up half a percent or something like that, not a noticeable amount but there is -- I thought the licensee, the applicant used good judgment to do that, just avoided a lot of problems.

MR. NAKANISHI: That provides the staff a lot of comfort in that respect.

DR. SIEBERT: Right.

DR. CORRADINI: So just to follow on so I understand, so on the next cycle this pattern clearly won't stay. It will rearrange which means that in the next cycle, the both steady state and the transanalysis will have to justify it to stay within the limits --

MR. NAKANISHI: Absolutely.

DR. CORRADINI: with another analysis.

DR. SIEBERT: Every refueling, there's a reload safety analysis that has to be performed that says that the reload that you're going to install, you know, twice the -- one time twice and the third time install fuel plus fresh reloads, has to meet the same envelope that is the maximum envelope for cores for that reactor. And so that's done for every reload, it has to be sent in and approved before you start up from that reload. So that's usually done a few months in advance of the actual refueling.

MR. NAKANISHI: So in conclusion, relative to the fuel methods review the staff finds that 33173P is appropriate for Hope Creek EPU and that's based on the finding that Hope Creek complies with all applicable topical report limitations, the methods are applicable to co-resident SVEA fuel for EPU cycle 15 and Hope Creek will be operating with the current experience base.

Finally, I did want to spend a couple of charts just summarizing the staff review of the safety analysis. And Mohamad Razzaque with Reactor Systems and additional support or additional team members provided this particular review but I'll just quickly summarize that the safety analysis was performed based on approved methodology in a manner consistent with staff approval and every event analysis showed acceptable results.

ASME over-pressure transient analysis are confirmed on a cycle specific basis as well as stability, LOCA. Also the PCTs aren't necessarily calculated every cycle but the MAPLHGR limits are confirmed to make sure that the analysis of record still remains. And ATWS was performed for this particular -- to address the impact of EPU and the results show that all licensing and regulatory limits are met.

DR. WALLIS: Is it fair to ask you what is the effect on the sort of margin to some of these limits? Is there any reduction in the margin as a result of the EPU?

MR. NAKANISHI: With respect to, you know, some of these cycle specific analysis, like transients and things like that, typically, you make sure your operation or operational limits are set such that you know, any impact --

DR. WALLIS: Do they get closer to the limits for the EPU or is the limit spread over more fuel or something? Is there any --

DR. SIEBERT: PRAs don't do that. It either fails or it doesn't.

MR. NAKANISHI: Right. Well, I guess the answer would depend on the particular analysis. I would think from my experience that things like over-pressure analysis would tend to get more limiting with EPU, although obviously, they'll still continue to meet any licensing and regulatory requirements. ATWS may be another area that may challenge a little harder but still remain within the regulatory limits.

DR. WALLIS: Yeah, but it doesn't say how much you're approaching the limits compared with what happened before.

MR. NAKANISHI: Right, and we could provide that information if you're interested.

DR. WALLIS: I just wondered if you knew it.

DR. MAYNARD: Any time you increase power, you are using some margin but it's not all within the fuel. You will spread it across more fuel assemblies. You will also take it away from operating limitations or operating limits and stuff. So it gets shared in a number of other places.

DR. WALLIS: I think with the power uprate it does get shared more than --

DR. SIEBERT: You could actually have a lower risk with a higher power core if the fuel is managed properly and what it does change is the consequence which is not a part of 1.174. You just have to have a high source term in some assemblies.

MR. NAKANISHI: So in summary, the staff found that the safety analyses were applied based on NRC's methods, analytical methods and codes. The scope of analysis is consistent with NRC accepted approaches and the results of the analyses show that the EPU impact on Hope Creek Safety analysis is acceptable. That concludes my presentation.

DR. ABDEL-KHALIK: Are there any other questions? Are there any questions for Mr. Nakanishi?

DR. SIEBERT: Thank you very much.

DR. ABDEL-KHALIK: Okay, at this time, Mr. Chairman, I'd recommend that we take a 15-minute break and when we come back we'll resume with the presentations on the steam dryer and power ascension testing.

CHAIRMAN SHACK: Okay, we'll come back then at 10:00 o'clock.

(Whereupon, the proceedings in the foregoing matter went off the record at 9:49 a.m. and went back on the record at 10:01 a.m.)

CHAIRMAN SHACK: We can come back in session.

MEMBER ABDEL-KHALIK: At this time, I'd like to call on Mr. Davison of the licensee staff to begin the presentation on the steam dryer and power ascension testing.

MR. DAVISON: Thank you, and good morning again. As stated, my name is Paul Davison. This open session discussion will provide an overview of Hope Creek steam dryer and the power ascension test plan.

The presentation itself is divided into five sections -- the design of the Hope Creek steam dryer, the design of our main steam piping system and its resultant low acoustic signature, the acoustic circuit modeling performed to develop the loads on the dryer, the dryer structural analysis results, and the power ascension test plan that will be implemented to confirm considerable margin identified in the steam dryer analysis.

On page 23, Hope Creek's steam dryer design is manufactured to the ASTM materials standards, the ASME welding standard, and General Electric's criteria to ensure structural integrity. Hope Creek's curbed hood dryer is the third generation of steam dryers designed by General Electric. This is an improvement to the square hood design used initially at Quad Cities.

This curbed hood design creates less turbulent steam flow through the dryer and into the main steam lines, which reduces the dryer operating stresses and reduces moisture carryover. Additionally, the dryer design was enhanced prior to its initial operational use.

General Electric approved modifications were implemented to address industry operating experience. They include the outer hood material thickness was increased from 1/8 to 1/2 inch. The center outlet plenum material thickness was increased from 3/16 to 1/2 inch. And the tie bar material thickness was increased from 1/2 by one inch to two by two inch cross section. Additionally, we increased the number of tie bars from 23 to 37.

The middle and inner hood to end plate joints were reinforced with external strips and internal backing welds. And the dryer support logs that are actually located on the internal diameter of the reactor vessel were leveled to prevent dryer rocking. No other modifications or repairs have been made to the dryer since startup, with the exception of the lifting rod bracket that we removed in our refuel outage number 12 due to mishandling.

Hope Creek's steam dryer original design and subsequent enhancements result in a very robust design for our EPU loading conditions. With respect to inspections, Hope Creek has implemented the requirements of BWR VIP 139. The baseline inspections were completed in refuel outage number 12 and 13 that ended in spring of 2006. No fatigue-related cracking was identified.

Next slide.

The background for Hope Creek's low acoustic signature or quietness is related to this diagram. Following the actual steam path, the steam dryer is positioned with its vein banks approximately perpendicular to the main steam line nozzles. The alpha and bravo main steam lines are shown to the right of the vessel and are mirror images of the charlie and delta main steam lines.

There are 14 target rock, two-stage safety relief valves with identical standpipe configurations. They are shown as the black dots. One blanked off standpipe for a spare SRV location is shown as a circle.

MEMBER ABDEL-KHALIK: Mr. Davison?

MR. DAVISON: Yes.

MEMBER ABDEL-KHALIK: What is the tech spec limit on the leak rate from the safety relief valves for Hope Creek?

MR. DAVISON: There is no specific leak rate tech spec. We do monitor the SRVs via acoustic and tailpipe temperature limitations. The focus of that is to ensure that the tailpipes are not leaking to add heat to the suppression pool.

MEMBER ABDEL-KHALIK: So there is no limit per se on the when you have to stop if the leak rate is excessive.

MR. DAVISON: I'll invite operations Bill Kopchick to respond to that.

MR. KOPCHICK: Yes, sir. My name is Bill Kopchick. I'm the Operations Superintendent in the Hope Creek Operations Department. Historic leakage from the Hope Creek SRVs has been minimal. However, we have encountered leakage from the SRVs to a small extent over time.

The way operators would monitor that would be with tailpipe temperatures. An analysis was done for each safety relief valve in its piping configuration by Engineering and placed into operating procedures, which are executed twice each shift. Specific tailpipe temperatures provide detailed guidance to operators as limits to initiate notifications to plant management that we would then activate our outage control center to evaluate SRV leakage at that time.

And the numbers vary from each -- for each SRV, whether it's somewhere between 280 to close to 300 degrees, which would key us off to then notify plant management we would have excessive leakage before we would actually have lift.

MR. DAVISON: And the ranges that temperatures that Bill talked about are specific to the configuration of the temperature elements and where they're mounted with respect to the actual exit of the actual SRV. But tech spec wise, there is no specific tech spec limit associated with tailpipe leakage.

MEMBER ABDEL-KHALIK: If an SRV were to

leak, that would effect the steam line velocity going past the safety relief valve, both upstream and downstream of that particular SRV. Is that correct?

MR. DAVISON: By an extremely minute amount, yes, that's correct.

MEMBER ABDEL-KHALIK: Okay. Perhaps we'll just wait until the closed session to discuss the possible impact of that.

MR. DAVISON: Okay.

MEMBER ABDEL-KHALIK: Thank you.

VICE CHAIRMAN BONACA: I had a question regarding the SRVs that the Subcommittee -- if there was any experience from the reactor's EPU with -- of SRV performance. And you told me that you would gather that information, if possible.

MR. DAVISON: Is the question, has there been EPU plant experience related to increased through-seat leakage?

VICE CHAIRMAN BONACA: Through SRVs, yes.

MR. DAVISON: We do not have specifics of -- or quantitative data from the industry with respect to increases in tailpipe leakage. The response that we provided at the Subcommittee had to do with the actual setpoint drift part of the issues that have been out in the industry. We don't have any quantitative data on tailpipe leakage for EPU plants.

