June 29, 2017

MEMORANDUM TO: ACRS Members

FROM: Maitri Banerjee, Senior Staff Engineer **/RA/**
Technical Support Branch
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS APR1400 SUBCOMMITTEE ON MAY 19, 2017, ROCKVILLE, MARYLAND

The minutes for the subject meeting were certified on June 27, 2017. Along with the transcripts and presentation materials, this is the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc with Attachment: A. Veil
M. Banks
C. Brown

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

MEMORANDUM TO: Maitri Banerjee, Senior Staff Engineer
Technical Support Branch
Advisory Committee on Reactor Safeguards

FROM: Ronald Ballinger, Chairman
APR1400 Subcommittee
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFIED MINUTES OF THE ACRS APR1400 SUBCOMMITTEE MEETING ON MAY 19, 2017

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on May 19, 2017, are an accurate record of the proceedings for that meeting.

/RA/	**June 27, 2017**
Ronald Ballinger, Chairman APR1400 Subcommittee	Dated

Certified on: June 27, 2017
Certified by: Ronald Ballinger

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE APR1400 SUBCOMMITTEE MEETING ON
MAY 19, 2017, ROCKVILLE, MD

The ACRS APR1400 Subcommittee held a meeting on May 19, 2017 in T2B1, 11545 Rockville Pike, Rockville, Maryland. The meeting convened at 8:30 a.m. and adjourned at 5:30 p.m.

ATTENDEES

ACRS Members/Consultant/Staff:

R. Ballinger, Chairman	J. Stetkar, Member
G. Skillman, Member	J. March-Leuba, Member
D. Powers, Member	C. Brown, Member
M. Corradini, Member*	J. Rempe, Member
Andrea Veil, Executive Director	C. Brown, ACRS Staff (DFO)
Stephen Schultz, ACRS Consultant	
M. Banerjee, ACRS Staff *	

NRC Staff, Consultants & Other Attendees:

TIM DRZEWIECKI, NRO	WILLIAM WARD, NRO
JIM GILMER, NRO	SYED HAIDER, NRO
MICHELLE HART, NRO	DAN PRELEWICZ, NRO
RAUL HERNANDEZ, NRO	JIM STECKEL, NRO
SHANLAI LU, NRO	CARL THURSTON, NRO
JOE STAUDENMEIER, NRO	PETER YARSKY, RES
CHRIS VAN WERT, NRO	DOUGLAS BARBER, Consultant
DAVE CARAHER, ISL	JIM SERVACIOUS, Consultant

KHNP and Other Attendees:

ANDY OH, KHNP	WOOCHONG CHOU, KHNP
SUNG JU CHO, KHNP	UNG SOO KIM, KEPCO E&C
JAEHOON JEONG, KHNP	YOUGGUN KIM, KHNP and KEPCO
DONGSU LEE, KHNP and KEPCO	KAEYEOL LEW, KEPCO E&C
Robert Lee, Westinghouse*	ROB SISK, Westinghouse

*Attending via telephone

SUMMARY

The purpose of the meeting was for the ACRS members to receive briefings on the Korea Electric Power Corporation (KEPCO) and Korea Hydro and Nuclear Power Company (KHNP) design certification application (DCA) and NRC staff's review specific to Chapter 15, “Transient and Accident Analyses.” The meeting transcripts are attached, and contain a description of matters discussed at the meeting. The presentation slides and handouts used during the meeting are attached to these transcripts. The meeting was open to the public.

The following list describes significant issues discussed during the meeting with the corresponding pages of the transcript referenced. Unless specifically noted, the chapter and section references belong to the Design Certification Document (DCD) Tier 2 submittal or the NRC staff’s safety evaluation report (SER). Due to transcription and other limitations some parts of the transcript are not intelligible. A best attempt is made to capture the gist of the discussion.

SIGNIFICANT ISSUES	
Issue	Reference Pages in Transcript
Chairman Ballinger convened the meeting. Mr. Bill Ward, NRO Project Manager, and Rob Sisk, Westinghouse, consultant to the applicant, provided short introductions.	5-6
Mr. Ung Soo Kim, KEPCO E&C, started the applicant’s DCD Chapter 15 presentation noting the scope and introducing other presenters. He also noted the documents submitted by the applicant to support Chapter 15 DCD. Then he presented Section 15.1, Increase in Heat Removal by the Secondary System. He mentioned four anticipated operational occurrences (AOOs) and one postulated accident (PA) under this category, and presented two of the scenarios. The analysis for inadvertent opening of a steam generator relief or safety valve assumed a manual reactor trip before an automatic trip on high power level is reached. The PA presented was the main steam line break analysis. Member March-Leuba asked about the possibility of re-criticality.	7-12 Slides 2-8
Mr. Kim presented Section 15.2, Decrease in Heat Removal by the Secondary System, and noted the loss of condenser vacuum to be most limiting among all the AOOs in this category, and the PA to be feedwater line break inside and outside the containment.	12-13 Slides 9-11
Mr. Kim presented Section 15.3, Decrease in Reactor Coolant System Flow Rate. He noted one applicable AOO, the complete loss of forced reactor coolant flow; and two PAs, with the reactor coolant pump (RCP) locked rotor event being most limiting.	14-15 Slides 12-14
Mr. Kim presented Section 15.4, Reactivity and Power Distribution Anomalies. This category consisted of many AOOS and one PA, Spectrum of CEA Ejection Accidents. For uncontrolled withdrawal of a CEA from a subcritical or low-power startup condition transient, member March-Leuba	15-22 Slides 15-22

wanted to know how KHNP selected the rod for maximum reactivity injection rate. A discussion followed. There was discussion regarding the variable overpower reactor trip analysis setpoint, and the transcript is hard to follow. The basis for selecting 20 kilowatts per foot for the peak linear heat generation rate (LHGR) was discussed.	
Mr. Kim presented the other AOOs, Section 15.4.2 to 15.4.7. Regarding 15.4.3, Control Element Assembly Misoperation, member March-Leuba's question started a discussion on radial peak distortion and xenon imbalance, and a question if there was a technical specification limit on power asymmetry was taken for later follow up. Also, regarding Section 15.4.4, startup of an inactive RCP, member March-Leuba question if the scenario should consider addition of positive reactivity to the core, was taken for later follow up. Regarding Section 15.4.6, inadvertent decrease in boron concentration in the reactor coolant system, member Stetkar's question clarified that the boron dilution alarm was from the startup neutron flux detectors.	22-34 Slides 17-21
Section 15.4.8, CEA ejection events (PA) - ACRS Consultant Schultz asked if the KHNP modeling philosophy, both steady state and transient embrace the most recent information and tools on fuel thermal conductivity degradation (TCD) issue. KHNP responded that at this time they applied a penalty to consider TCD, and this approach is reflected in the related topical and technical reports. KHNP has developed a code, not licensed yet, and is working with NRC staff to update the TCD affected areas to incorporate the penalty to be complete by Phase V.	34-42 Slide 22
Member Power questioned why the site boundary doses were so low for above control rod accident. A discussion on failed fuel assumption followed, and a question was asked regarding the assumption of a peak LHGR of 25 vs. 20 kilowatts per foot for other accidents.	42-46
Mr. Kim presented Section 15.5, Increase in Reactor Coolant Inventory. It includes two AOOs, with the CVCS malfunction that increases the reactor coolant inventory being the limiting one.	46-47 Slides 23-24
Mr. Kim presented Section 15.6, Decrease in Reactor Coolant Inventory. As non-LOCA events the steam generator tube failure and letdown line break are analyzed. Source term for these PAs was discussed.	47-52 Slides 25-57
Mr. Woochong Chon presented Section 15.6.5, LOCA Resulting from Spectrum of Postulated Piping Breaks. Large Break LOCA (LBLOCA): Mr. Chon noted the revision of the topical report on LBLOCA methodology would reflect the TCD issue. Best estimate calculations are done. He presented the regulatory bases, acceptance criteria, sequence of events, computer code used, modeling methodology, core and system performance, and analysis results. A	53-69 Slides 28-37

discussion on bypass flow followed. Member Rempe asked about containment accident pressure (CAP) credit, and a quantification in specific numbers used in the analysis. Members asked for an explanation why the downcomer level was much higher than the core level. A discussion followed. Regarding the methodology, a future ACRS review of the LBLOCA topical report was mentioned.	
Small-break LOCA (SBLOCA): Mr. Chon discussed the analysis methodology, assumptions, and results. Upon member Corradini's question, Mr. Kaeyeol Lew, KEPCO, explained the conservative artificiality imposed on the water level parameter after COMPERC-II code gave unrealistic results.	69-73 Slides 38-41
Mr. Chon presented the issue of loop seal clearing and reformation, noting the loop seal reformation calculation done for several break sizes. Members explored the applicability of the Semiscale data to APR1400 geometry, how loop seal clearing was defined, bypass flow assumed in the calculations, and sensitivity analysis to see how conservative the model was. A discussion regarding Slide 33 (and 36), reference points for downcomer water level and the core collapse water level took place. Member Rempe reiterated her question why the applicant (and staff) was confident regarding the code benchmarking against Semiscale (RAI 15.06.05-19).	73-83 Slides 42-43 105-107
Mr. Chon presented the Post-LOCA Boron Dilution Analysis. This addresses boron dilution when the slug in the loop seal formed by the condensed steam in steam generator tubes enters the vessel. The physical processes of mixing and conservativeness of the assumptions were discussed.	83-87 Slides 44-45
Post-LOCA Long-Term Cooling (LTC): Chairman Ballinger clarified that the Committee was required to write a letter (to the Commission) on the subject. Mr. Shanlai Lu from the staff clarified that staff had addressed some aspects of the GSI-191, ECCS sump strainer blockage issue, during the Chapter 6 presentation, was ready to make a presentation on Chapter 15 analysis aspect of LTC today, and any additional briefing may be done coincident with the future presentation on the LBLOCA topical report.	87-89
Mr. Chon presented Section 15.6.5, Post-LOCA Long-Term Cooling. He explained the two different cooling methods for large and small break LOCAs. He presented the evaluation model noting initial conditions, codes used, and that an NRC approved interim method was adopted for APR1400 calculation. He presented the results.	89-92 Slides 46-50
Mr. Youggun Kim, KEPCO E&C, presented post-LOCA long-term cooling in-vessel downstream effect. He presented analysis assumptions, chosen LOCA scenarios, the test loop and debris input sequence, and evaluation of	93-105 Slides 51-57

test results for the scenarios. A discussion on test results took place.	
Mr. Dongsu Lee, KEPCO E&C, presented Section 15.7, Radioactive Material Release from a Subsystem or Component. This addresses the analysis method and radiological consequences of such release.	107-108 Slides 58
Mr. Lee presented Section 15.8, Anticipated Transient without Scram (ATWS). Discussion on diverse protection system and reactor power cutback system took place.	108-109, 111-112 Slide 59
Mr. Lee presented Appendix 15A, Radiological Consequence Analysis. His presentation addressed design bases and features, analysis methods, design evaluation for LOCA and Non-LOCA cases, and radiological consequences for DBAs. Discussion took place on containment spray testing and coverage, use of site chi/Q, reduction factors for automatic selection of ventilation intakes for the main control room (MCR) dose calculation, and HEPA filters in calculating MCR dose.	109-127 Slides 60-71
Mr. Lee completed the applicant's prepared presentation with a summary of DCD Chapter 15, and a tally of open items from NRC review.	128 Slide 72-74
Mr. Jim Steckel, NRO, started staff presentation on Chapter 15. Dr. Shanlai Lu, NRO, discussed staff review perspectives and approaches.	129-134 Slides 2-5
Dr. Lu presented an overview of staff review of Section 15.0, Transient and Accident Analyses, including a snapshot of two open items in their safety evaluation report (SER), review of the methodology and codes used. Staff presentation on LBLOCA will be with their future presentation on the topical report. Member Corradini asked staff to elaborate regarding the application of RELAP5/MOD3.3K to LBLOCA analyses that the conclusion was pending on the review of LBLOCA topical report. A discussion followed on staff use of TRACE code for their confirmatory analysis. Mr. Steckel noted staff used TRACE to perform the bounding analysis and results were well within regulatory limits.	134-143 Slides 6-12
Ms. Michelle Hart, NRO, presented staff review of Section 15.0.3, Radiological Consequences. She mentioned the scope, methodology used, and results obtained. She mentioned her confirmatory analysis. She mentioned a finding in that the APR1400 design calls for automatic reopening of the (MCR) intakes on a periodic basis. A discussion followed.	143-151 Slides 10-12
Mr. Tim Drzewieki, NRO, presented staff review of Section 15.1, Increase in Heat Removal by the Secondary System, the AOOs and PA , the evaluated AOO, codes used by the applicant, staff's confirmatory analysis, input parameters, and results. Upon member March-Leuba's question a discussion on linear heat generation rate (LHGR) specified acceptable fuel design limit (SAFDL) followed. Member Skillman wanted to know if a feedwater heater relief valve failure event could be more limiting than the	151-163 Slides 13-16

steam line safety valve or relief valve lifting. Mr. Drzewieki described staff review of steamline break. The possibility and consequence of the return to power was discussed.	
Mr. Raul Hernandez, NRO, presented Section 15.2, Decrease in Heat Removal by the Secondary System. He presented the events evaluated, limiting event, acceptance criteria and findings. Then he presented Section 15.2.8, Feedwater System Pipe Break, feed water line break being a PA.	163-166 Slides 17-18
Mr. Chris Van Wert, NRO, presented Section 15.3.1, Loss of Forced Reactor Flow. He discussed the evaluation model, staff confirmatory calculations, and staff review findings.	166-167 Slides 19-20
Dr. Peter Yarsky, RES, presented results of staff's TRACE/PARCS confirmatory analysis for the loss of flow event. Member Rempe wanted to know how the modeling was done for APR1400. A discussion followed. Dr. Yarsky presented the sequence of events, comparison of his TRACE/PARCS analyses and DCD results, and concluded the DCD analysis results were conservative.	167-183 Slides 21-29
Mr. Van Wert, NRO, presented staff review of Sections 15.3.3-15.3.4, Reactor Coolant Pump Malfunctions. Of the two events evaluated, the RCP rotor seizer was the limiting event. He discussed the evaluation model, inputs and assumptions, and review findings.	183-186 Slides 30-32
Dr. Lu presented staff review of Section 15.4.1 to 15.4.3, Reactivity and Power Distribution Anomalies, consisting of three events. He presented the events, staff audit of the analytical methods, and event analysis results.	186-190 Slides 33-35
Mr. Drzewiecki presented staff review of Section 15.4.4, Startup of Inactive RCP. Discussion on possible RCP startup in Modes 1 and 2 followed, and member Skillman observed that there was no provision in the TS to prevent that possibility. The staff will review the need for strengthening their SER.	190-195 Slide 36
Mr. Carl Thurston, NRO, presented staff review of Section 15.4.6, Inadvertent Decrease in Boron Concentration. He discussed the open items in staff SER.	195-199 Slide 37
Dr. Lu presented staff review of Section 15.4.7, Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position.	200 Slide 38
Mr. Drzewiecki presented staff review of Section 15.4.8, Control Element Assembly Ejection Accidents. He discussed staff review of the three analyses involved, methods uses, assumptions and initial conditions, and results. Then Dr. Yarsky presented the staff's confirmatory analysis method and results that concluded applicant's analysis was conservative. A long discussion ensued when member March-Leuba questioned the values of maximum ejected worth obtained using the staff's realistic method	200-214 Slides 39-43

and the applicant's analysis.	
Mr. Drzewiecki presented staff review of Section 15.5.1, Inadvertent ECCS Actuation, the three cases analyzed by the applicant, input parameters and results. Then he presented staff review of Section 15.5.2, CVCS Malfunction that Increases Inventory in RCS.	214-216 Slides 44-47
Mr. Drzewiecki presented staff review of Section 15.6.1, Inadvertent Opening of a Pressurizer Pressure Relief Valve, noting that in the SRP this event is an AOO, but the applicant evaluated it as a postulated accident, a SBLOCA. Then he presented staff review of Section 15.6.2, Failure of Small Lines Carrying Primary Coolant Outside Containment, very briefly.	216-217 Slides 48-49
Mr. Drzewiecki presented staff review of Section 15.6.3, Steam Generator Tube Rupture. Member Stetkar wanted to know why there was no steam generator overfill.	217-219 Slides 50-51
Mr. Jim Gilmer, NRO, presented staff review of Section 15.8, ATWS. Similarity with CE System 80+ design led to a long discussion on event termination, and the Chapter 7 open item regarding the diverse protection system (DPS). It was pointed out that because of this open item, Slide 52 conclusion that APR1400 design meets the ATWS rule was premature.	219-226 Slide 52
Mr. Gilmer presented staff review of Section 15.6.5, Large Break LOCA. He noted that staff review conclusion was pending the review of a topical report on the subject and related issues. He mentioned the significant issues including identification of an error in the input of moderator temperature coefficient reactivity table in the RELAP codes used for LBLOCA.	226-232 Slide 53
Syed Haider, NRO, presented staff review of Section 15.6.5, Small Break LOCA. He noted there was a technical report on the subject and that for PWRs the most challenging transient for peak cladding temperature was always in limiting SBLOCA due to loop seal formation and the potential core uncovery. He noted the scope of his presentation, methodology and computer codes used, APR1400 RC loop conceptual design, effect of the loop seal, and the conservatism in applicant's analyses. Discussion on liquid level in the reactor core and bypass flow took place. Validation of CEFLASH code using data from Semi scale test SUT-8 was discussed. Member question regarding the impact of the design differences between Westinghouse and CE reactor on this validation was addressed. Loop seal clearing was another question addressed. Staff review of SBLOCA break spectrum analysis, staff's TRACE confirmatory calculations, and conclusion of conservatism in applicant's analyses were presented.	233-261 Slides 54-58
Dr. Joe Staudenmeier, RES, presented staff's large and small break LOCA confirmatory analysis. He described the APR1400 TRACE model staff	261-279 Slides 59-69