MEMBER SIEBER: The square standpipe that you have for the SRV that is not installed, I presume that blanked off where the valve would have attached had you had one.

MR. DAVISON: That's correct. The standpipe itself is identical.

MEMBER SIEBER: So that represents an additional acoustic source?

MR. DAVISON: That's correct.

MEMBER SIEBER: Okay. You've taken that into account?

MR. DAVISON: Yes.

MEMBER SIEBER: Okay.

MEMBER ABDEL-KHALIK: Please continue.

MR. DAVISON: Hope Creek does not have any main steam line branch dead legs for SRV connection points. For comparison purposes, the Susquehanna branch dead leg locations on the alpha and delta main steam lines are shown in red.

Susquehanna experienced significant acoustic resonance attributed to these main steam line branch dead legs. Hope Creek's lack of main steam line branch dead legs precludes similar low frequency acoustic resonance.

After the main stop valves, which are just beyond the outboard main steam isolation valves, or MSIVs, the main steam line diameter increases from 26 to 28 inches. This is a beneficial feature that reduces flow-induced vibration.

The small --

MEMBER SIEBER: What is the flow velocity in the 26-inch segment? Do you know?

MR. DAVISON: The question is -- perhaps Dr. Bilanin can help me with the steam velocity in the 26- versus the 28-inch steam line itself.

DR. BILANIN: Alan Bilanin. I believe it's 165/167 feet per second.

MEMBER SIEBER: Okay.

MR. DAVISON: And that's at -- that will be at the -- that's the EPU flow rate of 167 feet per second.

MEMBER SIEBER: So it's lower than some plants.

MR. DAVISON: That is correct. Other plants -- Vermont Yankee was similar at 168, Quad Cities at 202, and Susquehanna is actually lower at 153.

MEMBER SIEBER: Okay.

MR. DAVISON: Okay. The smaller picture

provides a reference for the main steam line strain gauges located at the upper and lower drywell elevations on the main steam lines. Each location has eight strain gauges located at 45-degree intervals around the main steam line outside diameter.

In relation to other plants, Hope Creek has comparable main steam line flow velocities to Vermont Yankee and Susquehanna, as I mentioned, and our velocity is significantly lower than actual Quad Cities.

Quad Cities experienced significant acoustic resonance attributed to the electromatic relief valve standpipes. At CLTP, Hope Creek does not experience any acoustic resonance. This is due to our larger standpipe diameters and lower main steam line velocity.

Hope Creek's predicted SRV standpipe resonance at EPU conditions is expected to be lower than what Quad Cities experienced at their original licensed power. Hope Creek is expected to just transition to the onset of acoustic resonance at EPU.

Overall, Hope Creek's curved hood modified dryer, in conjunction with the lower main steam line velocities and absence of main steam line branch dead legs results in no main steam line acoustic resonance at CLTP.

In summary, we have a quiet plant with respect to acoustic resonance.

Next slide.

Hope Creek utilized Continuum Dynamics, Incorporated to perform the steam dryer analysis. This included Revision 4 of the acoustic circuit model for the dryer load definition and finite element analysis for actual dryer stress.

This slide covers the ACM, or acoustic circuit model, which was utilized to determine the pressure-induced loading on the steam dryer due to steam flow. The Committee has previously reviewed the CDI acoustic circuit model Revision 2, which was successfully used at Vermont Yankee. Rev 4 is the same model.

The incorporation of an additional source to improve low frequency load predictions in the zero to 60 Hertz range. The 60 to 200 Hertz range portion of the model remains unchanged.

The acoustic circuit model utilizes the main steam line strain gauges to predict dryer loads. The ACM uses the sensors on the main steam lines to obtain the necessary pressure time histories by measuring the hoop stresses.

The ACM provides the mathematical means to convert the pressure loads in the main steam lines back to the drawing itself. The ACM was validated by using the Quad Cities instrumented dryer data to compare actual dryer loading with predicted loading from the acoustic circuit model itself. The comparison also enabled the biases and uncertainties to be developed.

Next, CDI benchmarked Quad Cities' data at Hope Creek's specific EPU main steam line flow Mach number, and a second blind comparison was performed at a higher main steam line flow Mach number. Both benchmarks demonstrated predictable results.

MEMBER ABDEL-KHALIK: Now, but the results for the higher velocity corresponding to EPU conditions, this is sort of a completely blind calculation inasmuch as you don't have any steam line data for --

MR. DAVISON: That is correct.

MEMBER ABDEL-KHALIK: -- that case. And, therefore, the loading is based on a ratio between the loading at that current licensed thermal power and what you would expect at the extended power uprate condition based on the results of a scale test.

Now, in -- so the scale test, I assume, is true to the geometry of your plan.

MR. DAVISON: Yes. But I just wanted to go back and make sure I was clear in my discussion about Quad Cities. Quad Cities has an instrumented dryer, so we were able to actually measure actual loads on the dryer itself.

The CDI model that was developed using the strain gauge data on the main steam lines was then used to look at specific points to say based on what the actual flow is in the Quad Cities steam lines, and the measured loading on the actual dryer, did the model predict accurately?

MEMBER ABDEL-KHALIK: Right.

MR. DAVISON: We did that, one, to develop the model. CDI then utilized our specific Mach number as another specific data comparison point, and then a third just randomly picked higher Mach number above our Mach number but within, obviously, the operating parameters of Quad Cities, and then utilized the model again to ensure that it was able to predict the right loading on the dryer.

MEMBER ABDEL-KHALIK: Right. I fully understand the process.

MR. DAVISON: Okay.

MEMBER ABDEL-KHALIK: The question is: as part of this process, you needed a small-scale test to provide you with a ratio in the loading between the EPU conditions and the current licensed thermal power, how that ratio varies with frequency.

And that is obviously dependent on the geometry. The question is: before building that scale model, have you walked down and verified the as-built drawings of your steam lines?

MR. DAVISON: The scale model testing that we performed was validated to be similar to what is actually installed in the plant. However, I would like to invite Dr. Alan Bilanin to talk about specifically what the scale model test results were and were not used for.

MEMBER ABDEL-KHALIK: At this time, I'm just concerned about the geometry of the scale model test.

DR. BILANIN: He asked the question: do you have accurate as-built drawings of the main steam lines that were provided to Continuum Dynamics? And we believe the drawings that were provided to us, in fact, are very accurate in terms of as-built.

And then, the scale model that was developed was built approximately simulating the as-built configuration. The only differences were on some diameters, because commercial available piping was used, that the scale isn't exactly 1/8 scale. So the model that we used approximated 1/8 scale for the diameters of the piping.

The actual inlets, standpipes, and valves, were literally built to a thousand -- a thousandth of an inch at each scale to the actual as-builts as we understand them.

MEMBER ABDEL-KHALIK: So, but again the question remains: have the as-built drawings been verified ahead of time before being supplied to whoever built the 1/8 scale model?

MR. DAVISON: I believe they were. I'd like to verify that.

MEMBER ABDEL-KHALIK: Thank you.

MR. DAVISON: But I think it's important to talk -- the scale model testing, what it was and wasn't used for for development of the acoustic circuit model. We did not use the scale model testing for your acoustic circuit model, correct, Alan?

DR. BILANIN: That is a correct statement. In general, one could go from a CLTP load to an EPU load, if no acoustic resonance is anticipated, by simply scaling each frequency by velocity squared. Okay? That would be a standard technique, and other people are doing that as well.

So the scaling that you talked about to go from CLTP to EPU was essentially velocity squared for all frequencies, except at approximately 110 Hertz where the resonance is anticipated.

MEMBER ABDEL-KHALIK: Right.

DR. BILANIN: Okay? The 1/8 scale test confirmed the velocity squared scaling at all frequencies except at 110, and then the 1/8 scale test came up with a bump up factor that was larger at that frequency range.

MR. DAVISON: And, Alan, that was 118 Hertz, correct?

DR. BILANIN: 110 to -- 110 to 120.

MEMBER ABDEL-KHALIK: Now, the question is really, in that scale model test you predict a resonance frequency for the safety relief valve standpipes at about 118 Hertz or so. But you predict that to happen at power levels less than the current licensed thermal power. Is that correct?

DR. BILANIN: That's correct. And, therefore, I mean --

MEMBER ABDEL-KHALIK: Based on everything we have heard, none of that had been observed at the current licensed thermal power.

DR. BILANIN: That's a correct statement.

MEMBER ABDEL-KHALIK: No indications of resonances of the safety relief valves have heretofore been observed.

DR. BILANIN: That's correct.

MEMBER ABDEL-KHALIK: And the question is: well, how good is this 1/8 scale model test if it predicts something that has not been observed in the past?

DR. BILANIN: That's a good question, and it's actually an excellent question. The answer is very good. What you do is you set the Mach number on the inlet to the main steam lines to be CLTP, and then, because in fact subscale testing has additional friction. You can't match friction perfectly between a subscale test and a full-scale test.

The actual increase of Mach number as the flow goes down the steam line increases because of frictional effects, so that the subscale tests are biased to have a higher Mach number at the inlets to the main steam lines, and that is why you set the subscale test up to give you conservative results.

MEMBER ABDEL-KHALIK: Nevertheless, you are using the results of the subscale tests to give you these low bump up factors at different frequencies. Granted, they are all proportional to the velocity squared for most of the frequency ranges, except near the anticipated resonance frequency of the --

DR. BILANIN: Where the bump up factors are conservative from the 1/8 scale test.

MEMBER ABDEL-KHALIK: I understand. But the question is, you know, how reliable are these numbers?

DR. BILANIN: They are conservative.

MEMBER ABDEL-KHALIK: Based on what?

DR. BILANIN: The Mach numbers that are set in the 1/8 scale test are set such that, in fact, at CLTP we see resonance. In the plant, you don't see resonance. And then, when you run the test at EPU conditions and take that ratio, the bump up factor is a larger bump up factor.

MEMBER ABDEL-KHALIK: It's larger because you have a resonance in that frequency range. But really, I mean, you're inferring that you will have a resonance somewhere along the way at a different steam line velocity, i.e. different power level, than what you had gotten from the 1/8 scale test.

And the question is: well, how can you infer that the strength of that resonance will be the same as what you had predicted from those 1/8 scale tests?

DR. BILANIN: Because we'll maintain again that we set the 1/8 scale Mach number at EPU conditions and CLTP conditions to be higher than actual -- than actual in the plant. So we biased the 1/8 scale to be conservative.

MEMBER ABDEL-KHALIK: Again, really, the question remains as to, number one, why does this 1/8 scale test assembly or test model predict something that has not been observed, and whether that sort of brings to question anything that you extract from the results of that test.

MR. DAVISON: Well, I think specifically because of the questioning that you're posing, as well as the pretty intense dialogue that we had with the staff, that the scale model testing was ultimately not utilized for our submittal.

The specific of the monitoring program that we'll have in place to validate and verify that the acoustic circuit model is accurately reflecting what the plant is doing, and having specific limit curves that will drive us to stop the power ascension if we exceed a limit curve to reevaluate, is what specifically is built in not only to the licensing conditions but our test plan.

MEMBER ABDEL-KHALIK: But those limit curves are based on the load bump up factors that were extracted presumably from the results of the 1/8 scale model.

MR. DAVISON: In that specific frequency range.

MEMBER ABDEL-KHALIK: Correct.

MR. DAVISON: Yes.

MEMBER ABDEL-KHALIK: Correct.

DR. BILANIN: But if in fact the strain gauge data remains below the limit curve, then in fact the stresses are in fact acceptable. If in fact during power ascension the limit curves are in fact exceeded, a new stress analysis will be performed.

MEMBER ABDEL-KHALIK: Okay.

DR. BILANIN: Okay? Again, the 1/8 scale test was used to give an indication of what the stresses will be at EPU conditions. During power ascension, the actual stress levels, if the limit curves are exceeded, will be checked by analysis.

MEMBER ABDEL-KHALIK: Okay. Thank you.

MR. DAVISON: Turning to slide 26, for the steam dryer stress analysis, the finite element analysis model was developed by CDI using the ANSYS

Version 10.0 code. All CDI activities related to the steam dryer finite element analysis were performed under their quality assurance program, which is consistent with the requirements in 10 CFR 50 Appendix B.

CDI's finite element analysis, harmonic domain methodology, was used to -- which results in more accurate stress predictions by enforcing one percent structural dampening across the entire frequency range.

The pressure field developed by the ACM is then applied to this finite element structural model. The stress response over the zero to 200 Hertz frequency range is calculated by the fast Fourier transform, the pressure histories from the main steam lines themselves.

CDI's modeling capability was validated by comparing model predicted results against an independently conducted shaker test on our abandoned Unit II steam dryer. Additionally, the mesh convergence study confirmed that the mesh size utilized by CDI results in minimal errors. and, finally, the analysis was confirmed by audits and independent third party reviews.

The results of the steam dryer analysis

performed at 115 percent power show that the lowest predicted alternating stress ratio is 2.18. All the biases and uncertainties --

DR. WALLIS: Explain to the Committee what you mean by "lowest alternating stress ratio."

MR. DAVISON: All the nodes of the dryer themselves were looked at. The lowest alternating stress ratio -- stress ratio being defined as allowable divided by the actual stress ratio.

DR. WALLIS: Thank you.

MR. DAVISON: 2.18 was the lowest number. I'm sorry?

DR. WALLIS: That's what I'm looking for -- a definition of alternating stress ratio.

MR. DAVISON: Allowable stress divided by actual stress.

In summary, Hope Creek is an acoustically quiet plant. ACM Rev 4 improved the low frequency loading prediction. The biases and uncertainties were accounted for in both the ACM and FEA, and we benchmarked against Quad Cities' instrumented dryer data, and the alternating stress ratio at EPU remains above two, providing significant margin against fatigue-related cracking.

MEMBER SIEBER: Did anybody ever calculate the alternating stress ratio for Quad Cities? Do you know what it is?

MR. DAVISON: I'll invite Dr. Alan Bilanin up to talk about the Quad Cities alternating stress ratio.

DR. BILANIN: It's Alan Bilanin. I'm afraid I don't have that information. That analysis was done by General Electric.

MEMBER SIEBER: Okay. Thanks.

MR. DAVISON: All right. The specific power test plan for a steam dryer is governed by our licensing conditions, as I previously mentioned. A controlled and well-monitored power ascension will be executed to confirm the considerable margins identified by the steam dryer analysis.

The acceptance criteria limits for strain gauge and accelerometer testing are categorized into two levels. Level 2 is 80 percent of the parameter's allowable limit. Exceeding a Level 2 limit would require a power ascension hold and subsequent reanalysis to -- prior to resuming power ascension.

A Level 1 is 100 percent of the parameter's allowable limit. Exceeding the Level 1 would require us to reduce power to the previously acceptable power level and then do the reanalysis.

For the dryer, three specific data sets are analyzed. First, the main steam line strain gauge data will be compared to our preestablished limit curves to validate Level 1 or Level 2 acceptance criteria as not being exceeded. We'll actually show an example of one in the closed session.

MEMBER ABDEL-KHALIK: Now, there is only one set of limit curves. Is that right?

MR. DAVISON: There is a limit curve established for each location on each main steam line, so --

MEMBER ABDEL-KHALIK: Right.

MR. DAVISON: -- alpha upper/lower, and the same for bravo, charlie, and delta.

MEMBER ABDEL-KHALIK: But, nevertheless, it's one set corresponding to the highest power level.

MR. DAVISON: Correct. It is -- well, what we'll show you is what we submitted to the staff. We will be resubmitting prior to power ascension based on the staff's 2.1 stress ratio, so we will resubmit them. But the example that I will show you will have the specifics of what the baseline is and what the Level 1 and Level 2 at each frequency node is.

Secondly, the main steam line accelerometers will be compared to our preestablished levels for Level 1 and Level 2 values to ensure that the vibration data is within acceptable limits. This also serves as an independent check of the main steam line strain gauge trending data. Data will also be analyzed to every one percent power.

And, finally, the steam dryer moisture carryover will be monitored as a secondary means to detect changes that would be indicative of a dryer failure. Moisture carryover is checked via the sodium 24 isotopic comparison of condensate versus reactor water cleanup samples, and that's done every 2.5 percent power.

MEMBER ABDEL-KHALIK: Would the trending include trending of these load bump up factors at different frequencies?

MR. DAVISON: Yes.

MEMBER ABDEL-KHALIK: Okay.

MEMBER MAYNARD: I thought you were also going to be monitoring or watching the water level, not necessarily the water level itself but the inputs to it there, because that was seen at Quad Cities as being --

MR. DAVISON: That's correct. My next piece is the fact that we will be watching reactor water level, and we'll be monitoring the instrument channel for divergences, as well as oscillations, because --

DR. WALLIS: Well, I think you explained to the Subcommittee that it's not the level that oscillates, it's that the pressure fluctuations are transmitted to the transducers which measure level.

MR. DAVISON: Yes. It looks like level oscillations, minor oscillations, and because we have different channels that measure based on the TAPs from the vessel itself.

DR. WALLIS: This is really measuring just pressure fluctuation, because the level isn't bouncing up at the frequency you're talking about.

MR. DAVISON: Right. But that will be observed as the oscillations interact or --

DR. WALLIS: I will --

MR. DAVISON: Right.

DR. WALLIS: Right.

MEMBER MAYNARD: But the phenomena is not changing water level. Its pressure pulse is basically at the level that's provided.

MR. DAVISON: That's correct.

Power ascension will be performed at a rate of one percent per hour. The power ascension coordination center, or PAC as we call it, will be staffed around the clock. Any deviations from acceptable limits or adverse trends will be reported to the main control room immediately.

A dryer data evaluation will be performed every 2.5 percent power and reviewed by the power ascension team. A dryer evaluation will be performed and reviewed by our Plant Operating Review Committee, or PORC, and subsequently submitted to the NRC for review at each five percent power plateau, they being 105 percent, 110 percent, and 111.5 percent power.

Since we will not exceed the 111.5 percent power during this operating cycle, a final plateau and the NRC submittal will be performed at that point.

The next slide, which --

MEMBER MAYNARD: I apologize if you already said it, but what about monitoring in the field? Are you going to have people out walking around and observing, listening?

MR. DAVISON: In fact, I'll cover that right now on -- you do have a color slide handout that was provided. We tried to clarify with the colors to make it stand out a little bit more. In fact, that will show you some of the tests, and I'll talk to that.

MEMBER MAYNARD: Okay.

MR. DAVISON: This actual test matrix helps by providing an overview of all of the testing that's performed at each power level.

DR. WALLIS: What are the two entries that say 111.5? There's a black one and a red one. Presumably, the red one is the real one, is it, or what is --

MR. DAVISON: That's correct. You'll note that the red one at 111.5, as well as 115 percent power, have the initial CF, which is crossflow, applied. So we are currently, as mentioned in the staff's kickoff around an Appendix K plan, or recovered instrument uncertainty margin, we have an AMAG crossflow system that accurately -- more accurately measures the actual feedwater flow.