used for confirmatory analyses, results for best estimate LBLOCA and SBLOCA break spectrum, comparison with applicant's results, and results for the fuel thermal conductivity degradation sensitivity study. Discussion on how pipe size and geometry could affect loop clearing took place. He presented comparison of TRACE calculation and KHNP LBLOCA and SBLOCA results and why there were differences. Dr. Staudenmeier presented staff's TRACE analysis of post LOCA long term cooling (LTC), LTC aspect of loop seal clearing (if a second PCT occurs), and thermal conductivity degradation.	
Mr. Dan Prelewicz, NRO, presented staff review of boron precipitation during long term cooling, noting a KHNP technical report that covers the boron precipitation analysis. He addressed the methodology (Interim Waterford Methodology), computer codes used, and changes made as a result of staff review. Mr. David Caraher, ISL (NRC consultant), addressed staff review of the long-term cooling boron dilution due to start-up of an RCP or reestablishment of natural circulation. He descried the staff questions and RAI response not yet docketed.	279-284 Slide 70-72
Mr. Caraher presented staff review of long-term loop seal clearing in a little more detail. If the cleared loop seal will reseal and produce a temperature increase was followed up by a KHNP calculation following a staff RAI. Mr. Caraher presented staff review of in-vessel downstream effects. He discussed the KHNP debris source analysis, and fiber only by-pass testing that were audited by the staff. The staff also audited the KHNP calculation for the available driving head across the debris bed, and inspected the KHNP fuel assembly head loss test facility watching test runs. The staff also reviewed the LOCA deposition model. The staff concluded due to use of metallic installation, containment cleaning program, and analytical margin, the staff did not have a concern. Member Stetkar brought out the containment over-pressure credit isitsue, and that could be handled by design, especially in new plants.	284-292 Slides 73-78
Dr. Lu concluded the presentation by mentioning the open items and that staff expected to resolve all issues before the end of the year.	292-294
Chairman Ballinger opened the public telephone line and asked for comments from the public attending the meeting. No comments were offered.	294
Members' closing comments: Member Rempe stated that another briefing on the LBLOCA methodology report would be useful. It was noted that an ACRS briefing on the topical report was being scheduled later in the year. She noted the staff's SER	295

was well done. Member March-Leuba applauded staff's ability to put together TRACE models of complexity in such a short time period. Member Powers noted that: 1) the dispersion of release around the main control room was done in an extraordinarily approximate method, and the staff needed to assure that a COL applicant would validate the analysis for their particular location; 2) regarding the sump pH calculation, the applicant did not mention the radiolytic production of carboxylic acids due to any organics in the water; 3) it was not apparent how the applicant took into account the depletion of the buffering capacity as the phosphates and borates reacted with various contaminants that would come into the sump as a result of accidents; and 4) how the applicant handled both the coulombic effects and short-range effects in the curves was not clear. Member Skillman noted regarding section 15.6.5.2, Large Break LOCA, the figures of the core water level and the downcomer level did not have the same datum, and should be fixed. Consultant Schultz noted the open items yet to be resolved are worthy of additional attention, and mentioned the overarching impact of fuel TCD degradation. Member Corradini emphasized the scheduling (of briefings) on the methodology with TCD being considered with the recalculation of some of the key accidents or transients.	
Chairman Ballinger adjourned the meeting at 5:30 p.m.	304

Following is a list of questions and comments the members raised for which a response was not available at the meeting:

MEMBER REQUESTS AND QUESTIONS	
Items	Reference Pages in Transcript
KHNP to follow up – Member March-Leuba's question if there is a technical specification limit on power asymmetry.	24
KHNP to respond if startup of an inactive RCP scenario should consider addition of positive reactivity to the core. Mr. Sung Ju Cho, KHNP, provided a response, but the transcript is illegible.	28, 128
Regarding Section 15.4.4, Startup of Inactive RCP, the staff will review the need for strengthening their SER.	194

REFERENCE:

1. Korea Electric Power Corporation and Korea Hydro & Nuclear Power Company, Ltd., “Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd Application for Design Certification of the APR1400 Standard Design,” December 23, 2014 (ML15006A098).
2. U.S. Nuclear Regulatory Commission, “Advanced Power Reactor 1400 Design Certification Application – Safety Evaluation with Open Items for Chapter 9
3. CD containing RAI responses.
4. Topical Report, APR1400-F-A-TR-12004-P, Rev. 0, Realistic Evaluation Methodology for Large-Break LOCA of the APR1400 (ML13023A081)
5. Technical Report, APR1400-F-A-NR-14001-P, Rev. 0, Small Break LOCA Evaluation Model (ML15012A025)
6. APR1400-F-A-NR-14002-P, The Effect of Thermal Conductivity Degradation on APR1400 Design and Safety Analyses (ML15012A026)
7. APR1400-F-A-NR-14003-P, Post-LOCA Long Term Cooling Evaluation Model (ML15012A019)
8. APR1400-Z-A-NR-14014-P, Rev. 0, ATWS Evaluation (ML15128A280)

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
APR1400 Subcommittee Meeting

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Friday, May 19, 2017

Work Order No.: NRC-3079 Pages 1-311

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

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

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

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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APR1400 SUBCOMMITTEE

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FRIDAY

MAY 19, 2017

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

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

COMMITTEE MEMBERS:

RON BALLINGER, Chair

CHARLES H. BROWN, JR.

MICHAEL CORRADINI (via telephone)

JOSE A. MARCH-LEUBA

DANA A. POWERS

JOY REMPE

STEPHEN SCHULTZ

GORDON R. SKILLMAN

JOHN W. STETKAR

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER BROWN

ALSO PRESENT:

DOUGLAS BARBER, Consultant

DAVE CARAHER, ISL

WOOCHONG CHOU, KHNP

SUNG JU CHO

TIM DRZEWIECKI, NRO

JIM GILMER, NRO

SYED HAIDER, NRO

MICHELLE HART, NRO

RAUL HERNANDEZ, NRO

JAEHOON JEONG

UNG SOO KIM

YOUGGUN KIM, KHNP and KEPCO

DONGSU LEE, KHNP and KEPCO

ROBERT LEE*, Westinghouse

KAEYEOL LEW

SHANLAI LU, NRO

ANDY OH, KHNP

DAN PRELEWICZ, NRO

JIM SERVACIOUS, Consultant

ROB SISK, WEC

JOE STAUDENMEIER, NRO JOHN

STECKEL, NRO

CARL THURSTON, NRO

CHRIS VAN WERT, NRO PETER

YARSKY, RES

Maitri Banerjee, ACRS Staff*

* Over telephone

AGENDA

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

(8:30 a.m.)

CHAIR BALLINGER: Our recorder is not here so the recordings are being made by Theron in the booth, which means that when you speak, you really need to, including me, need to be cognizant of pushing the little button to make the green light come on. Because otherwise he won't hear you.

All right. We have additional people present here that weren't here yesterday. Not the least of which is Joy Rempe.

And we have multiple people on the phone. The public line is open and it's muted. Contractors, there are two contractors that are on the line, I hope. Are they there?

Jim Servacious or Doug Barber? Well, they're supposed to be there. And Mike, Member Corradini is on the line. And he's the only one that will remain un-muted. Everybody else should be on mute. Maitri is also on the line and her phone should be muted.

So that's the procedure for today. One more thing, be sure to disable all the noise making machines that are in the room.

And I think we can pick up from yesterday.

Does anybody, you want to make any comment?

MEMBER WARD: Yes. This is Bill Ward talking. I just wanted to thank you for yesterday. I understand it went pretty well.

And I hope that today is another successful day. Only three more after this to get through Phase 3. And we're looking forward to getting through this rapidly today. Thank you.

CHAIR BALLINGER: Let's hope today we don't pull a Brett Favre, go long.

MEMBER WARD: No, not today.

CHAIR BALLINGER: Okay. So with that being said, the floor is yours.

MR. SISK: Thank you. Rob Sisk, Westinghouse. I have no opening comments, so I'm going to turn it over immediately to Mr. Ung Soo Kim to lead us through the Chapter 15.

MR. U. KIM: Yes.

MR. SISK: Okay, thank you.

MEMBER STETKAR: It's probably good, when you speak, to identify yourself, because on the recording we do not have the benefit of our reporter knowing who is speaking. So just until the reporter gets here, just identify yourself when you start to speak. And that was John Stetkar saying that.

MR. U. KIM: Okay. Good morning everyone. I am Ung Soo Kim. I am working in safety analysis department at KEPCO E&C.

This presentation is for the DCD Chapter 15, Non-LOCA and the LOCA analysis for APR1400 design. Next.

(Off microphone comments.)

MR. U. KIM: Okay. The contents are provided on this slide. Especially, the summary of the radiological consequences will be presented in Chapter 15A.

Here we see on overview of the section with the DCD Chapter 15. Today four people will present this chapter.

First, I present Non-LOCA analysis from Section 15.1 to 15.6. Then Dr. Chon Woochong will present LOCA analysis, Subsection 15.6.5.

Then Mr. Kim Youggun will present about long-term cooling.

Finally, remaining sections for radiological consequence will be presented by Mr. Lee Dongsu.

The following documents are submitted for DCD Chapter 15.

Now, I'm going to talk about Section 15.1, increase in heat removal by the secondary system. This section handles four anticipated operational occurrences and one postulated accident, as in this slide.

Among four AOOs, the inadvertent opening of a steam generator relief or a safety valve is quantitatively analyzed as the limiting AOO. Also, the postulated accident steamline break is analyzed.

In this slide, the red mark part indicates quantitatively analyzed events. And will be explained.

Inadvertent opening of a steam generator relief or safety valve. Due to the opening of a steam generator relief valve, the main steam flow increased, but by no more than 11 percent of nominal full-power steam flow rate.

This makes a decrease in core inlet temperature and consequently resulting in core power increase, by temperature feedback effect. So in this event, DNBR decreased and the major concerned parameter is the minimum DNBR.

In this analysis, as a single failure reactor trip override mode of FWCS, failure is assumed. Because this makes feedwater not decreased after the reactor trip and sustains cooldown by feedwater.

MEMBER MARCH-LEUBA: Give me a second. This is Jose March-Leuba. I see in the figure there is a scram at about, what, 1,500 seconds after. So there is no automatic scram for this system? For this scenario.

MR. U. KIM: Pardon?

MEMBER MARCH-LEUBA: Is there an automatic reactor scram for this scenario? Or is it manual? Why is it --

MR. U. KIM: It's manual. Manual reactor trip.

MEMBER MARCH-LEUBA: It's manual?

MR. U. KIM: Yes.

MEMBER MARCH-LEUBA: So the high power is never reached?

MR. U. KIM: Never reached.

MEMBER MARCH-LEUBA: What is the high-power scram?

MR. U. KIM: Reactor power trip. About 116 percent.

MEMBER MARCH-LEUBA: Sixteen.

MR. U. KIM: Yes.

MEMBER MARCH-LEUBA: So the power is less than 116?

MR. U. KIM: Yes.

PARTICIPANT: So, the 11 percent increase, that's because of a restrictor in the steam generator outlet line, is that it?

MR. U. KIM: The general percent is the general climate of the --

MR. JEONG: Oh, okay.

MR. U. KIM: -- installed in the steamline.

Next, steamline break. What steamline break result in, excessive RCS cooldown and makes the core reactivity to increase. Degradation in the fuel cladding performance may occur from this event.

So SLB analysis case are chosen, in two aspects. That is, to maximize potential to post-trip return power, and the second is, to maximize potential for degradation in fuel cladding performance.

For SLB analysis, main steam isolator valve or safety injector pump failure are considered as a single failure.

From the analysis result, it is confirmed that post-trip return power does not occur. And the minimum DNBR remains above the fuel design limit.

Now, I'm going to talk about the Section 15.2, decrease in heat removal by the secondary system. As you see, there are a total of seven AOOs and the one post-trip accident, in this section. Among --

MEMBER MARCH-LEUBA: Sorry, can you go back to Slide 8? Okay. I'm looking at the reactivity. Jose March-Leuba.

I'm looking at the left plot, the reactivity.

MR. U. KIM: Reactivity, yes.

MEMBER MARCH-LEUBA: So I see that you inserted rodes instantly, I mean, within probably ten seconds, and then you have a reactivity of minus nine. But then later, as it's cooling down and you essentially get to see, you follow the total line? I mean, do you not go re-criticality, but you are very close? I mean, you say that post-trip RTP does not occur?

MR. U. KIM: Yes. Does not occur.

MEMBER MARCH-LEUBA: But, with uncertainties --

MR. U. KIM: Yes.

MEMBER MARCH-LEUBA: -- if you put a little bit, it may have occurred.

MR. U. KIM: Yes, it is uncertain.

MEMBER MARCH-LEUBA: Is this a full scram? I mean, is it an assumed failure of the rods?

MR. U. KIM: Full scram.

MEMBER MARCH-LEUBA: Full scram?

MR. U. KIM: Yes.

MEMBER MARCH-LEUBA: So really, to keep the reactor shutdown, you need boron, otherwise you will have gone, without the boron, you will have not maintained shutdown conditions?

MR. U. KIM: Boron is injected by safety injection. In this figure, this line is boron injection by safety injection system.

MEMBER MARCH-LEUBA: But the boron safety injection is the one that keeps you from going by critical, is correct?

MR. U. KIM: Yes, subcritical for maintaining.

MEMBER MARCH-LEUBA: Okay, thank you.

CHAIR BALLINGER: Are we sure we have a clear answer on that?

MEMBER MARCH-LEUBA: I think I understand it.

MR. U. KIM: Okay, I will continue. For the Section 15.2.

Among all AOOs in this section, the loss of condenser vacuum is most limiting. And quantitatively analyzed, and other AOOs, are bounded by this LOCV event.

Though postulated accident in this section, feedwater line break, is also analyzed too.

Loss of condenser vacuum. When a LOCV occurs, an immediate feedwater termination and turbine trip, making a complete reduction in steam flow, conservatively assumed to occur.

Therefore, abrupt reduction in heat transfer from the RCS to secondary system occurs. This fast decrease in RCA cooldown rapidly pressurizes the RCS. So system peak pressure is concerned in this event.

The analysis results show that the RCS and main steam system pressure increased. But they are below acceptance criteria.

Next, the feedwater line break. A feedwater line break makes rapid depletion of affected steam generator liquid mass.

This reduce heat transfer capability between the RCS and the secondary system. So rapid RCS heat up and pressurization occurs. And the system peak pressure is major concerned parameter.

In analysis, the break size of the feedwater line is determined by sensitivity analysis, in order to get limiting one.

Through the analysis, we identified that the RCS and main steam system pressure increase. But they are below acceptance criteria.

Next, the Section 15.3. DCD Section 15.3 consist of one AOO and two postulated accidents. Flow controller malfunction is categorized as a BWR event. So it is not applicable in the APR1400.

Loss of forced reactor coolant flow. A complete loses of forced reactor coolant flow event is more limiting than any partial loss of forced reactor coolant flow. Because the reactor trip, the reactor will trip at the same time for both cases.

As a result of simultaneous loss of electrical power to all RCPs, a complete loss of forced reactor coolant flow event occurs. A reduction of coolant flow causes an increase in core average temperature, system pressure and decreasing in margin to DNB.

Turbine trip and loss of feedwater are assumed to occur at the same time with this event. And the most adverse combinations of initial condition for each aspect are determined by the parametric studies.

NRC approved of the computer codes, COAST, HERMITE, CETOP and CESEC-III are used for event analysis. As a result of the analysis, it is confirmed that all event acceptance criteria were satisfied.

Reactor coolant pump rotor seizure and reactor coolant pump shaft break. This event can be caused by the mechanical failure of the RCP. Or RCP shaft break.

This event has similar system behavior. And the flow coastdown for the RCP locked rotor event is faster than the coastdown for the shaft to break. Therefore, the RCP locked rotor event is most limiting than RCP shaft break.

A reduction of coolant flow causes an increase in core average temperature, system pressure and decrease in margin to DNB.

Loss of feedwater flow and remaining RCP flow coastdown are assumed to occur at the same time with the loss of offsite power occurrence.