With our EPU, we are not including that measurement uncertainty recovery in our submittal. However, we will utilize the crossflow system to make sure that our feedwater is measured as accurately as possible, so the reason that that specifically says CF is we'll go to 111, as indicated in our control room, which will actually be 97 -- or 97 percent, after we rescale everything, and then we will wait the necessary duration to enable crossflow, apply that to get a more accurate feedwater flow, and then take the final set of data again at that slightly -- we believe slightly increased power level, because of the failing phenomenas that occur on feedwater flow ventures.

So that's all that's delineating is that we will stop at our first 111.5, we will apply crossflow, bring the plant to 111.5 with the necessary reactivity maneuvers, and then do the data set again.

Okay. The chart defines the testing and data collection categories across its -- across the top of the chart, and its associated power levels, where they will be executed down the first column. The four columns that are shaded identify the tests I've been specifically discussing associated with the steam dryer.

The remaining columns are the balance of testing to be performed to ensure adequate plant performance at EPU conditions. So in addition to just a data collection and analysis, we will also be doing many other things. We will be doing plant walkdowns with engineering and plant operations personnel, just to detect any physical audible type changes in the plant as we increase power.

There is also a significant number of tests from core performance, which is just more data collection, radiation surveys, and then digital EHC, which is reactor pressure step changes, feedwater step changes that will be going on -- not related to the dryer, but, still, that's kind of our whole battery of testing and monitoring that we'll be doing.

The red bolded rows are the power plateaus I spoke about that we'll be holding for NRC review. I talked about the correction factor, why you see that for 111 and 115. In addition, the testing that's delineated in this table will also be performing dryer inspections in our refueling outage during the spring of 2009. That will follow approximately nine months of operation at EPU power.

And although we'll stop at 111.5 percent power due to our high pressure turbine limitations, we did include the power ascension testing that would be recommenced in the future once those issues are resolved.

CHAIRMAN SHACK: You had some discussion with Said about the limit curves. And when you're doing the -- you're trending the main steam line gauge readings, are you comparing those with your predicted values at each of these steps?

MR. DAVISON: Yes.

CHAIRMAN SHACK: Or just the ultimate limit value that you're going to be able to tolerate?

MR. DAVISON: We'll be comparing them specifically -- first check will be just a straight check of, did it violate the Level 1 or Level 2 limits? The second piece will be, what is the actual trend? In other words, if we have a trend that would predict our next power change would put us at a Level 2, we wouldn't do that, because that would be an adverse trend.

So those are the -- we don't have specific limit curves drawn for 101, 102, you know, all the way up. So that's what we'll be doing the trending for.

And that actually concludes my presentation, pending questions.

MEMBER ABDEL-KHALIK: Are there any open session questions for Mr. Davison?

(No response.)

Okay. Thank you. We'll now move to the open session presentation by the staff on steam dryer and power ascension testing.

MR. MANOLY: Good morning. I'm Kamal Manoly, the Branch Chief of the Mechanical and Civil Engineering Branch in NRR. I would like to introduce the team that worked on the review of the Hope Creek power uprate review, particularly the dryer evaluation that you are interested in.

Tom Scarbrough is not available, but he was involved in the review. Dr. Carney is filling in for him, and he took over the lead responsibility. And we had contractors from Argonne, supported by Dr. Steve Hambrick and Dr. Ziada.

And I would like to just, based on my -- our presentation to the Subcommittee, I got a sense that you'd like to get an understanding of our basis for the review and the reasonable assurance determination of the adequacy of the steam dryer at Hope Creek. So I wanted to give you the -- what I call the seven major elements that give us that comfort feeling about the adequacy of the dryer and the reasonable assurance.

First thing is really our extensive review, which included multiple rounds of RAI with over 100 questions pertaining to steam dryer specifically and audits -- a two-day audit at CDI last year with four staff members and three contractors. Typically, we don't do that for every safety evaluation review amendment.

Hope Creek -- the number 2 element is Hope Creek steam dryer is a robust curved hood design, which is third generation GE steam dryer design, leading to less turbulent flow through the steam dryer and into the main steam lines. That's another second element. The third element -- base on baseline inspections, the BWR VIP guidelines, the Hope Creek steam dryer has not experienced fatigue cracking over 20 years of operation. To our knowledge, no main steam line legs at the Hope Creek, which are known to cause acoustic high peaks.

Substantial fatigue stress margin I think Mr. Davison mentioned. It's a factor of 2.1 for EPU, which includes end-to-end bias errors and uncertainties, which is comparable to that accepted by the staff through DOI and endorsed by ACRS by your Committee.

The last element is plant monitoring during power ascension, which includes five attributes. First is captured in the licensing condition. Number two is slow and deliberate increase in power. Number three is hold points trending in inspection -- and the walkdowns. And number four, the steps to follow if Level 1 or 2 limits curves are exceeded. And, finally, the inspection program that Hope Creek has committed to do, according to the BWR VIP 139, and the long-term monitoring of plant parameters for indications of steam dryer failure.

As we all know, if there are fatigue

failures, it will take place fairly quickly based on number of cycles, probably three months or four months. You will get -- if that kind of phenomena would exist, you will see it right away. So --

CHAIRMAN SHACK: That's why I wonder why you make such a point of the 20 years of operation. I mean, if you're -- if you're below, it will last forever. If you're above, it will be gone in a couple months.

MR. SHAH: Actually, some of the -- I think product failure could take several years, not like in other plants, could take five, six, seven years. So --

MR. MANOLY: After that, I think Dr. Chakrapani will go through the detail of the presentation, and we'll proceed on.

DR. BASAVARAJU: I am Chakrapani Basavaraju, and in this open session I will give you some details of staff review. Just to reiterate, this Hope Creek steam dryer is a design that -- it's a curved hood type of design, an improvement over the square hood and slant hood types.

And this steam dryer was modified and strengthened in 1986 before it was put into operation. And as was told, these were -- these dryers have not experienced any fatigue cracking during the past 20 years of operation.

Next slide.

The NRC staff, with contract support from ANLT, has evaluated the steam dryer analysis as well as the steam dryer input loadings, and NRC also performed an audit to review the steam dryer calculations and the model test facilities and the analysis performed, ACM as well as finite element.

And the uncertainties in steam dryer analysis are quantified, and still this dryer maintains a significant margin to fatigue limit of 13,600. Approximately, it's half, so it's like 8,000 -- 7,000 psi.

CHAIRMAN SHACK: Now, you have a margin of about 2.1. What margin would cause you discomfort?

DR. BASAVARAJU: Until now, based on the experience of the last two EPUs, we were targeting to maintain a margin of two for EPU conditions. For current license power for this Hope Creek it has a margin of three, and for EPU it has a margin of 2.1.

So this gives a reasonable assurance that the Hope Creek steam dryer is within structural limits for CLTP and the extrapolated EPU conditions. In addition --

MR. MANOLY: Let me just add, I think after we evaluate the Susquehanna results, we may have a different view on what margin we wanted to go to. But that's -- because they're going to do measurements on the dryer and compare that to the estimated --

CHAIRMAN SHACK: But they're only going to compare those with the Model 2 of the acoustic circuit model.

MR. MANOLY: Yes, yes, that's --

CHAIRMAN SHACK: Right?

MR. MANOLY: True. Yes. I mean, it's an evolving technology. I mean, what --

CHAIRMAN SHACK: You might make decisions before you have that data.

DR. BASAVARAJU: So continuing, during the power ascension phase of EPU, the dryer will be -- the dryer data will be monitored on an hourly basis, and the trending of the main steam line strain gauges taken. And there is a deliberate slow power ascension, and there will be higher percent power levels, and there will be evaluations and walkdowns.

And the data will be submitted for NRC's review, and that will be compared to limit curves, which is to reach the full power in the unit. And there is a Level 1 and Level 2. Level 2 will be 80 percent, 20 percent below the margin.

And whenever we see, whenever there are trends to showing indication of any resonances, the power ascension will be stopped and lower -- to the next lower step, and then the delta will be monitored and the dryer will be reevaluated with the observed data to make sure that the integrity of the steam dryer is maintained, and the evaluations will be submitted to the NRC.

And, additionally, the steam dryer will be inspected to BWR VIP 139 inspection guidelines to make sure that no fatigue-related cracks developed. And then, the EPU startup procedure is also submitted to NRC. And the walkdowns and inspections and the procedures used for the steam dryer were also reviewed by NRC.

So, in conclusion, we have reasonable assurance that the steam dryer, with all of the end-to-end uncertainties included, maintain significant margins for CLTP and extrapolated EPU conditions. And the license conditions established the origins for monitoring and evaluating the plant data during power ascension and take appropriate steps and actions if there is an exceedance or any resonance peaks noted during the power ascension phase.

And also, the long-term steam dryer inspection program gives confidence that no fatigue-related cracks are developing. And with this, we -- the staff has reasonable assurance that the steam dryer is acceptable for EPU operation.

So that concludes the open session of our status review.

MEMBER ABDEL-KHALIK: Thank you.

MEMBER ARMIJO: Will the steam dryer be inspected after every refueling outage, or every other refueling outage? How frequently will that be done?

DR. BASAVARAJU: BWR VIP guidelines gives specific details of what portions of the steam dryer will be inspected with every refueling outage, which portions, what susceptible areas from the past experience are inspected. So there will be inspections at every refueling outage, but specific areas --

MEMBER ARMIJO: Right. The most vulnerable will be looked at --

DR. BASAVARAJU: Right.

MEMBER ARMIJO: -- more frequently, I know that. But is it going to be every refueling outage there will be some sort of a fatigue inspection or inspection for fatigue or other damage?

DR. BASAVARAJU: Yes.

MEMBER ARMIJO: Okay.

MR. SHAH: I think according to the licensing condition, it will be inspected two times after each -- during the refueling of the plant, and these licensing conditions will expire.

MEMBER ARMIJO: Okay. That's what I wanted to know. Thank you.

MEMBER ABDEL-KHALIK: Are there any other questions for the staff during this open session?

(No response.)

If not, we will now proceed to a closed session where the licensee and the staff may present proprietary information. I will call on the designated federal official to make sure that those who have the appropriate clearances to participate in these closed sessions actually do.

(Whereupon, the proceedings went into Closed Session.)

MEMBER ABDEL-KHALIK: Are we reconnected on the phone? Dana, are we reconnected on the phone? Could you please reestablish the open phone connection?

(Pause.)

MEMBER ABDEL-KHALIK: Okay. Thank you.

We are back in session. This is an open session.

At this time, I'd like to go around the table to see if members have specific comments regarding the presentations we heard today and/or things that we heard during the Subcommittee meeting, if they had attended.

Mr. Sieber?

MEMBER SIEBER: Okay. I've reviewed the application and the SER, attended the Subcommittee meeting, and today's meeting, which is -- further elaborates on issues that arose during the Subcommittee, and I conclude that I see no impediments to the staff's issuing a license change to -- for the EPU condition.

MEMBER ABDEL-KHALIK: Thank you.

Dr. Banerjee?

MEMBER BANERJEE: So I am in agreement with Jack, but I have a more general remark which I think does not necessarily apply to this specific application, but is one which I think the staff should take cognizance of, because I feel that we are on very sort of shaky ground when it comes to connecting these measurements, which are being made in the steam lines to what is actually happening in the dryer.

And we have data now from various plants, which -- and we will have more data in the near future from Susquehanna where this sort of connection, using some sort of a more defensible model than we have seen to date, could be done. And I would urge the staff to do whatever is necessary to develop such a model in as short a time as possible, so that we don't have to go around this mulberry bush again and again and again, trying to connect these measurements which have been made in the steam lines as to what is happening in the dryer.

MEMBER ABDEL-KHALIK: Thank you. Dr. Armijo?

MEMBER ARMIJO: I share Mr. Sieber's conclusion. I think the EPU is in good shape. A lot of the -- I attended Subcommittee meetings as well. There are a number of things that weren't mentioned at the full Committee that I think both the staff and the applicant should be commended for.

I think the work that has gone into the plant materials and water chemistry to prevent unexpected failures or to mitigate against well-known failure mechanisms has been excellent, good resistant materials, good water chemistry. And we didn't discuss that today, but that was a plus.

The core and the fuel have been very conservatively designed for power uprate. Prudent measures have been taken. I don't think there will be any problem with respect to the core and fuel. I think the steam dryer -- I think everything that can reasonably be done has been done. The plant's geometry is such that the steam dryer isn't -- won't operate under the -- will actually operate under milder conditions than the previous plants that have had problems. They do have a quiet plant.

They've strengthened -- substantially strengthened the dryer, so that will help. And, of course, all of the instrumentation that has been put in the steam lines and the monitoring and the slow ascension, I think the steam dryer will be in good shape. So I think the -- everyone is very well prepared, and the EPU should be granted.

MEMBER ABDEL-KHALIK: Dr. Powers?

MEMBER POWERS: My general impression, and

I have only had the input here, is that the applicant has done a good job and it has been well reviewed. And this has become -- we have accommodated this problem a lot.

Professor Banerjee is correct -- it's not an easy thing to do, and we need to make it a more routine sort of thing internally. But basically this looks like it's in pretty good shape.

MEMBER ABDEL-KHALIK: Thank you. Dr. Bonaca?

VICE CHAIRMAN BONACA: Yes. I have attended the Subcommittee meeting, as well as this meeting, and I don't see any showstoppers for these plants. I think that they have a convincing application, a good SER. But I second the comments of Dr. Banerjee. I think that's an important view, and I think that that should be pursued by the staff.

I have no further comments.

MEMBER ABDEL-KHALIK: Dr. Shack?

CHAIRMAN SHACK: Roughly the same sort of thing. I mean, we're accepting this acoustic model, which really lets us predict the stresses on the basis of a very limited database for validation. I mean, we have some -- I'm comfortable in this particular case, because we end up with reasonable margins.

But again, you know, it's going to be difficult when you start showing data less than a factor of two is to know whether you really have characterized the uncertainties in the agreement in the model well enough, and we really need more validation.

MEMBER ABDEL-KHALIK: Dr. Wallis?

DR. WALLIS: Well, I'm not a member. I'm a consultant. I have submitted my report following the Subcommittee meeting. And, of course, you've read it with understanding, and I see no reason to change what I wrote there as a result of what I heard today.

MEMBER ABDEL-KHALIK: Thank you. Mr. Maynard?

MEMBER MAYNARD: I agree with everything that has been said. I do believe that the staff and the applicant both did a good job of preparing the application, reviewing the application, and very impressed with having the backup information and actual data and everything available, which I think made our review a lot easier and much more coherent.

I'm confident that the monitoring program they have in place for the dryer will identify problems early. I think we'll identify if there's any issues.

I think the acoustic monitoring is good. I think that over time we may be able to develop a lot more confidence in the actual quantitative aspects of it. I'm not relying totally on the quantitative aspect. It's more the monitoring that has been done that I think will allow trends to be caught or unexpected deviations, and I think some of the other monitoring, such as the water level, the pressure pulses affecting level instrumentation, some of those things are what provide me the overall confidence.

I am concerned, like has been mentioned before, we may be starting to focus too much on dryers. Maybe we need to start focusing on some other things. Everybody is sensitive to the dryer issues. Everybody is dealing with those, and I think that as a Committee we need to take a look at what are we putting our time in on, and, you know, are we -- if everybody is focusing on the same thing, who is focusing on some of the other things that might really need to be looked at.

My last thing is on the 1/8 scale test. I'm a little concerned that we -- some of our questions and the staff questioning, I would hate for us to start discouraging tests of this nature. I think that there are some good things that come out of it. I think we have to be careful that we don't make it where applicants say, "Well, the heck with it. I'm just not going to put the money in doing some of this stuff." So I think we need to make sure that we recognize some of the benefit from some of the scale model testing, too, as well as some of the limitations and stuff. So that's my comments.

MEMBER ABDEL-KHALIK: Dr. Bley?

MEMBER BLEY: I did not attend the Subcommittee meetings, but I would second everything, especially the things Mr. Maynard has said. The presentations and the situations seem a little cleaner than the other cases I've seen.

MEMBER ABDEL-KHALIK: Dr. Corradini?

MEMBER CORRADINI: I also did not attend the Subcommittee. Everything I've heard was from today. I guess the one thing I'd emphasize, I don't disagree with anything we've heard from any of the other members. I guess the one thing I'd emphasize is I guess if there was a couple messages to send, one message is that if the applicant wants to do things experimentally to learn more, we should encourage that. We should not discourage it.

Second thing is I think Sanjoy's point about -- that the staff has got to come up with a technique that they feel confident in, so they can move forward on a regular basis is very important.

The only other thing is is that I'm trying to think from all of the other presentations that we had prior to the dryer, we might want to ourselves discuss privately what other things, as we continue to deal with these extended power uprates, what we might prioritize once this becomes regularized and everything is all hunkydory, at least from the standpoint of analysis, what other things concern us that are coming up, and decide to plan on learning more about it, because some of the other things are of interest, it just didn't turn out in this case to be of any consequence.

VICE CHAIRMAN BONACA: I will also point out that the presentation to the Commissioners in two months will include a presentation on issues associated with EPU. And I think that it's -- you know, we will have to in fact sit around this table and probably the next meeting --

MEMBER CORRADINI: Maybe that's a reason to think it through.

VICE CHAIRMAN BONACA: -- finalize those bullets and --

MEMBER BANERJEE: We have raised such

issues with other EPUs.

MEMBER CORRADINI: Right. But to -- as others have said, I will just -- I'm just repeating it, is that there is other things that we need to focus on -- our attention on. This might be a time, assuming staff, as you are suggesting, is trying to develop a regular approach to this.

MEMBER ABDEL-KHALIK: Thank you.

At this time, on behalf of my colleagues, I'd like to express our appreciation and thanks to both the licensee and the staff for the quality of the application and the review.

I would like to point out that the Committee will begin deliberations on a draft letter. Because of the change in the schedule, we will do that immediately after lunch today. So if either the staff and/or the applicant would like to remain for those discussions, I invite you to do so.

At this time, I'd like to pass on the gavel to Dr. Bonaca. This session is closed -- this session --

(Laughter.)

VICE CHAIRMAN BONACA: We are going to take a break for lunch, and we are going to get together again at 1:00.

(Whereupon, at 11:54 a.m., the proceedings in the foregoing matter went off the record for a lunch break when the proceeding resumed in Closed Session.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: Advisory Committee on Reactor Safeguards

Docket Number: n/a

Location: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

James Salandro
Official Reporter
Neal R. Gross & Co., Inc.