Most adverse combinations of initial conditions for each aspect are determined by the parametric studies. As a result, it was confirmed that all event acceptance criteria were satisfied.

The DCD section is Chapter, DCD Section 15.4, consists of six AOOs and one PA. As shown in this slide.

Subsection 15.4.5 is categorized as a BWR event. So it is not applicable for APR1400 design.

Uncontrolled CEA withdrawal from a

subcritical or low-power startup condition. Uncontrolled withdraw of a CEA is assumed to occur as a result of single failure in the control element drive mechanism, CEDM control system, reactor regulating system or as a result of operator error.

The withdrawal of CEAs from subcritical or low-power conditions add to the reactivity to the reactor core. Causing the core power level and the core heat flux to increase, with the corresponding increase in the reactor coolant temperatures and reactor coolant system pressure.

0.001 percent of rated power is chosen as initial power level. Which is high logarithmic power reactor trip bypass set point.

To maximize the reactivity insertion rate, the maximum CEA withdrawal rate and maximum differential control CEA bank worth are assumed, for this analysis. And the most limiting initial conditions are selected.

As a result of analysis, it is confirmed that all event acceptance criteria were satisfied.

MEMBER MARCH-LEUBA: This is Jose again. I have a couple of questions on this. First, you assume the maximum reactivity injection rate for the rod that has more weight, more worth, or do you run the calculation with multiple rods and see which one is worse?

MR. U. KIM: Mr. Chon, can you answer this question?

MR. OH: Yes. This is Andy Oh from the KNHP Washington Office. Member Jose, could you question again?

MEMBER MARCH-LEUBA: When you perform this calculation, do you assume a particular rate of reactivity injection or do you inject multiple rods and see which one is worse?

Do you perform a full 3-D calculation, injecting this rod, that rod, that rod and see which one is worse, or do you figure out which one has more worth, in terms of dollars, and inject that one only?

MR. JEONG: Okay. I am Jaehoon Jeong from KEPCO Nuclear Fuel Company. And we selected the most limiting reactivity insertion rate from the nuclear rule.

MEMBER MARCH-LEUBA: So you only perform one calculation for the most limiting rod, based on the steady state rod worth?

MR. JEONG: Yes. Based on the sensitivity study analysis.

MEMBER MARCH-LEUBA: Yes. Okay. And

now, following up. Select the most limiting initial conditions.

The reactor has shutdown, how do you peak initial conditions? I mean, how do you get the most limiting ones?

MR. JEONG: We selected the ten to the minus three percent power at the actual initial power level. Because that power level is high level engine power trip to the reactor bypass set point.

So for this event to occur below that reactor power level --

MEMBER MARCH-LEUBA: Yes.

MR. JEONG: -- then this event will terminate by the high power reactor trip. So as you know, the hydraulic reactor power trip set point is 0.05 percent power levels.

And then if this event were higher than that engine power level, however, then that high reactor power doesn't occur. So it cannot occur.

In that case, variable overpower trip will be occurred to terminate this event. So variable overpower trip set point is about 14 percent.

MEMBER MARCH-LEUBA: Fourteen percent?

MR. JEONG: Yes, 14 percent.

MEMBER STETKAR: I'm sorry, I've heard

that twice. This is John Stetkar. I've heard that twice.

And in the tabulations, in both Chapter 7 and Chapter 15, the variable overpower set point trip is 103.5 percent. So why are you using 114 or 16 percent or whatever you said?

MR. JEONG: Oh. In this analysis, the variable or step, we assume the step variable overpower trip. Yes.

MEMBER STETKAR: Okay. So you're not using the variable overpower trip in this analysis, you're only using the step?

MR. JEONG: Step. Yes.

MEMBER STETKAR: Okay.

MR. JEONG: Variable overpower trip consists of step rate and --

MEMBER STETKAR: Plus a rate. Right.

MEMBER MARCH-LEUBA: Completely different question. Twenty kilowatts per foot. The last bullet.

Is that SAFDL, a specified acceptable fuel design limit or --

MR. JEONG: Yes. That is a melting limit.

MEMBER MARCH-LEUBA: Say again?

MR. JEONG: Fuel melting limit.

Calculated by the fuel rod design group.

CHAIR BALLINGER: This is Ron Ballinger. Is that accounting for burn-up dependent conductivity?

MR. JEONG: That definitely, fuel rod design would consider about the burn-up dependent.

CHAIR BALLINGER: Okay.

MEMBER MARCH-LEUBA: Yes, because 20 kilowatt, this is Jose --

CHAIR BALLINGER: That's pretty high.

MEMBER MARCH-LEUBA: -- 20 kilowatts per feet is very high.

CHAIR BALLINGER: It's pretty high.

MEMBER MARCH-LEUBA: But typically, in the reactors I'm more used to, which is BWR, we don't have an LHGR limit for transients. It's more of a steady state condition to prevent LOCA problems. Yes.

During the transients, you don't worry about the LHGR. So this is an acceptable limit that has been approved in the past.

I mean, what's the basis for this 20? I mean, it's a very round number. I mean, it looks like an arbitrary number.

MR. JEONG: Actually, this limit came from the fuel design group. And they consider about the burn-up dependent, the burn-up dependent in this limit.

When they set this limit.

And maybe, as you want to proceed, maybe the melting limit, LHR limit, maybe it decreased less than the 20 kilowatts could hit. But they consider about the peak power reduction over there.

So they concluded that the 20 kilowatts per foot limit is acceptable to apply the --

CHAIR BALLINGER: This is Ron Ballinger again. I'm still questioning whether 20 kilowatts per foot is the right number, when you account for burn-up dependent conductivity decrease.

So it is a nice round number, just my gut feeling, Steve Schultz might be able to say something, but my gut feeling is that that's too high.

MEMBER SCHULTZ: We can talk to the staff about this, this afternoon and see about their evaluation associated with it, but to pick the number, it ought to consider type of cycle as well as other features. Because low-power condition can happen at any time in cycle.

CHAIR BALLINGER: Yes.

MEMBER SCHULTZ: There will be a burn-up effect on the overall result. The basis for selecting 20 kilowatts per foot ought to be better known.

MEMBER MARCH-LEUBA: And being so high, my intuition tells me that you are going to hit the DNBR limit, way before you hit the LHGR limit. So it's probably relevant, but if we are using it, we need to know why.

MR. U. KIM: I'll continue. Next is, uncontrolled CEA withdrawal at power.

The cause of occurrence and transient phenomenon are similar with former DCD Subsection 15.4.1. One hundred and two percent of core power is assumed as an initial power level.

To maximize the reactivity insertion rate, the maximum CEA withdrawal rate and maximum differential control CEA bank worth are assumed. And the most limiting initial conditions are selected.

As a result of analysis, it is confirmed that all event acceptance criteria were satisfied.

Next is, CEA assembly misoperation. Dropped CEA or CEA subgroup, statically misaligned the CEA and single CEA withdrawal are included in this event.

Four-finger single CEA drop is the most limiting case. Regarding to the required thermal margin, among these cases.

A single CEA drop result from an interruption in the electrical power to the CEDM housing code or single CEA.

The dropped power begins to increase because of the negative MTC feedback effect. And eventually returning to the initial power level.

The hot pin radial peaking factor starts to increase, because of the dropped rod and xenon redistribution effect.

One hundred and two percent of core power is assumed for analysis as the initial power level. Maximum radial peak distortion is considered and the most limiting initial conditions are selected.

As a result of the analysis, it is confirmed that all event acceptance criteria were satisfied.

MEMBER MARCH-LEUBA: Wait. Jose again. Radial peak distortion, did I hear you say that you consider xenon transients for radial? How do you distort the radial power?

MR. U. KIM: How about Mr. Jeong?

MR. JEONG: This is Jaehoon Jeong. We also, not only consider the static distortion for both of the xenon redistribution factor --

MEMBER MARCH-LEUBA: So you --

MR. JEONG: -- analysis.

MEMBER MARCH-LEUBA: -- assume a xenon

imbalance?

MR. JEONG: Right.

MEMBER MARCH-LEUBA: Okay. And that's, again, the most conservative limiting initial conditions comes along. I mean, you just put one that is bounded by experience?

Because in principle, the xenon imbalance is not bounded physically. You will have something like this or like this. So it's based on operating experience or what?

Or maybe you have a tech spec limit on asymmetry?

MR. JEONG: This is Jaehoon Jeong KEPO. Actually, the Non-LOCA analysis field casts the most limiting transients from nuclear degeneration. And we believe that the nuclear degeneration rate determines the most limiting case.

MEMBER MARCH-LEUBA: Is there a technical specification limit on power asymmetry?

MR. JEONG: Actually, I am not sure. I'd be happy to check that.

MEMBER MARCH-LEUBA: Can you come back on that?

MR. JEONG: Okay.

MEMBER MARCH-LEUBA: And I'm sure, since

you're going to go ask questions, the peak linear generation rate, the LHGR, how is that defined?

Is that the heat flux coming out of the clad, is the power generation inside the pellet, it's instantaneous or integrated?

You understand the question?

MR. JEONG: I understand. This is Jaehoon Jeong. We calculate the linear heat generation rate based on the actually LHGR limit times nuclear power.

MEMBER MARCH-LEUBA: Nuclear power?

MR. JEONG: Yes. We use nuclear power.

MEMBER MARCH-LEUBA: So you're assuming the instantaneous nuclear power?

MR. JEONG: Yes.

MEMBER MARCH-LEUBA: Not the integrated heat flux?

MR. JEONG: Not integrated peak power being calculated.

MEMBER MARCH-LEUBA: Integrated peak power. So that's very conservative for a fast transient.

MR. JEONG: Right.

MEMBER MARCH-LEUBA: And this was Jose.

MR. U. KIM: Okay. I am Ung Soo Kim again. The next event is startup of an inactive RCP.

The startup of an inactive RCP can result in increase or decrease in core average temperature, depending on the primary and the secondary side temperature condition. And the coolant temperature change can result in an increase in core reactivity.

The startup on inactive RCP, during power operation, is not applicable. Because power operation with an inactive RCP is not allowed by the tech spec.

So this event is analyzed in Mode 3 through 6. And this event is analyzed with respect to potential loss of minimum required shutdown margin, by using isothermal temperature coefficient and maximum possible heat-up or cooldown temperature.

As a result of analysis, it was confirmed that a return to critical core condition does not occur.

MEMBER MARCH-LEUBA: For this transient, this is Jose again. For this transient, do we worry about a inhomogeneous boron concentrations?

Like you have been shutting down the reactor and putting boron in the vessel, but now my cold-leg has lower boron, and I start the pump and I flush the boron out. So temporarily, you can get an increase in reactivity.

Is that a possibly scenario? A inhomogeneous water concentration in the vessel.

There's more water in the vessel than in the cold-leg.

MR. CHO: I am Sung Ju. We use the isothermal temperature coefficient. And this value also came from the, calculated by the nuclear design guide.

MEMBER MARCH-LEUBA: Yes, but do you understand the question?

MR. CHO: Yes, I know.

MEMBER MARCH-LEUBA: You been pumping boron --

MR. CHO: Yes, yes. Yes, you know, you asking about the inhomogeneous boron concentration.

MEMBER MARCH-LEUBA: Yes.

MR. CHO: If the boron concentration is lower than the ICS and the boron, then there's possible dilution a little bit in the core.

MEMBER MARCH-LEUBA: Yes.

MR. CHO: So it might be, add more positive reactivity in the core. That is your question, right?

MEMBER MARCH-LEUBA: Correct. Yes, that is the question.

MR. CHO: But where -- it didn't need to be, check the nuclear design guide to consider about the case.

MEMBER MARCH-LEUBA: Yes. Maybe if the

event I'm postulating is not possible, because when you have an inhomogeneous water concentration, because you're pumping a lot of water into the vessel, is right after a shutdown where the pumps were running.

MR. CHO: Right. Yes.

MEMBER MARCH-LEUBA: And you only, if when your pumps are off you already have inhomogeneous the whole cycle, then it's not possible. But I would like for you to verify that that's the case.

MR. CHO: Yes.

MR. U. KIM: Okay, it's Kim again. Next is inadvertent decrease in boron concentration in the reactor coolant system.

The inadvertent decrease in reactor coolant boron concentration, may be caused by malfunction of the CVCS or improper operator action. This results in a positive reactivity addition to the core.

The maximum dilution flow rate, minimum RCS mixing volume, minimum shutdown margin, maximum critical boron concentration and minimum inverse boron worth are assumed for the conservative analysis.

As a result of the analysis, it was confirmed that sufficient time, more than 30 minutes, is a variable for the operator to take a corrective action, prior to loss of minimum required shutdown margin.

MEMBER MARCH-LEUBA: This is Jose again. Sorry to ask so many questions, I'm just curious. I'm not complaining, it's for my education.

This is a shutdown type of event, right? You're not on power?

The addition of boron at power or at zero-power?

MR. CHO: This is Sung Ju. We analyzed from Mode 1 through Mode 6.

MEMBER MARCH-LEUBA: Oh, so you analyzed both of them?

MR. CHO: Yes.

MEMBER MARCH-LEUBA: Okay. So while you're at power, the operator will have an immediate feedback because peak power will start to rise.

My question was going to be, if you're at zero-power, and there is no feedback, what additional clue the operator has in the control room, to take action in 30 minutes?

MR. CHO: Operator can recognize this event by the boron dilution system.

MEMBER MARCH-LEUBA: So it's accurate and fast enough to do that? So there will be an alarm on low water?

MR. CHO: Yes. We setup, it is separate by analyzing this event to allow the alarm too enough early.

MEMBER MARCH-LEUBA: And do you have an idea, you probably don't know, at what time the alarm comes? Does it come in two minutes, five minutes or in three hours? I mean --

MR. CHO: Oh, we, based on the operation detection time and total dilution time, we can find the minimum, how can I explain.

MEMBER MARCH-LEUBA: Let me rephrase that question. Is this 30 minutes in operation after the alarm occurs?

MR. CHO: Oh, right.

MEMBER MARCH-LEUBA: Okay.

MR. CHO: That's right. That's right, around 30 minute.

MEMBER MARCH-LEUBA: Okay, thank you.

MR. U. KIM: Okay. Ung Soo Kim again. I will continue. Inadvertent loading and operation of a fuel assembly in an improper position.

The inadvertent loading and operation of a fuel assembly in an improper position event is initiated by the interchanging two fuel assembly, in a core.

Depending on the enrichment difference between interchanged fuel assemblies, the core power distribution may be effected, either slightly or enough, so that core performance would be degraded.

For analysis, spectrum of misloading is considered. The ROCS code is used to calculate both nominal expected radial power distribution, and the radial power distribution resulting from misloading.

As a result of the analysis, peaking factor from this event would not increase more than that assumed in the CEA drop event. So the DNBR value for this event is greater than the DNBR limit.

MEMBER STETKAR: This is, let me, this is John Stetkar. Just for clarification. I'm trying to do things real-time here.

You mentioned a boron dilution alarm. Since this is a public meeting, I'd like some clarity on what is the boron dilution monitoring system, since it has an alarm.

MR. CHO: I'm Sung Ju. We have two independent boron dilution alarm systems. A system adapted for the alarm.

MEMBER STETKAR: What are those independent boron dilution alarm systems?

In particular, I'm reading your technical specifications, 3.3.14 under boron dilution alarms, and it takes credit for startup neutron flux. I don't see independent boron dilution alarm systems here. Is that your boron dilution alarm?

MR. CHO: Yes.

MEMBER STETKAR: Okay. So it's the two start up --

MR. CHO: Systems.

MEMBER STETKAR: -- flux channels.

MEMBER MARCH-LEUBA: Okay, so it's a --

MEMBER STETKAR: You get a high neutron, you know, you have a startup --

MEMBER MARCH-LEUBA: Okay. So this is a neutronic --

MEMBER STETKAR: It's a neutronic. That's what I wanted to clarify that there is something that it's real-time monitoring --

MEMBER MARCH-LEUBA: Boron --

MEMBER STETKAR: -- boron concentration.

MEMBER MARCH-LEUBA: I am not --

MEMBER STETKAR: There are --

MEMBER MARCH-LEUBA: -- PWR guy.

MEMBER STETKAR: There are other designs, that I have seen, that have different methods for detecting boron dilution. And I don't want to talk about those because they're different designs.

As best as I can tell, this design is a rather standard reliance on startup flux. Unless I'm missing something. If I'm missing something, please clarify it.

MR. OH: That's correct. That the startup --

MEMBER STETKAR: Andy, identify yourself, just for the record.

MR. OH: Yes. This is Andy Oh, KNHP Washington Office. And I think, Member Stetkar, your understanding is correct.

MEMBER STETKAR: Okay.

MR. OH: That startup power, to have two independent channels --

MEMBER STETKAR: Right.

MR. OH: -- that generated boron dilution alarm.

MEMBER STETKAR: That's fairly standard for a lot of plant designs. I just wanted clarity on the record so that we understood how the operators would detect that condition.

MEMBER MARCH-LEUBA: Jose here. When you are at power then, what is the alarm based on, just high-power?

MR. CHO: At power condition, yes. Definitely there is some many other, our system can be detected. One is high --

MEMBER MARCH-LEUBA: And this is slow enough that the operator will see it even before the alarm happens. So I'm not really concerned about that power, it's more the zero-power condition. Thank you.

MR. U. KIM: I'll continue. This is Kim again. Spectrum of CEA assembly ejection.

A CEA ejection event is postulated to occur as a result of a mechanical failure of the CEDM housing or its associated nozzle. The CEA ejection adds positive reactivity to the core, which results in a rapid power increase for a short period of time.

This power excursion is terminated by the combinations of delayed neutron and Doppler feedback effect, and finally by the reactor trip.

For the conservative analysis, maximum ejected rod worth, minimum effective delayed neutron fraction and minimum Doppler coefficient, are assumed for this analysis. NRC approved code STRIKIN-II, CETOP and CESEC-III codes are used for this accident analysis.

As a result of the analysis, it is confirmed that all event acceptance criteria were satisfied.

Now I'm going to talk about --

MEMBER SCHULTZ: Excuse me, Steve Schultz. Before we leave the control element assembly ejection, this is probably the best place, or one place, to bring up thermal conductivity degradation. Because there has been an open item associated with that, related to this event. But that topic comes up many times in discussion.

My question, I know that you've been working to address that here and you've provided some results to demonstrate the change in margin to limits, with taking into account thermal conductivity degradation, but I have a larger question.

And that is, I notice you're identifying, non-appropriately, that the codes that are being used here are NRC approved codes. In the transient evaluation.

Thermal conductivity degradation applies to both the steady state codes, of course, and also transient codes. And this has been an issue that's been in the U.S. and international realm for quite some time.

Many years we've been talking about thermal conductivity degradation, and it seems to me that as we look to develop this new reactor design, an issue that is this pronounced, in terms of affecting the fuel performance over the course of its burn-up range, ought to be addressed thoroughly. Not just in steady state evaluations by the fuel performance group, but by those that use thermal conductivity, the codes that use thermal conductivity, in their transient evaluations as well.

Now, what happened in the U.S. industry and other industry, nuclear industry, is that considerations were given to thermal conductivity degradation. And different analyses that had been done are considered, to see if there was an effect, how much the effect was, how much a reduction in margin there would be if the evaluation were done with this new tool, this new calculation of thermal conductivity.

And all that was taken into account by the NRC. They had approved codes, they didn't modify those codes, they didn't re-approve the codes.

But it just seems to me that as we move forward with a new application, for a reactor that's going to be in operation for 60 years or 70 years hence, an issue that's this distinct, and accepted as a burn-up dependence and thermal conductivity that wasn't understood 20 years ago, is understood, it ought to really be addressed.

Not just in the steady state thermal performance, but also integrating that into the transient evaluation. So we know that the, what the real margin to limits are. Not just for one event, for this particular one.

But how does it just assure that it is incorporated in the effects associated with small break LOCA, large break LOCA, other transient analyses.

Do you have, can you give me an appreciation for kind of the modeling philosophy, both steady state and transient, where you, are you embracing thermal conductivity degradation or are you just kind of addressing it the way it was done in the last five or six years and calling it quits?

MR. JEONG: I am Jaehoon Jeong. NRC and the KHNP have had a lot of discussions for TCD. And actually, our trip performance code does not monitor the TCD effects.

So we had discussions with the steps and we almost conclude that we're going to add some penalties considered TCD. So that penalty will be applied to all safety emergencies, effected by TCD. Not only LOCA emergencies, but (unintelligible).

We are (unintelligible) with TCD (unintelligible) and we will revise our technical reports and topical reports and TCDs.

MEMBER SCHULTZ: Do you have a longer-term plan to address it in fuel performance and transient analysis modeling, within the codes that you are in the process of developing for analysis?

It bothers me that this, for a while it was reasonable to say, well, we'll apply a penalty because we didn't get it right.

MR. JEONG: Right.

MEMBER SCHULTZ: But we've been doing this for the last ten years almost, and it's time to, when you're coming up with a new design, new application, the codes ought to be representative of our current state of knowledge. And just to say, well, we'll apply penalties that that will be it, I think it's insufficient.

MR. JEONG: Jaehoon Jeong again. Yes, I agree with you. And actually, we have a code, which TCD model is applied. We have.

But when we submit this TCD, that code has not been finished. But now we have finished. So next time, maybe, we have a chance to revise our topical report that we will now apply the code.

MEMBER SCHULTZ: Thank you for the additional information.

CHAIR BALLINGER: Okay, this is Ron Ballinger. Let me be clear on this.

We're operating with DCD Revision 0, and are you saying that your models now incorporate burn-up dependent conductivity, and then in the future modification of the DCD revision, that will be included?

MR. JEONG: I am Jaehoon Jeong. No. I think that is a new code that currently we are applying to Page 3B. That is (unintelligible) code approved by NRC.

But we thought that it will take too long time to revise, I mean to incorporate the TCD effect, on that code. Instead of that, we developed a new code.

But in this, and currently, at this time, we don't have any plan to use the new code, because it will take too long time to license the new code. Because the stats never been seen in that code.

MEMBER REMPE: But I think, I've heard --

CHAIR BALLINGER: You have to tell us your name.

MEMBER REMPE: This is Joy Rempe. But I think what I heard you say, is that once you and the staff agree on what the penalty will be, you're going to redo all of the analyses, so we'll see all of the effects.

And that was the question I had too, because I'm having trouble tracking all the different places that we've been told, well, this will be effected by TCD.

So we'll have to be looking at new plots to compare the results sometime in the future here, right?

MEMBER STETKAR: Well, this is John Stetkar, to be clear, sometime in the future should be before the advisory committee on reactor safeguards, (unintelligible) find a letter on the certified design, for this APR1400. Not a decade from now.

MEMBER REMPE: So this is Joy again. And so yes. But that would be, we're going through, I always forget my tiers, but whatever tier we're going through now and reviewing it, they're interim letters. And so when we come back to review and write the final letter, we're going to have to go through and look at a lot of different plots. And that's what I was curious about too.

CHAIR BALLINGER: Yes, this is Ron Ballinger again. So on the record we're clear, this will come up at the staff, I'm sure, that when we write a final letter on the DCD, we will have had an opportunity to review the revised calculations that are impacted by burn-up dependent conductivity.

MR. SISK: This is Rob Sisk, let me kind of maybe be clear. TCD penalty is being incorporated into --

CHAIR BALLINGER: Yes.

MR. SISK: -- the design.

CHAIR BALLINGER: Okay.

MR. SISK: It is an open item. We're working with the staff to resolve the implementation of the TCD penalty and to all the various, I don't want to just say safety analyses, but throughout Chapter 4, Chapter 6. Many different places where TCD has potential ramifications.

So that is being completed, as we speak. And it will be completed as a part of the Phase 5, as we get to SER with no open items, those analyses will be completed.

That is not a new topical report or a new technical report. I guess a new DCD revision that will be. It is a completion of the analyses to support the DCD that's currently under review.

MEMBER REMPE: This is Joy again, and although I'm hearing yes, and seeing shaking heads on one side of the room saying, yes, you'll see updated curves, I see Member Stetkar saying, no, we won't see those plots. But what's the answer here, Member Stetkar, will we see --

MEMBER STETKAR: I don't know. This is John Stetkar. I don't know what we're going to see. My sense is that we're going to see some sort of ad hoc penalty.

MEMBER REMPE: But I'd like to see the calculations with that ad hoc penalty incorporated, for places of interest.

MEMBER STETKAR: I don't know what we're going to see.

MEMBER REMPE: But we can request that in our letter? Okay, thank you.

MR. U. KIM: Okay.

MEMBER POWERS: This is Dana Powers. You indicate that, for this analysis, the doses at the site boundary are below the allowable criteria. Those are the 10 CFR Part 100 criteria?

MR. U. KIM: How about 10 CFR?

MR. OH: This is Andy Oh, KNHP Washington Office. Member Dana Power, could you ask it again?

MEMBER POWERS: Well, I am asking, he says that the doses at the site boundary are below their allowable criteria limits. I'm asking, or the first question is, are those the 10 CFR Part 100 limits?

MR. OH: Part 100.

MEMBER POWERS: 0.5 rem at the site boundary. For the first two hours of the event.

MR. LEE: My name is Dongsu Lee. In our (unintelligible) analysis, as it relates to our value, is Victor 10 CFR 52.47, limitation. And it's SRP 15.0.3.

MEMBER POWERS: Forty-five rem at the site boundary. Yes.

The question is, is that -- also, is it true that the dose limits in the control room are below the criteria?

MR. LEE: You're right. The TDC, I'm Dongus Lee, (unintelligible) calculated the ability to, based on the TDC 19. Yes.

MEMBER POWERS: Why at the site boundary, why are the doses so low?

MR. LEE: Can you see show my presentation, Page 80.

MR. OH: I think we have some special session for discussing this item in a later part, can you talk --

MEMBER POWERS: That would be fine.

MR. OH: -- about it later?

MEMBER POWERS: If it's more convenient to talk about it there. It's just that at this peak, fuel, you're going to blow the fuel part in the affected region.

And the question is, are they dosed as low simply because a small amount of fuel is affected and consequently your release is small or is it because of natural and engineered processes that the release is small so your dose at the site boundary small?

MR. LEE: Yes. In our presentation, it's Page 71. Can you show that, our results?

And our CEA calculation at that time is that we consider, we did not take any credit of the depletion in the spray (unintelligible) that our result is shown in the table.

MEMBER SCHULTZ: This is Steve Schultz. Do you have any intermediate results regarding fuel failure or the condition of the fuel that was demonstrated by the limiting event?

MR. LEE: You mean the fuel handling accident?

MEMBER SCHULTZ: No, no, this is for the control --

MR. LEE: Oh --

(Simultaneously speaking)

MR. LEE: -- CEA ejection we used at the ten percent DNBR experience.

MEMBER MARCH-LEUBA: Can you repeat, this is Jose, can you repeat?

MR. LEE: Ten percent. Ten percent of --

MEMBER MARCH-LEUBA: Ten percent of the

fuel failed?

MR. LEE: Yes, failed.

MEMBER MARCH-LEUBA: Even though you had use over criteria, less than 230 no fuel melt, but you still assume that ten percent of the fuel failed anyway?

I mean, I think that the calculation says, I will survive the rejection without fuel failures. But then you assume ten percent failure just in case?

MR. LEE: Yes, right.

MEMBER MARCH-LEUBA: That's conservative.

MR. LEE: Yes.

MEMBER MARCH-LEUBA: While we're on that, can we go back to the original slide? Since I have the microphone and I don't have to identify myself.

I see here that the criteria has changed for this event. Now we say no fuel melting instead of 25 kilowatts per foot. Is that because at 20 kilowatts per foot was violated and you went to a more, less restrictive criteria? Or is it because somebody else did the calculation?

Do you understand the question? Okay, for all of our events, AOOs, we assume we want to keep 20 kilowatts per foot of peak power.

Here we don't judge this event by that criteria, we go a less restrictive criteria, which is not fuel melting. Is there a reason for it?

MR. JEONG: Actually, we calculated a kilowatt, this is Jaehoon Jeong, sorry. When we calculate the kilowatts per foot for the AOO, in the case, we use the nuclear power. So we apply the same methodology in CEA ejection.

And nuclear power definitely is a little high. So in that case we cannot mitigate the 20 kilowatts per foot.

MEMBER MARCH-LEUBA: So it is larger than, greater than 20 kilowatts per foot?

MR. JEONG: Definitely.

MEMBER MARCH-LEUBA: So you went to this less restrictive, more fuel dependent limit instead of the --

MR. JEONG: Right.

MR. U. KIM: Okay. I am Ung Soo Kim again. I will continue. Now I am going to talk about DCD Section 15.5, increase in reactor coolant inventory.

As you see, there are a total two AOOs in this section. Between these two AOOs, the CVCS malfunction, such as pressurizer level control system malfunction, is most limiting and quantitatively analyzed.

Chemical and volume control system

malfunction. The PLCS malfunction maximizes charging flow and minimize the letdown flow, so the RCS inventory is increased. And this increase in the RCS pressure until the reactor trip occurs.

By the way, because this pressure transient is due to RCS coolant inventory increase, not to thermal expansion, there is no significant power and coolant temperature transient, before reactor trip. The analysis result shows that the system pressure remains below acceptance criteria.

From now, I am going to talk about the Section 15.6, decrease in reactor coolant inventory. As non-LOCA event, there are one AOOs and the one postulated accident in this section.

Pressure relief valve is handled, oh I'm sorry, the evaluation of an inadvertent opening of a pressurizer pressure relief valve is handled in Subsection 15.6.5, presenting small break LOCA.

Letdown line break and steam generator tube rupture are quantitatively analyzed in this section.

Failure of small lines carrying primary coolant outside containment. The direct release of reactor coolant may result from a break or a leak outside the containment of a letdown line, instrument line or a sampling line.

And the double-ended break of the letdown line outside the containment is selected for the analysis in this section because it results in the largest release among them. The reactor coolant release also make RCS depressurization.

In analysis, it is conservatively assumed that operator take action to terminate the primary system fluid loss at 30 minutes after the event initiation.

From the analysis result, it is confirmed that the minimum DNBR remain above the fuel design limit and radiological acceptance criteria are satisfied.

Next. Steam generator tube failure. Steam generator tube rupture accident is penetration of the barrier between the RCS and the secondary system. This results in radiological release and RCS depressurization.

So, radiological consequence and minimum DNBR are majorly evaluated in this section.

For analysis, double-ended rupture over steam generator U-tube, at full-power condition, is assumed. And primary-to-secondary leakage and steam generator release mass are used as input to dose calculation.

Analysis results show that the minimum DNBR remain above the fuel design limit and the radiological acceptance criteria are satisfied.

From now, presentation for LOCA analysis go on by Dr. Chon Woochong.

MEMBER POWERS: Can you come back? This is Dana Powers again.

MR. U. KIM: Okay.

MEMBER POWERS: The release you get in this is just contaminated coolant?

MR. U. KIM: Pardon?

MEMBER POWERS: The radiological release you get here is just contaminated coolant?

MR. U. KIM: Just the fuel and iodine concentration in the RCS.

MEMBER POWERS: Just what's in a coolant that gets expelled?

MR. U. KIM: Hold on.

MEMBER POWERS: You're not damaging fuel here with a U-tube rupture?

MR. SISK: This is Rob Sisk. Can you speak up a little bit and restate your question?

MEMBER POWERS: What I'm asking is, what is the radiological release here? I believe it just to be contaminated coolant.

MR. U. KIM: Yes, contaminated coolant.

MEMBER POWERS: What is the assumed coolant concentration of radionuclides?

MR. LEE: My name is Dongsu Lee. When we calculated the consequence on the SGTR, at the time we considered an iodine spike effect based on the (unintelligible) condition.

MEMBER POWERS: And what did you assume for that spike?

MR. LEE: The PIS and the GIS. We considered both. The results shown in the table, previously I showed.

For instance, iodine spike and events generated a spike, are constant.

MEMBER POWERS: I'm still unclear what you assume for the spike. I assume you assume 500, but I --

MR. LEE: Yes.

MEMBER POWERS: -- don't know that that's the case.

MEMBER SKILLMAN: Wouldn't your assumption be your maximum, I'm Dick Skillman, wouldn't your assumption be your maximum dose equivalent iodine permitted by your tech specs?

MR. LEE: Can I show the presentation?

MR. U. KIM: Which page?

MR. LEE: Presentation, Page Number 68. 66, sorry. 66. Yes, the source term of the primary coolant and lubricants in iodines are assumed to exist with a tech spec LCO.

MEMBER SKILLMAN: Yes.

MR. LEE: Their number.

MEMBER SKILLMAN: That's the bottom. I'd marry your dose equivalents iodines with .1 microcuries per cc. That's your tech spec limit, and that would be entrance position for this accident.

MR. LEE: Right.

MEMBER POWERS: Thank you.

MEMBER SCHULTZ: Steven Schultz. What Member Powers is asking is with regard to the iodine spike concentration and what is it. Is this the pre-accident iodine --

MR. LEE: Yes, that the --

MEMBER SCHULTZ: -- and the event-generated iodine spike value.

MR. LEE: For the PIS, that we modified at the 60 times for the (unintelligible) and GIS case, event-generated iodine case at the time (unintelligible) Based on the appearance rate as we (unintelligible) three times for that. Yes, (unintelligible).

MEMBER SCHULTZ: Is it 335 is --

MR. LEE: I will check, again. I'm not sure that number, it really confused me.

MEMBER SCHULTZ: Thank you. I appreciate you checking.

MR. U. KIM: Okay, Ung Soo Kim again. From now, the presentation for LOCA will be conducted by Dr. Woochong Chon.

MR. CHON: Good morning. My name is Woochong Chon. You may remember green streetlight. Last time I explained the green streetlight is good signal for our projects.