551st ACRS Full Committee

NRC Staff Review of Proposed Extended Power Uprate For Hope Creek Generating Station

April 10, 2008

Opening Remarks

Catherine Haney

Director

Division of Operating Reactor Licensing

Office of Nuclear Reactor Regulation

Introduction

John G. Lamb

Senior Project Manager

Division of Operating Reactor Licensing

Office of Nuclear Reactor Regulation

Introduction

- Background – Hope Creek
  - Boiling Water Reactor – GE Type 4
  - Mark 1 Containment
  - Licensed July 25, 1986
  - Approved Measurement Uncertainty Recapture 1.4 % (46 MWt) increase above original licensed thermal power (3,293 MWt) – July 30, 2001
- PSEG Hope Creek Proposed Extended Power Uprate (EPU)
  - 3,339 to 3,840 Megawatts Thermal (MWt)
  - 15% increase (501 MWt)

Topics for April 10th

- Containment Analyses - PSEG
- PRA – PSEG
- Materials, Containment, & PRA – NRC
- Fuel Methods – PSEG & NRC
- Steam Dryer – PSEG & NRC

Materials Engineering Topics

John G. Lamb &

Matthew A. Mitchell

Branch Chief

Vessels & Internals Integrity Branch

Division of Component Integrity

Office of Nuclear Reactor Regulation

Reactor Vessel Integrity

- Reactor Vessel Material Surveillance Program
  - Licensee has implemented BWRVIP Integrated Surveillance Program
- Reactor Vessel P-T Limits
  - Existing P-T limits remain valid for 32 EFPY
- Reactor Vessel USE Analysis
  - Acceptable with all materials remaining above the 50 ft-lb screening limit

Containment Analyses

John G. Lamb &

Richard Lobel

Senior Engineer

Containment & Ventilation Branch

Division of Safety Systems

Office of Nuclear Reactor Regulation

Containment Analyses – Topics Reviewed by the Staff

- Containment functional design
  - Peak containment pressure and temperature
  - Hydrodynamic loads
- Subcompartment analysis
- Mass and energy release
- Combustible gas control
- Containment heat removal (no credit for containment accident pressure in determining available NPSH of ECCS pumps)
- Secondary containment functional design
- Additional review items
  - Bulletin 96-06

Containment Analyses

- Large break LOCA is the limiting event
- Hope Creek did not need to take credit for Containment Overpressure – complies with RG 1.1
- Conservative analyses
  - ECCS Net Positive Suction Head (NPSH) Available is greater than NPSH Required
  - Suppression pool temperature is bounding

Hope Creek
Constant Pressure Power Uprate
NRC Staff Review of Risk Evaluation

Donald G. Harrison, Chief

Balance of Plant Branch

for

Probabilistic Safety Assessment Branch

Division of Risk Assessment

Office of Nuclear Reactor Regulation

EPU Risk Review Guidance

- Hope Creek CPPU application is not "risk-informed" (i.e., not evaluated per RG 1.174 acceptance guidelines)
- Application and review follows RS-001, Rev. 0, Matrix 13, "Risk Evaluation"
- Staff review is consistent with SRP Section 19.2, Appendix D, "Use of Risk Information in Review of Non-risk-informed License Amendment Requests"
  - Determine if "special circumstances" exist that could potentially rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements
  - In so doing, confirm that risks are acceptable

Safety Evaluation Revisions

- Addressed ACRS Subcommittee Comments
  - Percentage error corrected
  - Staff fire and seismic estimations replaced with qualitative observations based on information provided in the licensee application and RAI responses
  - Related discussions changed to reflect the above revisions
- Revised staff approach/text is consistent with prior EPU safety evaluations

EPU Risk Review Conclusions

- Licensee has adequately modeled and/or addressed the potential risk impacts of the proposed EPU and determined they are acceptably small.
- The proposed EPU does not create "special circumstances" that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements.

Staff Conclusions – Materials, Containment, & PRA

- Reactor Pressure Vessel meets the NRC regulations.
- All Containment parameters remain below design limits.
- Risks are acceptable because RG 1.174 risk acceptance guidelines are met.

Fuel Methods Evaluation
Hope Creek EPU

Tony Nakanishi
Reactor Systems Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

Dr. Peter Yarsky
Nuclear Performance and Code Review Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

Review Objective and Scope

Objective

- Assess applicability of approved GE Fuel Methods to HCGS EPU conditions

Scope

- Limited to topics included in NEDC-33173P-A "Applicability of GE Methods to Expanded Operating Domains"
- Applicability of GE Methods to co-resident SV96+ fuel

NEDC-33173P Overview

- NEDC-33173P-A describes applicability of GE neutronic and T-H methods for BWR EPU and M+ applications
- Staff approved NEDC-33173P-A with limitations and conditions
- ACRS concurred with staff conclusions
- HCGS referenced NEDC-33173P for EPU application

Review Approach

Ensure compliance to plant specific application process specified in staff SE on NEDC-33173P-A

- Topical Report Limitations
- Key core parameters within operating experience

Additional assessment of co-resident SVEA 96+

NEDC-33173P-A Compliance

- HCGS complies with all applicable limitations specified in NEDC-33173P-A

Applicability of GE Methods to Co-Resident SVEA-96+

- SVEA 96+ bundles operate with in-channel P/F ratios at pre-EPU conditions, well within operating experience
- SVEA 96+ bundle loading is appropriate
- TGBLA06/PANAC11 adequately models SVEA 96+ for HCGS EPU Cycle 15

Review Conclusions

NEDC-33173P-A is appropriate for HCGS EPU

- HCGS complies with all applicable Topical Report limitations
- GE Methods is applicable to co-resident SVEA 96+ for HCGS EPU Cycle 15
- HCGS EPU will operate within the current experience base

Safety Analysis Review

Review Area	Result	Acceptable?
ASME Overpressure	Peak pressure 1285 psig, <1375 psig ASME Limit	Y
Transient Analysis	Cycle-specific OLMCPR determined	Y
LOCA	Cycle-specific MAPLHGR determined	Y
Stability	Cycle-specific Option III setpoints determined	Y
ATWS	Peak pressure 1437 psig, < 1500 psig ASME limit PCT 1446°F < 2200°F Peak suppression pool temperature 199°F < 201°F	Y

Safety Analysis Review Conclusions

- Safety analyses are based on NRC-approved methodologies, analytical methods, and codes
- Analysis scope is consistent with NRC-accepted guidelines and generic evaluations (ELTR-1, ELTR-2, and CPPU)
- EPU impact on Hope Creek safety analysis is acceptable

Component Evaluation for Hope Creek Proposed EPU Amendment

Kamal Manoly, Chief
Mechanical & Civil Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

April 10, 2008

NRC Review Team

- NRR/DE

Kamal Manoly
Dr. Chakrapani Basavaraju
Alexander Tsirigotis

- NRO/DE

Thomas G. Scarbrough
Dr. Cheng-Ih (John) Wu

- Argonne National Laboratory

Dr. Vikram N. Shah, ANL
Dr. Stephen A. Hambric, Head, Structural-Acoustics Department, Applied Research Laboratory, Pennsylvania State University
Dr. Samir Ziada, Professor and Chair, Mechanical Engineering Department, McMaster University

Hope Creek Steam Dryer Performance

- Curved hood design improvement over square and slanted designs
- Steam dryer modified based on operating experience from other nuclear power plants to improve its structural integrity prior to initial plant operation in 1986
- No significant steam dryer performance issues during 20-year plant operation

Steam Dryer Review

- NRC staff with contract assistance evaluated steam dryer analysis with numerous information requests and meetings
- NRC audit at licensee contractor (CDI) office in May 2007 reviewed steam dryer calculations, scale model test facility, and computer analysis
- Uncertainties in steam dryer analysis covered by margin to fatigue stress limit
- Reasonable assurance that Hope Creek steam dryer is within structural limits for CLTP and extrapolated EPU conditions

Power Ascension

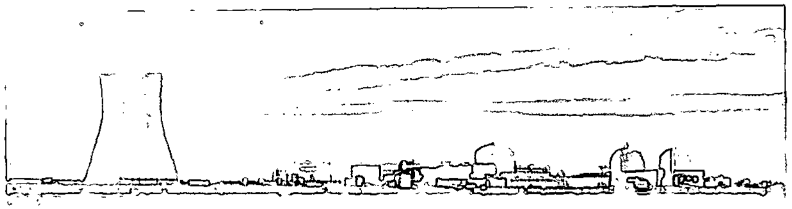
- Hourly monitoring and trending of Main Steam Line strain gages
- 96 hour hold at 5% power levels for data evaluation, walkdowns, and NRC staff review
- If unacceptable data trend or limit curve exceeded, stop power ascension and evaluate data
- Submit updated dryer reports during and following power ascension
- Long-term steam dryer inspection program

Development of License Conditions

- Provide slow and deliberate power ascension with hold points and data evaluation
- Formalize plans for steam dryer and plant instrumentation and other activities
- Specify EPU startup procedure contents
- Provide for licensee/NRC interaction to address plant data, evaluations, walkdowns, inspections, and procedures

Conclusion

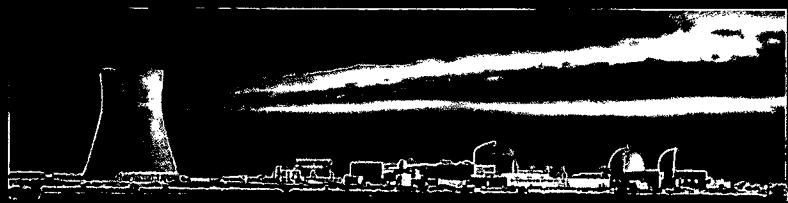
- Steam dryer analysis with uncertainties reflects significant margin to structural limits for CLTP and extrapolated EPU conditions
- License conditions establish provisions for monitoring and evaluating plant data during power ascension, and taking prompt action if necessary
- Long-term steam dryer inspection program established
- Steam dryer acceptable for EPU operation