In Korea, the different time difference between Korea and United State is 13 hours. So right now Korea is 10:41 p.m.

Yes, that kind of a big-time difference can make a more good progress in this project. During the day time, in Korea, my colleagues are working hard in Korea, and during the Korea nighttime, some of the engineers can work in the United States. So we are using 24 hours a day.

(Laughter)

MR. CHON: We can make big progress. And I hope this meeting is also, here, a part of good progress today.

MEMBER POWERS: Well, where does the Korean barbecue fit into this?

MR. CHON: The best Korean barbecue name is (unintelligible) I recommend that one.

Okay, I will start about the Subsection 15.6.5, LOCA resulting from spectrum of postulated piping breaks. It's kind of a real subsection in Chapter 15, but it includes large break LOCA, small break LOCA and post LOCA long-term cooling part.

And after that, the post LOCA long-term cooling downstream effect will be presented by another time.

In large break LOCA, the topical report realistic evaluation (unintelligible) for large break LOCA of APR1400, is under the review.

The CAREM, code accuracy based realistic evaluation model, is used for large break LOCA analysis.

The revision of topical report in DCD Section 15.6.5, large break LOCA, are going to reflect the thermal conductivity degradation.

And large break LOCA is applying BE methodology, and small break LOCA is applying Appendix K. And small break LOCA and long-term cooling analysis results will be compound, is compound, the satisfaction of acceptance criteria.

Okay, next slide explains about the acceptance criteria for ECCS for light water nuclear power reactor. It is 10 CFR 50.46.

This section refers Reg Guide 1.157, which is the BE calculation of ECCS performance. And also, Reg Guide 1.206 and NUREG-0800 and 1230. And finally, the NUREG-5249, which is CSAU, are applied to LOCA analysis.

Okay, this slide explains about brief description of large break LOCA. APR1400 safety injection system consists of four mechanically independent trains.

With four direct vessel injections, as shown in this figure. The injection diagonal, four DVI nozzle locations.

And one safety injection pump and one safety injection tank are installed in each train. So we have four SI tank and four SI pumps.

Both SIP and SIT flows are injected into the upper annulus through the DVI nozzle, as shown in this figure. DVI nozzle location is indicated in green.

MEMBER MARCH-LEUBA: While you have the figure, this is Jose. While you have the figure, later on the loop seal clearing, we're going to be concerned about bypass flows between the upper plenum and the downcomer. Can you point in there where they are?

There is a seal, metal seal, on the top upper plenum that we were told, that is assumed, (unintelligible) with three percent bypass flow, but is expected to be less than one. Can you point where it is in the figure, right there?

MR. CHON: It's not clear in this figure, but there is some gap between the top upper plenum, downcomer region.

MEMBER MARCH-LEUBA: Okay.

MR. CHON: There is a bypass region from the downcomer to the upper head.

MEMBER MARCH-LEUBA: And you're expected to be doing normal operation to be one percent leak flow? Or what do you expect the leak flow to be?

We were told in a different presentation that it was assumed to be three percent, but very conservative.

MR. CHON: Right. Yes. That's what I would assume. But in design spec, can you check --

MR. JEONG: Okay, I am Jaehoon Jeong. The upper head bypass flow rate is about .5 percent.

MEMBER MARCH-LEUBA: 0.5 percent? Still pretty large.

MR. JEONG: Yes, a little bit large.

MEMBER MARCH-LEUBA: Yes. And do we have any experimental relevance for that is that based on tolerances for manufacturing?

Because, this becomes critical when you assume the loop seal clears, or does not clear.

MR. JEONG: Actually, as I remember, we don't have experimental data for the bypass. But we assume the total bypass be about three percent.

MR. CHON: Right.

MR. JEONG: Including operator bypass.

MEMBER MARCH-LEUBA: Oh, so they all the bypass flow?

MR. JEONG: All bypass flow rate.

MEMBER MARCH-LEUBA: What about --

MR. JEONG: But the upper bypass will be about .5 percent.

MEMBER MARCH-LEUBA: Yes. And what are the other bypass?

MR. JEONG: I'm sorry, this is Jaehoon Jeong. The other bypass is, I mean core bypass flow rate will be three percent.

MEMBER MARCH-LEUBA: Yes.

MR. JEONG: This one is (unintelligible).

MEMBER MARCH-LEUBA: Oh, you're talking core bypass, correct?

MR. JEONG: Yes.

MEMBER MARCH-LEUBA: That is not upper plenum to downcomer?

MR. JEONG: No.

MEMBER MARCH-LEUBA: This is bypass in the core. Okay.

MR. JEONG: Right.

MEMBER MARCH-LEUBA: Thank you.

MR. CHON: Okay, this is Woochong Chon again. I will continue the next slide.

I think this figure is already shown in the topical report for fluidic device. The Rev 10 side figure shows the fluidic device installed inside of safety injection tank. SI tank injection fluid is one of the important factor in large break LOCA analysis.

Fluidic device makes a high flow rate and low flow rate. Two different flow rates. The duties of fluidic device in safety injection tank, are described in separate topical report of fluidic device design, which is approved by NRC couple months ago.

The right-hand side graph shows the schematic SIT mass flow rate, applied to large break LOCA analysis. With high flow and low flow region.

And next slide describes about the large break LOCA code, and methodology. RELAP5/Mod 3.3K calculate the thermal hydraulics part and CONTEMPT4/Mod5, calculate the containment back pressure calculation.

Those two-code exchange mass and energy and pressure, as a boundary conditions.

MEMBER REMPE: This is Joy Rempe. And I had a question that you don't have to answer now, but this setup with CONTEMPT and RELAP is how I assume you're taking credit for CAP, right? I mean, you use that back pressure to show that things are going to be okay.

When, and maybe I missed it, but when I was looking through the material for this, I never saw just a solid statement saying, we have to take credit for how many PSIG, for what duration of time.

And could I have some numbers to understand how much CAP credit is taken, at some point, in our interactions? If not today, later.

And maybe I missed it, maybe it is somewhere in Chapter 15, but I didn't see it anywhere explicitly stated.

MR. CHON: CAP?

MEMBER REMPE: Containment accident pressure. How much are you relying on the pressure in the containment? Am I saying this clear enough, what I am trying to ask for?

(Off microphone comment)

MEMBER REMPE: Okay, thank you. I just couldn't get a feel from what I was reading.

MR. CHON: Okay.

MEMBER REMPE: Thank you.

MR. CHON: Okay. The large break, this is Woochong Chon again. And the large break LOCA methodology, CAREM, is developed based on CSAU, which is NUREG-5249.

The uncertainties quantified by non-parametric statistics and 181 sample, (unintelligible) sampling calculations, are performed.

CAREM introduced experimental data covering process for confirmation of uncertainty parameters and their ranges and distributions.

Okay, this slide shows large break LOCA scenario specifications for APR1400. The X axis is time after break, and Y axis is water level. The solid black line is core water level, and red dot line is downcomer water level.

So, the CAREM divided by four stages of scenario. First one is blowdown and refill, early report and rate report.

The first part does decrease over pressure, is ended at this time. Which is, blowdown is ending.

And the refill is start until water level is reached, the core water level is reached, to the bottom of active core, at this region.

And then only the flood region is started, until SI injection is terminated. After that, rate report period will be continued.

MEMBER MARCH-LEUBA: This is Jose. Just a question. Is the black line, the core level, is that the collapse water level or is it two-phase water level? Is there any voids?

MR. CHON: This collects the water level.

MEMBER MARCH-LEUBA: Okay. So that -- and why is the downcomer level so much higher than the core level? Why doesn't the flow drop? I mean, this is natural circulation with very little flow, if any.

MR. CHON: This difference?

MEMBER MARCH-LEUBA: Yes. Why the difference in the elevation?

MR. CHON: The core pressure is much higher than downcomer part.

MEMBER MARCH-LEUBA: So that's because the loop seals are closed?

MEMBER CORRADINI: No, this is, if the loop seals are clear, the whole thing blows down.

MEMBER MARCH-LEUBA: Okay, so why is the

downcomer elevation six meters higher than the core?

MR. CHON: Okay.

MEMBER MARCH-LEUBA: Mike, you need to identify yourself.

MEMBER CORRADINI: I'm sorry, Corradini. There's no water left in the system after 20 seconds.

MR. CHON: I'm talking about the 150 seconds.

MR. JEONG: I am Jaehoon Jeong. During that period the core is still has not been quenched (phonetic) at that time. And that means that a lot of boiling occurs during the cool. So the core pressure is a bit higher compared to the downcomer pressure, okay. The high core pressure prevents the reflooding.

MEMBER MARCH-LEUBA: Okay. I'll, maybe I'll ask the staff in their complementaries if they will use these results.

MEMBER CORRADINI: So can I, this is Corradini, may I just make sure we're clear? So in the Time Region 3 the difference in elevation is about six meters, is that correct, between the downcomer and the core?

MR. CHON: Yes, it is.

MEMBER CORRADINI: So that's about .4 bars. So that's not a very large amount of pressure.

And if I have any sort of boiling I essentially will, the flow of the SIPs go out the break preferentially then going up through the core.

So I'm basically leaking fluid as soon as I fill the downcomer. And so then the rest of it is just driven by essentially what it can flow through to, to make up for the boiling process. So I assume that .4 bars is the pressure drop going through the core just because of boiling.

MR. CHON: If you see this figure the active core height is 3.81 meters. So that level is around here. So core is not covered. So still a lot of boiling occurred in the active core.

It create higher pressure. So that core levels differences came from the boiling from the active core.

MEMBER MARCH-LEUBA: Mike, this is Jose. Maybe you can help me here. But if the water level in the core is at three meters meaning there is no mass flow rate going out of the core, there is very little mass flow rate coming into the core it's just only sufficient to compensate for what they were boiling off.

And I don't see how you can have any pressures up there caused by friction.

MEMBER CORRADINI: Well I don't think it's the pressure drop so much as that you, your only place for the steam to go is to go through the generators and then out the break. So you've got a delta p, of about .4 bars.

MEMBER MARCH-LEUBA: What I think is happening here is that the pressure in the upper plenum is increasing, as the downcomer. And that happens when you have the loop seals closed.

But maybe the pressure loop is not the cause is the steam that caused the hot leg, okay. I'll ask the staff. Maybe they have some better, maybe they have looked at this.

MEMBER CORRADINI: I'm not sure, I guess the way to ask your question is you think the loop seals are refilled and I don't think they are. I think you've lost inventory. It's all going out the break.

But maybe in the process of going out the break you filled a couple of loop seals. That's what you're asking?

MEMBER MARCH-LEUBA: What I'm asking is I cannot conceive that the pressure drop across the core be a half a bar if there is no mass flow going through the core. There is no velocity. It's just pool boiling.

MEMBER CORRADINI: Yes, but you're

essentially at a few percent power so you can do the calculation. There are about 3,400 megawatts so that's something like 68 megawatts.

So that's something like about 30 or 40 kilograms a second of boiling. So that's a fairly substantial boiling rate by decay heat.

MEMBER MARCH-LEUBA: Okay. We'll do our numbers off line.

MR. CHON: Okay. This is Woochong Chon again. I move on to next slide. This right hand side figure shows the CAREM diagram. CAREM consists of three elements important that are effectively seen as CSAU.

However, Step 9 is the big, the major difference between CSAU and CAREM. Step 9 checks experimental data covering using the uncertainty parameters determined in Step 8.

If this Step 8 is, fails and if the data covering stages failed then Step 8 repeats until the covering is satisfied. Non-parametric statistics is used in experimental data covering as well as in plant calculations and detailed information is given in this red parenthesis (phonetic).

MEMBER MARCH-LEUBA: I am confused. What do you iterate on? You say if experimental uncertain parameters don't cover the result analysis, the results?

MR. CHON: In Step 8 we performed SET and IET code conversion here. And then in Step 9 we checked all the CAREM calculation results cover experimental data or not.

So if it's not covered then go back to Step 8 and change certain parameters or --

MEMBER MARCH-LEUBA: Do you perform more experimental data or --

MR. CHON: No, we put APR1400 we performed some special experiments. But for this case we apply the current experimental datas, not specific ones.

MEMBER MARCH-LEUBA: On this iteration what do you adjust on every step?

MR. CHON: We have total 29 uncertainty parameter ranges. We have adjusted the ranges or if there is data covering we also modify the model. It is a little bit different really because of its direction to cover experimental data. Those details are discussed in topical report.

MEMBER STETKAR: That's what I wanted to clarify. You have submitted this methodology as a topical report for the staff to review. Is that correct?

MR. CHON: Yes.

MEMBER STETKAR: We haven't seen it. Well we have a copy of the REV0, the topical report. We'll ask the staff about the review because some of these details I suspect are in that topical report and are probably better to discuss, you know, when we see it.

MEMBER CORRADINI: Yes, this is Corradini. I think if that was John I think the topical which is a very small 800 pages.

MEMBER STETKAR: 847 I think if my count --

MEMBER CORRADINI: I stopped reading. But I think staff actually has a couple of open items that they're going to come to if I, I pre-read some of the staff stuff on this.

MEMBER STETKAR: Yes. We may as a Subcommittee, and it's up to the Subcommittee and the staff, want to have a separate briefing on that since it is a topical report. If it were a technical report it would be under the purview of this chapter, but not necessarily.

MR. CHON: Okay. This is Woochong Chon again. Let's move on to the next slide. This slide shows the noding diagram which is applied in the CAREM methodology. The active core is modeled with two hydraulic channels and 20 axial nodes. And that --

MEMBER CORRADINI: So this is Corradini. Just to be sure, I think I know the answer but I want to make sure, when you say two channels so you have a hot channel and an average channel?

MR. CHON: That's correct.

MEMBER CORRADINI: Okay, thank you.

MR. CHON: And also the outcome is modeled with six radial channels and ten axial nodes. And steam generator, we have two steam generator models here and one pressurizer model here. And two groups each aside intact loop and broken loop, two groups are modeled.

And as I said before, currently we are working on the revision of large break LOCA part with applying thermal conductivity degradation. But I briefly explained the general large break LOCA here.

One hundred percent double-ended guillotine break in pump discharge rate is selected as a limiting case. Once inside they show us the water level versus time. These figures pretty much show same as what I show in the scenario part.

The upper part is downcomer. There are six downcomer channels, so six lines are in here and one solid line is core level.

MEMBER CORRADINI: This is Corradini. Just so I understand, can you explain the logic of the six downcomer channels because I assumed there was four because you have four potential cold legs. But why six? Can you explain that please?

MR. CHON: We have four cold legs and two hot legs. So each leg is located in one channel.

MEMBER CORRADINI: But I thought you said six downcomer channels.

MR. CHON: Right. Six downcomer channels. Four cold legs are connected with four channels and two hot legs are connected to another two channels. So a total of six channels.

MEMBER CORRADINI: All right. I think I understand. Thank you.

MR. CHON: No problem. And right hand figure graph shows the PCT obtained from 181 simple random sampling calculations. Actually this case is 124 cases.

After discussion with NRC staff we increased the sample, random sampling calculations up to 181. So there's a blowdown peak and reflood and quenching has occurred. Okay, next slide explains about large break LOCA licensing PCT.

It is combination, summation of 95/95 Simple Random Sampling PCT plus delta PCT for bias calculation, plus another delta PCT for time step and consideration. That is 10 celsius.

MEMBER CORRADINI: This is Corradini. How did you get to the 10 Celsius? Is that in the methodology document somewhere?

MR. CHON: Yes. That, let me ask that part to one of the colleagues.

MEMBER CORRADINI: That's all right. If it's in the methodology document I'll go back and look. I just wanted to know where to look for it.

MR. CHON: Yes, it's in the methodology topical report.

MEMBER CORRADINI: Okay, thank you.

MR. CHON: No problem. And finally the acceptance criteria including PCT, clad oxidation and hydrogen generation will be compound for APR1400 design through the final division calculations.

Now I'm going to move to small-break LOCA. This slide shows about the CENPD conservative evaluation model for small-break LOCA analysis. The details of the CENPD methodology is described in CENPD-137P and Supplement 1.

This CENPD methodology consists of multi-code system with CEFLASH-4AS for blowdown hydraulics and COMPERC-II refill/reflood hydraulics, STRIKIN-II hydraulic calculations during blowdown and

PARCH EM hot rod calculation during pool building period. Those multi-code system with CENPD methodology is used in System80+ CESSAR, this document in SBLOCA analysis.

And APR1400 design is the same as System80+ design in terms of loop arrangement and safety injection system design. This slide shows the small-break LOCA modeling diagram with CEFLASH-4AS.

CEFLASH-4AS is used for, it has just a one volume of core including upper plenum, core and lower plenum. And it has two downcomer nodes. And there is a two steam generator part and the intact loop cold legs and pumps suction legs are combined with one loop.

Broken loop part has two loops design. That is the DVI line break location in 28. Next slide shows the small break LOCA analysis initial conditions.

According to the conservative methodology the initial power is 102 percent of normal operation power. Loop and worst single failure of ECCS are selected.

Fifteen DVI line break and 17 cold leg break analysis were performed. One break at the top of the pressurizer was analyzed and also one rupture of in-core instrument tube was evaluated.

Next slide shows small break LOCA result.

The red solid line with circle symbol is the DVI line break result. And red dot line or symbol is pump discharge leg result. The DVI line break result has 15 cases and pump discharge leg has 17 cases.

As you can see in this figure, the peak cladding temperature, the highest peak cladding temperature is occurred at the DVI line break with a break size of 0.1364, 50 squared.

MEMBER CORRADINI: So this is Corradini. May I ask a question? I want to make sure I understand. What's happening to the right of your peak that you get essentially a decrease immediately in peak clad temperature and then it stays almost like a, gets to a plateau and then decreases again?

Where in this is the accumulator? Is this because of timing of the SIP discharge that you get this unusual, I'm trying to understand that shape?

MR. CHON: For the DVI line break?

MEMBER CORRADINI: Well to the right of the DVI line break on your graph that the DVI line break is labeled at about 125 square centimeters. And then you have three points to the right which, the first point decreases significantly then the next two are about the same then it decreases significantly again.

Is this because of the timing of the SIP discharge?

MR. CHON: No, that's because of methodology, procedure. Yes, we used different code for small break size and larger break size the assumption of the core level is assumed much lower.

Let me explain this perhaps from my colleagues. I will, Mr. Lew, can you explain detail about that?

MR. LEW: This is Kaeyeol Lew from KEPCO Fuel Company. So the code has two kinds of, somewhat hydro (phonetic) calculation code. One is the CEFLASH-4AS. The other one is COMPERC-2 code.

So COMPOC2 code, after SIP injection COMPERC code collapse. So COMPERC-2 codes makes core level lower than the real level. So --

MEMBER CORRADINI: So let me, can I just repeat it back to you so I understand. So you've actually changed the computer analysis technique after we get to the right of the peak to a different code methodology?

MR. LEW: Yes, right.

MEMBER CORRADINI: Okay. And then what I think you said was that the two phase level is computed differently in the different methodologies.

MR. LEW: Yes, right.

MEMBER CORRADINI: Okay. Can you remind me what the two methodologies are for the two phase level?

MR. CHON: This is Woochong Chon. Not two methodologies. Just, it's very difficult to explain without figures. But the water level is redefined when we change different, move to the different code calculation.

MEMBER CORRADINI: Redefined meaning you actually change the elevation?

MR. CHON: Make right, make lower elevation of the water level conservatively. So that's the reason why the PCT is increased again.

MEMBER CORRADINI: Okay, all right.

MR. CHON: It's kind of a method to make a definition of the water level between two codes.

MEMBER CORRADINI: All right. For the moment, thank you. I think I get it, kind of. Thank you.

MR. CHON: No problem. Thank you. And I will move to next slide. This slide show us about the issue of loop seal clearing and reformation. Background is given in here at the bottom.

Loop seal reformation due to ECCS injection during the long-term cooling phase of a LOCA can cause suppression of two phase mixture level in the reactor core. If this level drops below the top of the active fuel, cladding heat and oxidation can occur.

The difference between top of the core and top of the horizontal pipe of loop seal is about two feet. APR1400 has deep loop seal design but shallow loop seal design. The loop seal reformation calculation for several break sizes were performed using CENPD small break LOCA methodology.

MEMBER REMPE: So this is Joy. And I know the staff and KHNP had a lot of discussion about what confidence you have in your ability to predict loop seal formation and clearing.

And apparently one of the responses back was that you had data from Semiscale. And I guess the staff got that discussion. But I was curious how prototypic that data is for your geometry.

And I think that adds more to what Jose was mentioning earlier. So could you elaborate on why you have confidence in your methodology based on the Semiscale data?

I know, I'm not fully aware of everything on Semiscale. But is it more for a Westinghouse geometry or is it applicable to your geometry?

MR. CHON: This is Woochong Chon again.

First of all in geometry part we have about 2.3 feet difference between the top of loop seal and top of active core.

That, if core, so if loop seal is filled with water that means the active core will be covered at this level, right. But that's assuming that collapsed the level.

Generally in core the void fraction is over 30 percent. But even though if we assume 20 percent of void fraction in core that mixture level will be covered over top of active core. That means core is not uncovered.

MEMBER REMPE: Okay. Is this documented if I go back to the RAI in sufficient detail that I can have more details?

MR. LEW: Yes, yes.

MEMBER REMPE: Okay. I'll look into it. Thank you.

MEMBER CORRADINI: So, yes, this is Corradini. To follow Dr. Rempe's question I want to make sure in this calculation for loop seal clearing what did you define as the point of clearing, at the bottom of the piping of the loop seal, at the top of the piping?

And also to get back to Dr. March-Leuba's question, the assumption is a three percent bypass flow? I want to make sure I understand the assumptions in computing loop seal clearing.

MEMBER MARCH-LEUBA: Mike, this is Jose. We, the three percent is called bypass flow. So it goes around the floor.

MEMBER CORRADINI: I understand that. I just want to make sure what they used in their calculations.

MEMBER MARCH-LEUBA: Yes, the relevant one is the 0.5 percent from the upper plenum to the downcomer.

MEMBER CORRADINI: Okay, excuse me, 0.5 percent, okay. And then for the loop seal clearing did you, when did you do it, when you got to the bottom elevation of the pipe, to the top, halfway? What was the assumption?

MR. LEW: This is Kaeyeol Lew from KEPCO Fuel Company. Sample method would assume the bottom elevation. So the loop seal pipe bottom elevation is assumed, was assumed.

MEMBER CORRADINI: Okay, okay, thank you.

MEMBER MARCH-LEUBA: Yes, but I don't really understand this logic at all. Can you go back to the slide with the loop seal or you want to say something? Go ahead and I'll wait.

MR. CHON: This is Woochong Chon again. Same CENPD methodology if you see the Slide 39 there's the loop seal figure and right hand side it is Node Number 21 and 11. The junction is five.

This junction is located, we assume in CENPD methodology, the junction is located at the bottom of loop seal. So that's, that assumption is also very conservative. We have space to the top of the loop seal part. That's the part of our assumption.

MEMBER CORRADINI: Can you, do you mind just to repeat that again please? This is Corradini. Can you just repeat that please?

MEMBER MARCH-LEUBA: Tell him the slide number you are saying?

MR. CHON: Slide 39?

MEMBER MARCH-LEUBA: Yes, if you see the Slide 39.

MEMBER CORRADINI: Yes.

MEMBER MARCH-LEUBA: Right hand side there is a loop seal Node Number 21 and 11.

MEMBER CORRADINI: Okay.

MEMBER MARCH-LEUBA: And those loop seal nodes is connected by Junction 5.

MEMBER CORRADINI: Yes, sir.

MEMBER MARCH-LEUBA: That junction location is assumed at the bottom of loop seal.

MEMBER CORRADINI: Right, as you had explained at the beginning. But you are saying that the, when you define bottom you're defining it by the location of the junction?

MEMBER MARCH-LEUBA: Yes, the location of the junction.

MEMBER CORRADINI: Okay, thank you. I got it.

MEMBER SCHULTZ: Steven Shultz. So when you say that it's conservative to locate, to model it this way, have you done the sensitivity to see how conservative? That is have you relocated that junction to an elevation that is at the middle or the top of the loop seal pipe and found out how conservative it is?

We always talk about conservatism. And I think it's fair to do that if in fact we run the calculation and demonstrate that there's a change in the temperature or there's a change in the system performance.

This is, small break LOCA the experience shows that you need to run the calculation to determine whether your assumption of conservatism is correct.

MR. CHON: Most case we perform

sensitivity study. But if physically it is clear then we don't. In this case if, the issue is if loop seal is filled by water then the core pressure have no place to release.

So if we assume at the bottom connection of loop seal node that means water is filled at the bottom and loop seal is not cleared, filled. But in actual case even though water is in the bottom of pipeline but we have still steam pass in the upper pipe region.

MEMBER CORRADINI: So I think Dr. Shultz, this is Corradini, I think Dr. Shultz is asking can you give us a reference where those sensitivities are? Are they in the LOCA methodology document? I don't think I know where to look.

MR. CHON: This case we didn't perform the sensitivity study for the location of junction in loop seal because it is pretty much clear.

MEMBER CORRADINI: Okay, thank you.

MR. CHON: No problem, thank you. And this slide, the next slide which is Page 43. This slide show us about analysis result in loop seal clearing and reformation.

The loop seal reformation shows slight core uncovery intermittently. And the PCT caused by loop seal reformation remains below 800 Fahrenheit.

CHAIR BALLINGER: Okay. This is Ron Ballinger. I've been searching for a convenient place to take a break. And so I'd like to take a break now until 20 minutes of the hour.

(Whereupon, the above-entitled matter went off the record at 10:25 a.m. and resumed at 10:40 a.m.)

CHAIR BALLINGER: Okay. We're back in session. Member March-Leuba would like to make some kind of statement.

MEMBER MARCH-LEUBA: Yes, we've been talking off line. Could you please go to Slide 33? Okay. So we've been talking about the red line which is the downcomer water level and the black line which is the core collapse water level.

We found out during the discussions either they both have different reference. So the zero, if both of them were a zero they would not agree.

The black line is reference to the bottom of the core, the core plate whereas the red line is referenced somewhere to the bottom. It's not the true bottom of the vessel but it's the bottom of the skirt. I'm not sure how to call it. So they have an offset and will always have an offset, correct?

MR. CHON: This is Woochong Chon. Yes. This is Woochong Chon again. That's called fuel skirt.

So active core bottom level is this level. So black line is based on this level.

But red dotted line is downcomer level is based on the fuel skirt location. It is lower than active core bottom.

MEMBER MARCH-LEUBA: By a few meters, right?

MR. CHON: Yes. That height is, I need to check the design data but is around two to three meters.

MEMBER CORRADINI: This is Corradini. Now I'm even more confused. What slide are you on?

MEMBER MARCH-LEUBA: 33.

MEMBER CORRADINI: Well can we go to the one where, that was the demonstration calculation, can we got to 30, now I don't remember. Just before we started talking about spectrum of postulated. It was large break LOCA results.

MEMBER STETKAR: You want the picture, Mike, 30? Is that the one?

MEMBER CORRADINI: Well I'm looking at the one where there's two figures on the slide. It's labeled Number 37 on mine which has --

CHAIR BALLINGER: Mike, I just sent you the latest, the version so we get convergence. If you want to, yes.

MEMBER CORRADINI: Okay, all right.

MEMBER MARCH-LEUBA: It is 36. In our screen it is 36. We have it on the screen.

MEMBER CORRADINI: Okay, all right. Let me get to it, sorry, excuse me. Yes, so 36. So the datums are different?

MEMBER MARCH-LEUBA: Yes, the reference zero, in this case the green line is two or three meters lower than the black line. So you always have an offset of within two, three, four meters because one is reference to the bottom of the downcomer skirt or what you want to call it.

The other one is referenced to the core plate which makes it, maybe the question is why are we plotting them like that?

MEMBER REMPE: Well especially if it's two or three meters why at, what is it ten seconds or something you get to where that they would be lower than the other. So we ought to understand.

MEMBER MARCH-LEUBA: Well, yes. I would love to have this figure in a Korean fashion.

CHAIR BALLINGER: I guess my question is and I would have to go back and look at the documentation, are these the figures that are out of the DCD?

MR. CHON: Yes. This came from DCD.

CHAIR BALLINGER: So in response to Member Rempe's comments, we need to get these fixed or somehow get some kind of --

MEMBER REMPE: An understanding.

CHAIR BALLINGER: -- well some figure caption or something that says, can we put that on the record? We probably ought to try to get the slides fixed as well because sooner or later we're going to have to go back and look at these things.

MR. CHON: Let's go back to Slide 44. Okay. This is Woochong Chon again. I will present about the post-LOCA boron dilution analysis. This slide shows about the issue of the post-LOCA boron dilution analysis.

Background given here. Following a LOCA a slug of water can be formed in the loop seal by the condensed steam in steam generator tubes. The slug enters the vessel through a cold leg and then travels along the downcomer.

Again, the slug moves into the lower plenum and it turns upward to enter the core. During this period it may cause reactivity excursion if the water slug is not sufficiently mixed with the borated water in the RCS.

And also we tested that the core should not reach recriticality when the boron dilution accident occurs. There is simple figures given in the next slide.

If condensed the water is accumulated in the loop seal and core cold leg part, that is the lined part will be unborated water. So in analysis result the two cases were studied.

First case is restart of one RCP and next to one is start of natural circulation. The mixing evaluation shows that the downcomer and lower plenum water mixes well with water from the loop seal.

KHNP has demonstrated that most of the result of one RCP and the initiation of natural circulation will not cause core recriticality.

MEMBER MARCH-LEUBA: Are we talking normal recording mode? Do I have to say my name?

CHAIR BALLINGER: I think we're in normal recording mode.

MEMBER MARCH-LEUBA: This remixing, I'm more familiar with BWR where the mixing of boron with water that is nonborated is a problem and it becomes stagnated. So do you have any experimental evidence of that cold distilled water, unborated water will mix with hot borated water? Will it go up to the bottom on a stagnator and slowly rise into the core?

MR. CHON: Let me ask my colleague.

MR. LEW: This is Kaeyeol Lew from KEPCO Fuel Company. So there was some misunderstanding that you said. So borated water has lower density, high enthalpy and then unborated water has high density, lower enthalpy. So --

MEMBER MARCH-LEUBA: So let me repeat that again and see if you understand. The clean water, the boron doesn't add that much weight to the water. It's the temperature that matters.

So the clean water is cold. The borated water is hot. So I can see how you can have sediment in the bottom.

MR. CHON: No, it's, this is Woochong Chon. That's a positive. The unborated water is condensed from the steam generator. So it's hot water.

MEMBER MARCH-LEUBA: But it's not hotter than the core. It's colder than the core. It's not 20 degrees. But it's colder than the core.

MR. CHON: Do you have any data the temperature difference between the core water and the loop seal water temperature differences?

MR. LEW: So ECCS water has lower -- so loop seal water, loop seal water.

MEMBER MARCH-LEUBA: ECCS correct. ECCS is cold water, is heavy.

MR. LEW: Yes. So high density slug of unborated water so cannot be penetrated. So colder water, so we assume so for mixing loop seal water and it's hot water we used conservative assumption.

MR. CHON: This is Woochong Chon again. The lower plenum water is not the, came from the core mainly injected from the SI system. So that water is colder than the condensed water.

MEMBER MARCH-LEUBA: Okay. So let me be the devil's advocate. The new loop seal clean water bypasses the core water in the bottom of the vessel and never picks up the boron which is what happens.

And there are experimental data that tells you that the cold borated water settles in the bottom and nothing goes into the core. When you reach a certain core flow, which according to four to eight percent core flow part of the cold borated water is at the bottom of the vessel and nothing goes into the core.

That's why ATWS and BWR becomes a problem and you have to raise the water level and do other things.

MR. CHON: Some part is like condense the high temperature water basically cannot pass through the downcomer because it is high temperature. And the lower plenum water is low temperature. So there is no, you know, the unborated water cannot penetrate in the lower plenum part.

MEMBER MARCH-LEUBA: They never mix it into the vessel.

MR. CHON: Physically. But we assume that water can penetrate to the lower plenum part and calculate.

MEMBER MARCH-LEUBA: Okay, thank you.

MR. CHON: That's very conservative assumption. Okay. The next slide the basic function of, now let's move on to the long-term cooling, post-LOCA long-term cooling.

CHAIR BALLINGER: Before we get on this I need to make sure that we're clear. On long-term cooling we have to write a letter specific to long-term cooling. And that means that we've spoken with the staff and others that we need to have a presentation from both the staff and KHNP on the same day in the same meeting.

And so we need to be careful of what we're saying here because I'm sure you're going to need to give this presentation again, if I'm not mistaken.

MR. CHON: Okay.

MEMBER CORRADINI: Ron, this is Corradini. I think staff is going to address that later today according to what I've read.

CHAIR BALLINGER: Well there may be some new information here.

MR. LU: Yes. Let me make a comment on this one. We do have a, do understand that the ECCS Subcommittee and that the full Committee needs to write a letter specifically responding to SRM regarding long-term cooling.

And that as a reality here specifically for a long-term cooling plan and then the really major issue of the downstream effects and also the strainer, the NPSH issue. So basically we're talking about GSI-191, right.

So a handful of GSI-191 has already been presented to the Subcommittee as part of Chapter 6. And NPSH margin analysis, the strainer, you know, pressure drop across the strainer, that part has already been covered by the staff.

So today we're also going to cover that simply because that's part of the Chapter 15. We do want to cover that part as our presentation there. So we also understand that if you go through this presentation and you still have specific questions related to long-term cooling and the related GSI-191 we can definitely support any additional discussion if you do want to have that one.

My understanding is that in October we plan to talk about large break LOCA topical and we also have fuel topical report, fuel seismic issue and thermal conductivity degradation. Those are really high, from our perspective, significant safety issues.