Hope Creek Generating Station Extended Power Uprate

Advisory Committee on Reactor Safeguards
April 10, 2008

HopeCreek
GENERATING STATION



Probabilistic Safety Assessment

E.T. Burns

Hope Creek Risk Management Team

HopeCreek GENERATING STATION

Background

ACRS Subcommittee presentation (March 21)

- Scope and quality
- Quantitative results for Internal Events
- Qualitative assessment of External Events
- Conclusion: Risk change resulting from EPU implementation is very small

ACRS Subcommittee requested additional detail regarding disposition of individual fire and seismic accident sequences affected by EPU implementation

Risk Evaluation Methods

- Identify plant configuration and procedural changes due to EPU
- Use updated PRA models for internal events consistent with ASME PRA standard
- Use available IPEEE fire and seismic PRA models
- Identify PRA elements affected by changes
- Incorporate hardware and procedure changes in PRA model
- Use realistic success criteria and limits
- Compare results with Reg. Guide 1.174 guidelines

Fire Risk Evaluation

Scoping analysis from IPEEE available

Conservatisms bias results of Fire Scoping Study

EPU effects

- No increase in combustible loading
- No new fire initiating events or increased fire frequency

Quantitative fire PRA model calculations:

- Examine critical fire scenarios from IPEEE
- Loss of equipment or access to equipment leads to CDF regardless of EPU implementation
- Negligible impact on decay heat removal scenario HEPs because of long times available for response/recovery actions

Fire Quantitative Results

Examine the 16 fire induced core damage scenarios (95% of fire CDF)

- Dominant EPU effect related to reduced time available for manual actions
- Results: ΔCDF = 7.0E-8/yr

Bound the residual fire induced CDF (5% of fire CDF)

- Use worst case effect of decrease in allowable time for crew action
- Results: ΔCDF = 3.0E-8/yr

Total risk change due to fire induced CDF

- Results: ΔCDF = 1.0E-7/yr

EPU has very small impact on fire risk profile

Qualitative Seismic Risk Evaluation

Seismic PRA scoping model available

89.6% of contributors are hardware failures

EPU effects

- No change in seismic qualification for SSCs
- No significant change in equipment mountings or anchorages
- No new seismic vulnerabilities identified
- Dominant contributors to seismic risk (89.6%) are related to equipment failures with no operator actions credited

Bounding Calculation: Seismic

Dominant contributors

- 89.6% of seismic sequences result in CDF due to direct hardware failure
  - No operator response can be credited therefore, no ΔCDF results for these sequences

Residual contributors

- 10.4% of the seismic sequences may involve some crew failure to lead to core damage.
- Sequence No. 3 (SDS-26) at 1.97E-7/yr contributes 1.4E-8/yr to ΔCDF due to changes in time available for manual action
- Assume all other residual seismic sequences (5.1%) have impact associated with reduced time available for crew response:

$$\Delta CDF_{\text{Residual Seismic}} = CDF_{\text{Seismic}} \times 0.051 \times 0.072$$
$$= 3.6\text{E-6/yr} \times 0.051 \times 0.072 = 1.3\text{E-08/yr}$$

Bounding Calculation: External Events

Quantitative Summary

Hazard	ΔCDF (per yr)
Seismic	2.7E-8
Fire	1.0E-7
Internal	6.8E-7
Total	8.1E-7

Result: Region III - very small risk [Compared with RG 1.174 Acceptance Guideline]

HopeCreek GENERATING STATION

Summary of EPU Risk Impact

Risk impact evaluated using standard PRA methods and incorporated seismic and fire scoping analyses

Quantified risk impact is a small percentage of current plant risk

ΔCDF is a very small risk change per Reg. Guide 1.174

Containment Analysis

Paul Davison

Site Engineering Director

HopeCreek
GENERATING STATION

DBA LOCA Containment Analyses

Containment Analyses Comparison

ECCS NPSH Determination

DBA LOCA Containment Analyses

Analyses

- Analyses at or above 102% of 3840 MWt
- Decay heat by ANS/ANSI 5.1-1979 with 2σ uncertainty
- Passive heat sinks credited in long term analysis

Results

- All containment parameters remain below design limits

Hope Creek DBA LOCA Containment Performance Results

Parameter	EPU Results	Design Limit
Peak Drywell Air Space	50.6 psig	62 psig
	298°F	340°F
Peak Bulk Pool Temp	212.3°F	218°F
Peak Wetwell Air Space	27.7 psig	62 psig
	212.2°F	310°F

RHR and Core Spray NPSH-Available Assumptions

- Bulk pool temperature = 218°F
- Containment pressure = 14.7 psia

NPSH-Required Based on Maximum Tested Flows

NPSH-Available > NPSH-Required

Containment Overpressure Not Credited

Fuel Dependent Response & Methods

Donald V Notigan

Nuclear Fuels Manager – BWR Design & Analysis

HopeCreek
GENERATING STATION

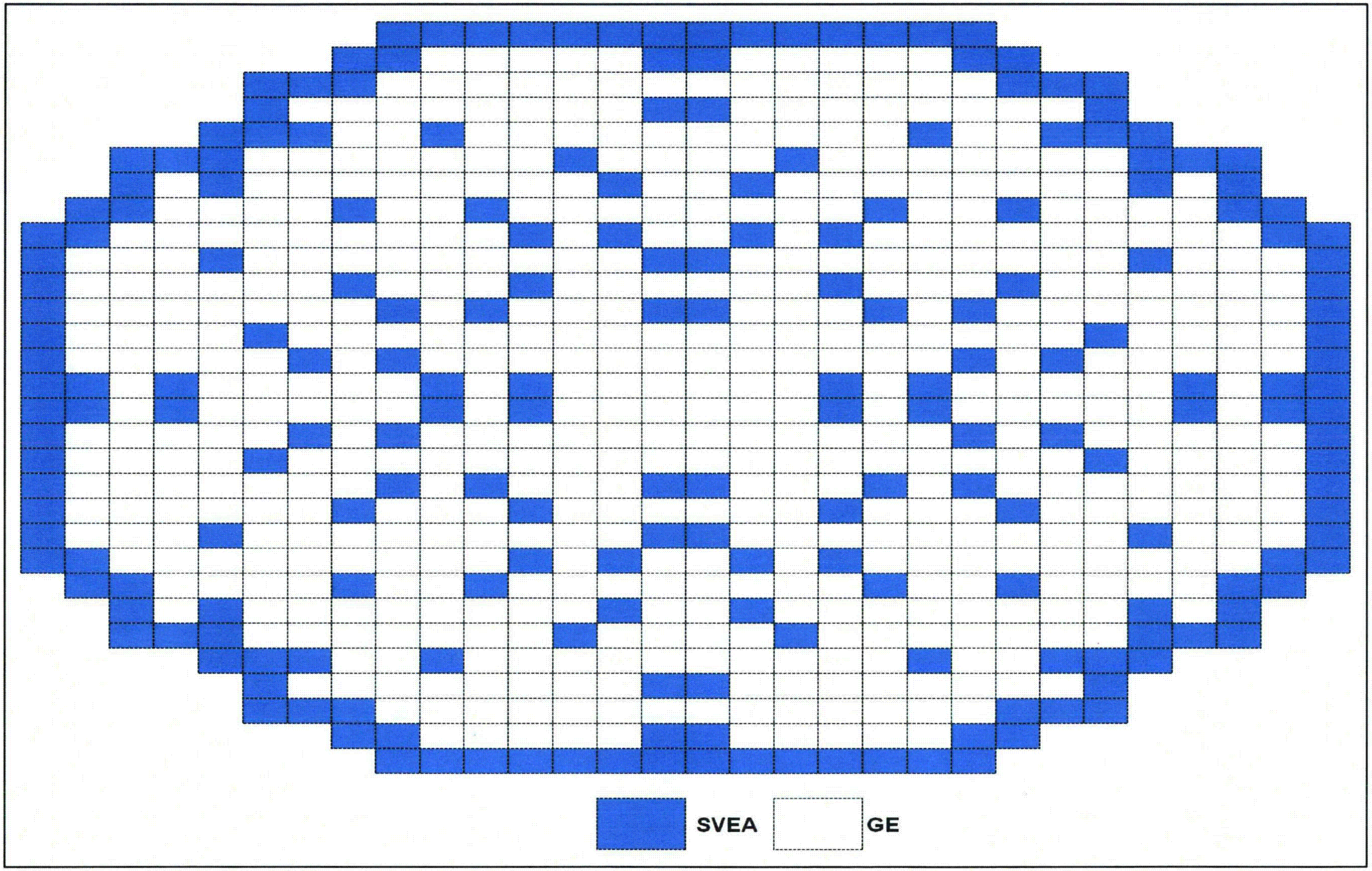
Core Loading Map for EPU Operation

Fuel Performance and Core Design

Safety Analysis

Conclusions

Hope Creek EPU Core Loading Map



EPU Core Design

- All fuel assemblies have PCI resistant design (barrier liner clad) and integrated debris filter features
- SVEA fuel is low reactivity fuel and in non-limiting core locations
- SVEA fuel will operate with maximum bundle powers below pre-EPU levels
- GE fuel delivers 81% of EPU core thermal power
- All EPU core design calculations and reload safety evaluations are complete