And regarding GSI-191 as of today after we go through with our presentation and then right now I think we are going to have a summary matter. And then you can see how much you need to dive into more and then we can definitely support any additional requests for that communication.

CHAIR BALLINGER: Okay. We will have to do this offline I guess.

MR. CHON: Yes. We have, this is Woochong Chon again. We have two post-LOCA long-term cooling presentations today. I will present the post-LOCA long-term cooling in DCD Chapter 15.6.5. After my presentation Mr. Kim will present about post-LOCA long-term cooling and in-vessel downstream effect, the GSI-191.

So two long-term cooling presentations will be filed. In Slide 46 the basic function of long-term cooling is to maintain the core at safe temperature level while avoiding the precipitation of boric acid in RCS.

In the long term operator action is needed to provide reasonable assurance that the core cooling is maintained until the plant is brought to a cold shutdown condition. There is behavioral difference between large and small break LOCAs in the long term.

The large break are adequately cooled by the safety injection flow because this flow is large due to the low RCS pressure. However, the large breaks use simultaneous hot leg and direct vessel injection to flush boric acid from the vessel.

In small break, the RCS will remain at high pressure and the safety injection flow rate will be too low for effective cooling. Thus small break requiring cooling of RCS by the steam generator until shutdown cooling can be initiated.

Next slide shows the long-term cooling evaluation model. The evaluation model is based on the CENPD-254-P-A which is approved version of methodology.

The long-term cooling calculation are performed by using four long-term cooling codes which is CELDA, a long-term depressurization and refill of RCS and NATFLOW, CEPAC and BORON. NRC approved interim method was adopted for APR1400 calculation.

The interim method provided resolution of issues to CENPD-254. Next slide shows that applying mixing volume change in APR1400. Limiting mixing volume in boron participation analysis is changed from top of the hot leg to the bottom of the hot leg.

In the original case of mixing volume the boric acid to precipitate until 3.2 hours. When the mixing volume decreased to the bottom of the hot leg then boric acid would be predicted to precipitate at 2.3 hours.

So we have more conservative initial conditions. Next slide shows post-LOCA long-term cooling result. Three results about the boron precipitation.

First one is no core flush. If you see the right hand side of the figure the temperatures boric acid concentrations. The right line is no core flush. The straight line at the middle is solubility limit.

So with no core flush there is boron precipitation will be occurring in this region. With core flush when the operator initiates simultaneous injection about two hours at this point there is no boric acid precipitation occurs with simultaneous injection core flush.

The 30 gpm flush the margin provided for the prevention of boric acid precipitation by the core flushing flow of 30 gpm. It is actual value.

Next slide shows the post-LOCA long-term cooling result, another result. The right hand side it shows the break area from the small break area to the big, larger break area. And right part is RCS pressure.

With applying larger break LOCA long-term cooling methodology we use simultaneous injection from 3.7 square centimeter break area. And for the small break LOCA long-term cooling method we applying from 34.4 square centimeter, smaller than that size of break.

So we have overlap region here. The overlap in break area for either the larger break or small break procedures can be used as illustrated in this right hand side figure.

And the results demonstrate that the break as large as 34.4 square centimeter are able to use shutdown cooling system for the long-term cooling and flushing of the core. The long-term cooling analysis itemized that the larger break procedures can flush the core for break area down to 3.7 square centimeter.

Therefore, the plant can be secured for all break sizes. That is the end of Chapter 15 post-LOCA long-term cooling analysis. Now Mr. Kim will present the evaluation of in-vessel downstream effect.

MR. Y. KIM: Good morning. My name is Youggun Kim from KEPCO E&C. I would like to introduce the evaluation of the in-vessel downstream effect for a APR1400. There is one part of the evaluation of the GSI-191 issue.

Going through we'll start with the origination. APR1400 according to the guidance of NEI 04-07 actually it's a hot leg line selected as the limiting case of the break location.

The generated debris would be RMI, reflective metallic insulation coatings laid into the raised concrete and aluminum inside the containment. And for conservatism, APR1400 assumes that all generated coatings and all debris are transported to the sump strainer in the IRWST.

In the strainer bypass testing fibrous debris at the strainer is established as 6.8 kilogram of the latent fiber. Testing concluded only fibrous debris since adding particulates may reduce the amount of the fibrous debris due to clogging at the strainer.

And the filter bag is used to collect the debris by passing it through the strainer. And bypass through fibrous mass is 1.67 kilogram through the four sump strainers.

MEMBER MARCH-LEUBA: So the 15 pound, the first bullet that's through one of the strainers or through four strainers?

MR. Y. KIM: It is the whole latent debris mass.

MEMBER MARCH-LEUBA: On all four?

MR. Y. KIM: Yes, all four.

MEMBER MARCH-LEUBA: Okay. So roughly 20, 25, 20 percent of the fiber goes through the strainer? I mean, 368 divided by 15?

MR. Y. KIM: That's right. And the fibrous debris mass per fuel assembly considering the 241 fuel assemblies in the core is calculated 6.93 gram, per fuel assembly. That is the result.

This is the result of the 6.93 gram is on equal for the in-vessel fuel assembly test. This is flow rate for the core at the time according to the LOCA scenario for the in-vessel downstream evaluation representative of LOCA scenario as selected as their hot leg break, cold leak break and the cold leak break if the hot leg switched over.

In the event of the hot leg break all the safety injection water go to the reactor core and the flow rate is equal to the full safety injection flow rate. That is the 4,940 gpm.

In the test, in the in-vessel housing fuel assembly test one fuel assembly was used so the flow rate per fuel assembly is calculated by dividing the total flow rate, dividing total flow rate by the total fuel assembly of 241 and the flow rate per fuel assembly is 20.5 gpm.

In the event of a cold leg break the flow rate to the core is equal to the boil-off rate at the moment. The maximum boil-off rate has calculated at the equation's start time over the 700 second and the flow rate cold fuel assembly is 3.65 gpm.

At the time of two hour of the cold leg break operators test simultaneous operation of the hot leg injection and the direct vessel injection because of the, because two safety injection pumps are for hot leg. And so the flow rate to the core is half of the total injected, safety injection flow rate.

This is the test to measure the pressure drop. We'll call it plus seven. This simulates the APR1400 fuel assembly. The schematic drawing and the photo of the test loop are given here.

The description for this test loop are on the next slide. The test facility is composed of four main parts. Test the column, the leaching tank. The circulation system and the control and monitoring system.

The test column has half, full length of plus seven fuel assembly. Pressure drop, pressure drops are measured at five points. Bottom -- for mid grid, bottom grid, for mid grid and top grid and top measure and full length.

At the leaching tank, a heater and a chiller are used to control the water temperature and a stirrer is installed to prevent debris settling. A recirculation pump and flow meter are installed downstream of the tank and the flow rate is adjustable.

The temperature are measured at four points bottom and top of the tester column are the lower part of the leaching tank. Account for the parameter water flow rate and the water temperature using monitoring system and flow rate. Temperature and pressure are recorded.

This slide I already explained so I'll skip this slide. The table on this slide summarizes the different types and amount of fuel assembly for the in-vessel test, fuel assembly test.

The in-vessel fuel assembly test is for measuring the pressure drop of the fuel assembly when the excess water with bypass debris flows into the fuel assembly during the long-term core cooling operation.

For the fibrous debris although the fibrous debris is 6.93 gram per fuel assembly, in actual test the 15 gram was applied for conservatism. And all other debris such as coating, particle and chemical compounds is assumed to bypass the sump strainer and comes to the core.

For hot leg breaker condition we have tested five conditions, five tests to evaluate. The particle to fiber ratio ranged from .5 to 10. The limiting result, I'll call that the particle to fiber ratio equals one.

At the time zero you can see the sequence of the tests in the right hand side of the graph. At the time zero or the particle were added at this point.

And then 9 grams and 6 grams of fiber was inserted with 25 minute interval this part. After two hours the fiber chemical compound was added. And after that differential pressure is increased considerably and then additional compound, chemical compound was inserted but it did not make the differential pressure increase.

The additional pressure of the pressure drop was 9.4 kilopascal. And therefore the test result criteria was 42.7 kilopascal with absorption margin.

MEMBER SKILLMAN: What was added at two and a half or three hours please?

MR. Y. KIM: The basis of the interval you mean?

MEMBER SKILLMAN: No. At three hours --

MR. Y. KIM: At three hours --

MEMBER SKILLMAN: -- you added some material that caused that differential pressure to increase --

MR. Y. KIM: Yes, right.

MEMBER SKILLMAN: -- so rapidly. What was the material that you added?

MR. Y. KIM: The chemical compound aluminum hydroxide.

MEMBER SKILLMAN: I understand. Thank you, thank you.

MEMBER MARCH-LEUBA: And the pressure drop you are reporting is across the complete fuel, just what pressure drop?

MR. Y. KIM: The pressure drop is a total length pressure drop.

MEMBER MARCH-LEUBA: Total length of the --

MR. Y. KIM: From bottom to top.

MEMBER MARCH-LEUBA: I'll ask you at the end of this. But maybe you can say now. Have you already strained it or say, yes, 15 grams is not sufficient to build up be of any relevance on the strainer?

MR. Y. KIM: Well the 15 gram is just applied to the test for conservatism. Actual, the bypass fiber --

MEMBER MARCH-LEUBA: I'm asking about the NPSH for the ECCS.

MR. Y. KIM: For NPSH test we used fiber, all the latent fiber debris. It is 15 pounds, 6.8 kilogram.

MEMBER MARCH-LEUBA: And it was satisfactory?

MR. Y. KIM: It was satisfactory. And the rest is the code break. The seven tests that had been run to evaluate the cold leg break condition. Particle to fiber ratio ranged from one to 16. The latter figure shows the pressure drop with the changing particle to fiber ratio.

The maximum pressure drop, I'll call that particle to fiber ratio, equals 50. At the time zero all the particles were added and then nine grams of fiber and 16 grams of fiber was inserted into the, in two hour interval.

After four hours all chemical compounds was added. After that the differential pressure increased considerably and the maximum pressure drop was 3.85 kilopascal.

For the test result shows that the absorption margin to maintain the core flow during a post-LOCA condition and from that acceptance criteria. So all the in-vessel fuel assembly testing results shows that there are sufficient margin.

This is the in-vessel test result, the fuel assembly test. And the --

MEMBER MARCH-LEUBA: Can you elaborate a little bit about the acceptance criteria, how it was determined?

MR. Y. KIM: Yes. This is the calculation result based on the WK-16793 (phonetic). The WK report is presenting the methodology how to calculate the acceptance criteria. These results are based on that methodology.

MEMBER MARCH-LEUBA: Okay, thank you.

MEMBER SCHULTZ: Excuse me, Steve Shultz. In the, each of these two cases you ran several tests.

MR. Y. KIM: Yes.

MEMBER SCHULTZ: In the tests that you're not showing here what was the variation in input parameter and how did the results change from test to test? You're showing a limiting result in each of these two slides for the cold leg and the hot leg.

I'm just wondering what the variation was in the other tests that were run in terms of the results. What did you vary in the input parameter and what was the difference in result?

MR. Y. KIM: Yes. Actually the variation is on the particle to fiber ratio. And we differed the ratio as shown in this graph. So the same test, this result. The maximum pressure drop is the point of this graph.

And finally found that the p:f ratio in the p:f ratio is 15 and this is the limiting case of the cold leg break.

MEMBER SCHULTZ: So the change was, seems more dramatic in the hot leg break in the low range of particle to fiber ratio and in the cold leg break across the spectrum of particle to fiber ratio there is some change but not dramatic. Is that a good summary, a fair summary?

MR. Y. KIM: Well I think that detailed information is actually describing --

MEMBER SCHULTZ: I'll take a look at that.

MR. Y. KIM: I'm sorry about that.

MEMBER SCHULTZ: But, thank you.

MEMBER MARCH-LEUBA: So when you changed the particle to fiber ratio do you keep the total mass as 15? So you're reducing the amount of fiber for the higher.

MR. Y. KIM: Yes, yes.

MEMBER MARCH-LEUBA: So the sum of the two is 15. So you're reducing fiber then adding particles as you move to the right?

MR. Y. KIM: Well we used the total fiber mass to 15 gram and in the range of the zero to 15 it made the difference.

MEMBER MARCH-LEUBA: So in that figure on the left that is in Slide 56 when it says two do you have 15 grams of fiber and 30 grams of particle?

MR. Y. KIM: That's right, that's right.

MEMBER MARCH-LEUBA: And then at six you have 15 grams of fiber and 100 --

MR. Y. KIM: That's right.

MEMBER MARCH-LEUBA: Okay. So you increase, you keep the same fiber and you increase the particles and the pressure level goes down?

MR. Y. KIM: That's right.

MEMBER MARCH-LEUBA: After cleaning the filter?

MR. Y. KIM: Yes. That was the test

result. And the WK Report shows the examples about the p:f ratio changes and the result and it shows the same range. So we believe that this test has the same result with the chemical.

MEMBER MARCH-LEUBA: I don't argue with experimental results. But this one is counterintuitive.

MR. Y. KIM: But I can explain why the --

MEMBER MARCH-LEUBA: And I haven't been following GS-191 like other members, whatever.

MEMBER SCHULTZ: The other question is, you know, the dramatic change occurs when you add the chemical. So did you, was the addition of the chemical varied in the tests or was it the same amount that you described in the chart each time?

MR. Y. KIM: Well in the cold leg break condition we used the water, the chemical compound in this table.

MEMBER SCHULTZ: Yes.

MR. Y. KIM: In this table.

MEMBER SCHULTZ: And for the hot leg?

MR. Y. KIM: For the hot leg first we input, we divided the total amount of chemical with two or three. But first we inserted the chemical compound there was the pressure drop increase.

But after more chemical compounds it does not make any pressure drop higher.

MEMBER SCHULTZ: There's no difference after --

MR. Y. KIM: Yes, so --

MEMBER SCHULTZ: -- a certain number?

MR. Y. KIM: -- in this case we don't have to put more in the compound.

MEMBER SCHULTZ: Thank you.

MEMBER CORRADINI: This is Corradini. Just so I make sure I understand your answer to Dr. Shultz, so 70 is an upper bound on the amount of chemicals you added even though you added them differently between the cold and the hot leg experiments. Is that correct?

MR. Y. KIM: Yes, the chemical compound, the total chemical compound for the hot leg condition and cold leg was the same.

MEMBER CORRADINI: Okay. And you saw no reason to have it phased addition after you saw the results for the cold leg or for the hot leg, excuse me? Okay, thank you.

MEMBER SKILLMAN: Let me pursue this a little further. If I look at the cold leg chemical addition I see 35 liters. And if I look at the hot leg

I would see 70 liters.

MEMBER CORRADINI: I think they added it twice, Dick.

MEMBER SKILLMAN: That's what I'm asking. You have two pink carrots. I'm sorry, there's just one. I see it. I withdraw my question. I understand. Thank you, okay.

MEMBER REMPE: Before we switch to a new topic can I circle back on something I mentioned earlier please, Mr. Chairman?

CHAIR BALLINGER: You always are, never mind. Circle back if you will.

MEMBER REMPE: Okay. I looked up the response to Question Number 15.06.05-19 about the use of the Semiscale facility and the special version of the code that KHNP used to predict loop seal clearing.

And in more recent times we have used CFD analyses against smaller scale facilities and then taken some parameters to try and simulate that with other codes. I did not see that in this response.

All I saw was we tuned it. We matched Semiscale and then we used it. And that's what I was trying to get to is what gives us confidence that because you can match Semiscale that it's appropriate for the APR1400?

Is there something else that I should be looking at? Am I misunderstanding what I'm reading in this RAI?

MR. OH: This is Andy Oh, KEPCO. This, could you speak to that again. What's the RAI number and what's that about?

MEMBER REMPE: In the draft SE that the staff provided when they brought up this issue about the applicability or to the code based on whatever, for the APR1400 they cited RAI to question Number 15.06.05-19. And I'll ask the staff when they come up too why they're confident.

But I didn't see it in their write up. They just said, yes, they had benchmarked it against Semiscale. And so I was curious because of some other activities I've been reviewing of what gave them confidence.

And so in more recent times we've seen people use CFD analyses for a Westinghouse prototype thing and then apply it to a CE with that confidence in their ability to predict what the CFD type of code what was going on in the Westinghouse geometry then they turned and used it for the CE geometry.

And so what I'm asking is why do you feel like being able to without using any CFD match

Semiscale, which maybe it is prototypic for the APR1400, I'm not familiar with that design. But what gives you confidence that a code that you can match to the Semiscale facility can be used for the APR1400?

And I don't see that in this RAI response. And so I was curious if there's something else. And you don't have to answer it now. I do plan to ask the staff and maybe they've got the answer.

MR. LU: Yes, we do.

MEMBER REMPE: Well then we'll count on the staff to help me feel better. Okay, so thank you.

MR. Y. KIM: I am Youggun Kim and from the next slide Mr. Dongsu Lee will go on with the presentation.