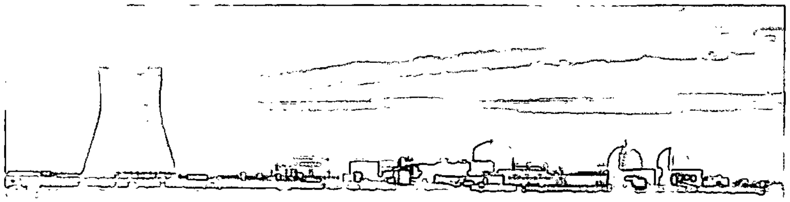
Safety Analysis

- All thermal limits were met with margin remaining for GE & SVEA fuel (MFLCPR, MAPLHGR and MFLPD)
- SVEA fuel does not contribute to core Safety Limit MCPR
- Key safety analysis parameters will remain consistent with EPU reference plant operating experience base
- All applicable limitations, conditions and adders from NRC approved Licensing Topical Report NEDC-33173P were fully incorporated into EPU safety analysis

Conclusion

- Applied NRC approved GE nuclear analysis methods
- All EPU cycle specific core design calculations and reload evaluations are complete
- Incorporated all applicable limitations, conditions & adders from NRC approved Licensing Topical Report NEDC-33173P
- SVEA fuel is non-limiting in EPU core operation
- Fuel performance is consistent with EPU reference plant operating experience base for key parameters important to safety

Safe Operation of Fuel for Hope Creek 115% Extended Power Uprate Confirmed



Steam Dryer & Power Ascension Testing

Paul Davison

Site Engineering Director

HopeCreek
GENERATING STATION

Steam Dryer Overview

Robust Steam Dryer Design

Quiet Plant

Dryer Analysis

Results

Steam Dryer Power Ascension

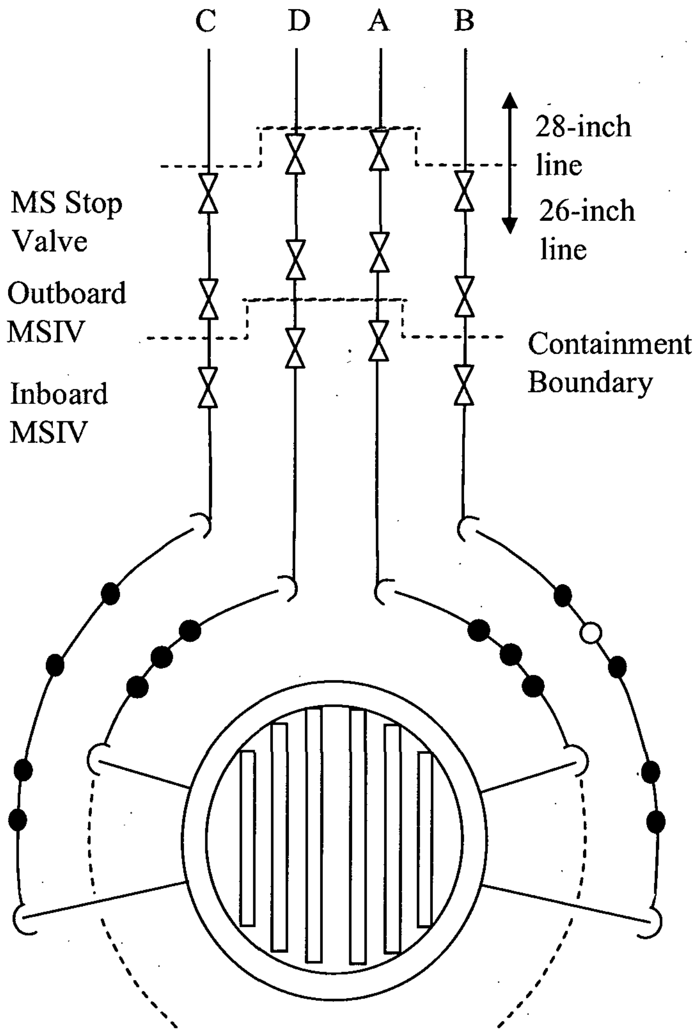
Curved Hood

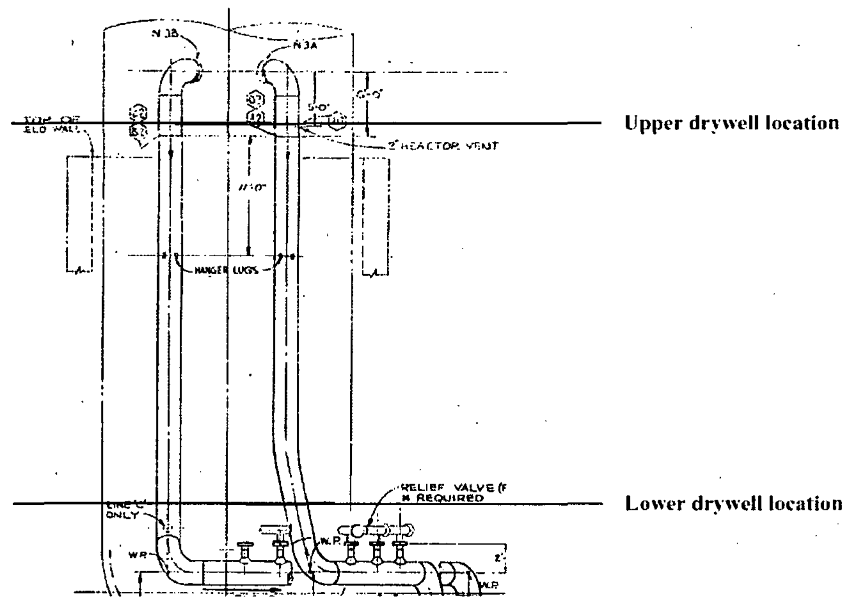
- 3rd generation of GE steam dryer design
- Modified on-site prior to operation

Baseline Inspections Done

- Per BWR VIP recommendations
- No fatigue cracking identified

Hope Creek Main Steam Lines





Dashed red-lines show location of <u>SSES</u> 26-inch MSL dead branches, which do not exist at HCGS.

HopeCreek GENERATING STATION

Acoustic Circuit Model Revision 4

- 0 Hz - 60 Hz - Incorporation of additional source to improve low frequency loading

Revision 4 identical to Revision 2 for 60 Hz – 200 Hz

Revision 4 Validation

- Based on benchmark of Quad Cities data at Hope Creek's EPU Mach number
- Blind benchmark of Quad Cities completed at higher Mach number
  - ACM predictability same at both power levels

Finite Element Modeling

- Standard code ANSYS 10.0
- Independently validated CDI's capability to model a complex structure
- Mesh convergence study completed
- Independent third party review

Results

- Lowest alternating stress ratio is 2.18
- All bias and uncertainty at EPU conditions

Steam Dryer Power Ascension

Monitoring

- MSL strain gages
- MSL accelerometers
- MSL moisture carryover

Evaluation

- Strain gage limit curves
- Power ascension rate of ≤ 1% CLTP/hr
- Collection of strain gage data at every 1% increase
  - Used for trending
- Evaluation every 2.5% power
- Power plateaus at each 5% power step and final EPU power

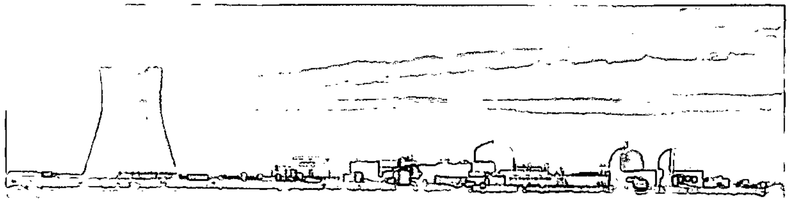
Reporting

- Provide data for NRC review at each plateau (5% power)

Power Ascension Test Matrix

% Power	MSL Strain Gage	Moisture Carryover	RW Level	MSL Accel	SRV Accel	FW Accel	Ext Steam Accel	Recirc Accel	Non Critical FIV	Rad Survey	Core Perform	Chem Data	FW Run-Out	DEHC Press Change	Sys Perf Mont	FW Step Change
90	X	X	X	X	X	X	X	X			X		X	X	X	X
100	X	X	X	X	X	X	X	X			X		X	X	X	X
101	X		X	X	X	X	X	X								
102	X		X	X	X	X	X	X								
102.5	X	X	X	X	X	X	X	X							X	
103.5	X		X	X	X	X	X	X								
104.5	X		X	X	X	X	X	X								
105	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**
106	X		X	X	X	X	X	X								
107	X		X	X	X	X	X	X								
107.5	X	X	X	X	X	X	X	X							X	
108.5	X		X	X	X	X	X	X								
109.5	X		X	X	X	X	X	X								
110	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**
111	X		X	X	X	X	X	X								
111.5	X	X	X	X	X	X	X	X			X				X	
111.5 CF	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**
112.5	X	X	X	X	X	X	X	X							X	
113.5	X		X	X	X	X	X	X								
114.5	X		X	X	X	X	X	X								
115	X	X	X	X	X	X	X	X			X				X	
115 CF	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**	**X**		**X**	

CF – Cross Flow Applied



Steam Dryer & Power Ascension Testing -

Paul Davison

Site Engineering Director

HopeCreek
GENERATING STATION

Quiet Plant

Steam Dryer Power Ascension

Pages 31-33 – Proprietary
(removed)

Conclusions

- Robust Steam Dryer Design
- Low MSL Acoustic Signature
- Improved & Benchmarked Modeling That Contain Biases & Uncertainties
- Dryer Stress Ratios Above 2.0
- Comprehensive Power Ascension Test Plan
- High Confidence in Considerable Margin Against Steam Dryer Fatigue