MR. LEE: My name is Dongsu Lee working in radiation protection team at KEPCO E&C. We can start my presentation I would like to correct the information I provide you on CA injection dose calculation.

For this event a ten percent high value of the power model built in the reactor coolant was used for application of aerosol equation effect. And that there's a spray it was not considered whether the CA was calculation.

The second information for the event generation, generated iodine spiking there are concurrent iodine spiking factor over 300.35 was considered tube rupture. Those calculation to compliance with the Regulatory Guide 1.183. So 300.35 was considered.

And I would like to start Chapter 15 at Section 15.7. In this section GWMS leak or failure events are described in Section 11.3.3. And LWMS leak or failure events has been deleted. And postulated radioactive release due to liquid-containing tank failure has been added to Section 11.2.3.

In the postulated fuel handling accident a fuel assembly is assumed to be dropped and damaged during fuel handling. The accident takes place in the containment or in the spent fuel pool inside the fuel handling area of the auxiliary building.

Let's move on to the next page. The ATWS is defined as AOO followed by the failure of the reactor trip portion of the protection system. According to 10 CFR 50.62, it is required to reduce risk from ATWS events for light-water-cooled nuclear power plants.

For ATWS, diverse protection system is installed in the APR1400. The DPS helps the PPS to address 10 CFR 50.62 requirements for reduction of risk from ATWS events.

The DPS design includes a reactor trip and auxiliary feedwater actuation. The DPS reactor trip provides a simple and diverse mechanism to significantly decrease risk from the ATWS events.

And the DPS auxiliary feedwater actuation provides additional assurance that ATWS events could be mitigated. The DPS functions are explained in detail in the Subsection 7.8.2.

From this slide I am going to talk about Radiological Consequence Analysis. This presentation consists of five parts as shown this. Let's move on to next page.

This slide shows the design targets and the design features for the dose analysis of DBA accidents. For the EAB and LPZ dose targets are taken from 10 CFR 52.47 and according to the SRP those limitations can be used for each DBA case.

And based on the GDC the limitation on the MCR worker is taken. To minimize accident release following systems are used. Safety injection system, auxiliary feedwater system, containment spray system is used.

These four kinds of actuation signals initiate the corresponding emergency systems. And limitation of leakage containment is lined by steel. And lastly MCR operators are protected by two designs selective air intakes and positive pressure in MCR. Let's move on to next page.

MEMBER POWERS: What is your unfiltered leakage in to the main control room?

MR. LEE: 100 CRF we used.

MEMBER SCHULTZ: So in terms of that value where did it come from? Is it, I understand it was a different value at one point in your analyses and you had used 300 in some of the earlier work that you had done. And that was modified to 100.

I'm curious to know where each of those numbers were derived.

MR. LEE: Based on that our domestic areas the unfiltered indication is 225 cfm. So our 100 cfm has some conductivity. So as an engineering judgment we decided that 100 is the outcome. And from the 300 to the 100 at the time we changed to that number based on that our consequence analysis.

MEMBER SCHULTZ: Okay, so the, when you say the values that you are familiar with are 25 cfm where, are those derived from testing at other facilities?

MR. LEE: Yes, right, testing at the other DC applicant numbers was bounded by our numbers 100 cfm.

MEMBER SCHULTZ: But those others were from facility testing?

MR. LEE: Yes, right.

MEMBER SCHULTZ: And those tests environments were the systems pressurized? Was there a pressurization system within the control room?

MR. LEE: At the TSP, yes.

MEMBER SCHULTZ: Okay, so that's, so the testing environment applies to the design of the APR1400?

MR. LEE: In our, within that the number is the testing should be satisfied.

MEMBER SCHULTZ: Right. Thank you.

MEMBER STETKAR: Before you go on we have not reviewed Chapter 7 of the DCD yet. And on Slide 59 I think you said that your, the diverse protection system initiates reactor trip and auxiliary feedwater.

It does not provide a diverse signal to trip the main turbine or does it? I'm trying to read parts of Chapter 7 in real time here and I don't see it.

MR. OH: This is Andy Oh, KHNP Washington office. DPS system is basically generated the turbine. However, for APR1400 we have the RPCS system with our protection and bypass system.

So when the RP, the power is over 75 percent and RPCS is available cases that the telemetry function is disabled. But the RPCS is disabled at that telemetry case function is enabled. So basically DPS has some telemetry function.

MEMBER STETKAR: It does somehow?

MR. OH: It does, yes.

MEMBER STETKAR: Okay. We'll look at that when we get to Chapter 7. Thanks, Andy.

MR. LEE: Okay. Let's go one, there. This slide shows the analysis method for accident dose calculation. For the LOCA melted core source term is assumed and the detailed assumption and the parameters are presented in the slide of 15A.3.

For the Non-LOCA events damaged the fuel and the mass release data based on the thermal hydraulic analysis I used and the detailed assumption and parameters are presented in Slide 15A.4. Based on the Alternative Source Term and dose criteria of Total Effective Dose Equivalent, radiological consequence analysis are performed.

And lastly RADTRAD code and the conservative atmospheric dispersion factors presented in Chapter 2 were used. These approaches are consistent with Reg Guide 1.183. Let's move on to the next page.

This slide shows the detailed assumption and the parameters for LOCA. For the LOCA, core fission product inventory which developed based on the 102 percent rated power and the 56.4 GWD/MTU burnup.

For the containment building conditions, the sprayed region is 75 percent of the total containment net free volume. Two volumes of unsprayed region per hour was used for air mixing rate.

It is assumed that the elemental and the particulate iodines are removed by CS containment spray system based on the model described in the SRP. And the ten percentile values of the PowerS Model built into the RADTRAD code was used for application of the aerosol deposition effect.

Let's move on to the next page. It is assumed that the containment purge is isolated at five second after LOCA onset. And for the ESF system leakage was assumed with two times of the design leakage. The ACU filtering is assumed.

In the post-LCOA condition, the pH of IRWST was, is evaluated to provide reasonable assurance that the minimum pH values can be maintained above a seven for 30 days in LOCA condition. The following materials are considered and the conservative radiation conditions are used. Let's move on to the next page.

MEMBER POWERS: What did you use for your dose in the water?

MR. LEE: When the LOCA conditions we used to measure the core radiation conditions, the gamma and the beta ray energy in the containment building at highest level we used.

MEMBER POWERS: And what did that dose run, dose rate?

MR. LEE: The number?

MEMBER POWERS: Yes, roughly.

MR. LEE: Four or five, ten to five gray.

MEMBER POWERS: Gray, I have to translate that, decent.

MR. LEE: But it comes up with your number. Okay. Let's move on to the next page. This schematic diagram shows the radioactivity transport model for LOCA.

Following a LOCA event, radioactivity is released from the fuel into containment and released into the environment through the containment low-volume purge and the containment leakage. Once the ESFs are actuated, radioactivity in the IRWST solution can be released to the environment from ESF equipment into the auxiliary building.

A reduction of the airborne radioactivity by containment spray is, natural depositions are credited. Let's move on to the next page.

MEMBER SCHULTZ: What type of testing do you have to, that's demonstrated the capability for this design's containment spray, the containment spray system? Have you done testing? Is it similar to other design containment spray systems that are in place?

MR. LEE: Containment, according to the, I'm sorry.

MEMBER SCHULTZ: I'm sorry, other systems that have been designed and tested. Is this a new containment spray design based upon experience or is it the same based upon experience?

MR. LEE: As far as I know the spray system should be, meet the requirement of the 90 percent area of the containment. That is according to NC 56.5. So in our DCD as far as I know that is it.

We tested that, the area of the spray nozzle. But at the time that we used it the minimum sprayed area, yes. So even though we used it the minimum spray area the area can satisfy that requirement.

MEMBER SCHULTZ: But you have done testing to demonstrate you've got, that if the spray is demanded you're going to, you will in fact have coverage like you expect?

MR. LEE: And there's some kinds of nozzles and we tested the area. And the smallest area we used to how much covered the section of that area, the containment area. So anyway that, yes.

MEMBER SCHULTZ: And the sprayed and unsprayed region that you're showing here is it in the way it's modeled or in the way it is physically?

MR. LEE: Yes, we calculate based on the physical structure design and this requirement is also is NRC requirements. So we --

MEMBER SCHULTZ: This is a simple diagram. But in fact you do, this spray is not going to get down into the lower containment area as shown on the diagram.

MR. LEE: The spray, 25 percent of the volume is that there is unsprayed region and we cannot credit that spray in this area. So we can credit that the mixing from the sprayed and unsprayed area. That is a direct authority that we can use the two types of sprayed area volumes for our we can use that.

MEMBER SCHULTZ: Okay, thank you.

MR. LEE: Let's move on to next page. From now let me introduce the dose calculation for non-LOCA cases. For the source term of the primary coolant, noble gas and iodines are assumed to exist with the Tech Spec LCO.

And secondary coolant is as well considered the Tech Spec conditions. And specifically for the iodine source terms two kinds of spike effects were considered to comprise with Reg Guide 1.183.

For the events which experience fuel cladding damage, it is assume that the fission product in gap are released to the primary coolant. The release fractions are used in conjunction with the core fission product inventory with the maximum core radial peaking factor of 1.8. The gap inventories are determined based on the Reg Guide 1.183.

Next page. For the steam generator leak rate, 0.3 gallons per minute is assumed for one steam generator.

For the non-LOCA cases the fuel cladding damage rate determined based on the thermal hydraulic analyses are used as follows. And DF of iodine in the steam generator can be determined based on the covered or uncovered tube condition by secondary coolant.

Let's move on to next page. As an example for non-LOCA cases, let me introduce the CEA ejection dose calculation. Radiological consequences for the CEA ejection events are calculated for two release cases.

First one is the containment release. Second one is release through the secondary system.

This slide shows the containment leaked.

For the containment leakage it is assumed that all activities in the gap of the failed fuels are instantaneously mixed throughout the containment air. And those are available for leak to the environment.

Reduction in airborne radioactivity in the containment by the containment spray system or by the natural deposition within containment can be credited. And next page.

And this slide shows that the release through the secondary system. For the release through the secondary system, activity release from the secondary system is considered.

Since the tube design leakage in the steam generator considered the activities consist of the initial primary activity and the failed fuel gap activity and the initial activity of the secondary side. The appropriate partitioning coefficient, flashing fraction and the fuel failure rate are considered for dose calculation.

Last page is that as it shows the results. Doses to the public at the EAB/LPZ for all DBA are well within dose limits of 10 CFR 52.47. And MCR habitability is ensured for all DBAs by complying the criteria in the GDC 19.

MEMBER POWERS: I understand how you get your doses at the site boundary. How do you disperse the radioactivity around the main control room?

MR. LEE: We calculate on site chi/Q based on the --

MEMBER POWERS: You actually used the chi/Q for the site itself?

MR. LEE: Yes, so we --

MEMBER POWERS: Doesn't that kind of do violence to the assumptions in the chi/Q?

MR. LEE: No.

MEMBER POWERS: I mean, chi/Q is assuming a Gaussian plume. But locally to the plant there can't possibly, I mean it's just not going to be a Gaussian. There's all the wake effects of the building and things like. How accurate is that assumption?

MR. LEE: We used ARCON-96 code that is, the Guide 1.194 as far as I remember we used that. And also is that we picked that, picked the collect the radiological data from the U.S. site. So we compare each site and so we picked the six site of the conservative cases.

And then we compared that. So we select the one. But at the time, to envelop the most kinds of U.S. sites, the sufficient margin was applied. That is

50 percent.

So we believe that this methodological data -- this chi/Q can be very conductive chi/Q based on that conductive material data in USA. And also provided that the 50 percent of margin.

MEMBER POWERS: Well you slapped a lot of conservatism on there. I can't argue with that. But I mean the reality is that the flows around a plant itself are really very complicated.

MR. LEE: Yes, right.

MEMBER POWERS: And where the inlets are and your leakage into your main, the unfiltered leakage into your main control room that become important here. Technically your main control room is going to be your site boundary limiting typically.

It's the main control room that's limiting. Incidentally I did a back-of-the-envelope calculation on your steam generator tube rupture and I came up with almost exactly your numbers.

MEMBER SCHULTZ: Another question on the control room. You've taken a, I guess I would call it you've got your system that allows automatic selection --

MR. LEE: Yes, right.

MEMBER SCHULTZ: -- of the ventilation

intakes. And it's the first time I've seen it applied, designed and applied. And you've taken some conservatism.

I know that the Reg Guide allows a value of, you know, a factor of ten I think for a dual system with selectability. Did, as you've evaluated that yourself, did you also come up with that you should have a factor of ten or do you think it should be higher than that?

Again it's the first time I've seen it applied in an automatic mode. And so I'd be interested in your perspective as you've evaluated it. Do you think a factor of ten is fair? I know you've taken something less for conservatism. But could you speak to that for a moment?

MR. LEE: Yes. I don't have any idea about the real testing about the selection. But in our methodological consequence analysis at the time it's that we can, as you mentioned that we can use the reduction factor of ten in case that is from the release point that the MCL intake should be different window trajection and also they have the auto selection function.

At the time that we can use a ten reduction factor. But as you mentioned, that is that we have, we would like to have some conservatism so we use that factor of eight not ten. So we have like 20 percent of the margin.

MEMBER SCHULTZ: But some of my question goes to, as Member Powers indicated, the chi/Q for the control room you get two intakes. But, you know, they're not separated, I presume by much, are they in the design?

You're automatically going back and forth depending on the level of activity sensed. So again, did you evaluate it and determine, yes, a factor of ten is appropriate or a factor or eight is appropriate?

MR. LEE: Yes. Based on our RAI that we did calculate the auto selection, the reopen, the function at one hour, each one hours select open and reopen and we can measure the levels. And at the time if that, same at the time close to that direction.

But if that wind direction was changing at the time I think that the intake point will, should be changing.

MEMBER SCHULTZ: Yes, so you've thought about it, you've evaluated it and you feel that you ought to have credit for a factor of eight?

MR. LEE: Pardon me.

MEMBER SCHULTZ: You feel you ought to have credit, given the system and its design, you feel you ought to have credit for a factor of eight and you feel that's conservative, a factor of ten could have been justified?

MR. LEE: So in our estimation in our methodological analysis at the time we used eight. So we have it at 20 percent.

MEMBER SCHULTZ: Yes.

MR. LEE: At the time we considered it reopened at the time eight percent of dose increasing. But we did it at the hour margins. So those rates cannot be changing.

MEMBER SCHULTZ: Okay, thank you.

MEMBER STETKAR: I think just for the record I believe, Steve, that the intakes are widely separated, aren't they. One, in fact they're on opposite ends of the building.

MEMBER SCHULTZ: No. I understand but in terms, Member Powers brings up a good point in terms of the chi/Q evaluation and --

MEMBER STETKAR: Sure, sure that's --

MEMBER SCHULTZ: -- there's a little fuzziness there.

MEMBER STETKAR: But in terms of --

MEMBER SCHULTZ: That's all I meant.

MEMBER STETKAR: In terms of the separation we're not talking about, you know, ten meters apart.

MEMBER SCHULTZ: Right.

MEMBER STETKAR: We're talking about a large distance apart.

MEMBER SCHULTZ: Right. Either side of the facility, yes.

MEMBER STETKAR: Yes.

MEMBER SCHULTZ: Definitely, I understand that. Otherwise you wouldn't get anything near a factor of eight. That's pretty large. But what I'm going back to is when the Reg Guide was developed.

There was a lot of discussion about allowances for credit for the different types of system. But because no one had that system available at the time there was limited technical discussion about exactly what justified the factor of ten.

So what I was looking for is that you have considered it, you think it's appropriate for your design and you said that's how you determined. Thank you.

MR. LEE: Yes.

MEMBER REMPE: So if on the doses, do you ever have to consider, this is the unfiltered release is, the dose that's shown here for the control room, right? And do you ever consider that you've released radioactivity to the environment and you have to rely on the HEPAs and is there any challenge to the HEPAs?

MR. LEE: And challenge to the --

MEMBER REMPE: If they degrade is what I'm trying to say. Is there any, I mean is there enough radiation released? Are you relying, how much are relying on the HEPA filters in the analysis, none because of the type of analysis you're doing?

MR. LEE: As far as I understand your question you want to know about any challenges to satisfy the dose criteria for the EAB or LPZ?

MEMBER REMPE: No, actually I'm still focused on the main control room dose. And that is from the unfiltered release, right?

MR. LEE: No, no. That is from the --

MEMBER REMPE: The filtered release.

MR. LEE: In the LOCA -- for example, in the LOCA coolant accident, a loss of coolant accident at the time we have to consider it, and also direct. And many things should be considered to make that, this dose rate.

MEMBER REMPE: Okay, so when you calculate this how much do you rely on the filters? How much of a decontamination factor do you rely on the HEPA filters is what I'm trying to get to?

MR. LEE: In the MCR?

MEMBER REMPE: Yes.

MR. LEE: Ninety-nine percent. But I would have to check that.

MEMBER REMPE: Okay. So then pick one of these. If there's a dose of estimated of 2 rem or something then at that how much is coming, what would happen if the HEPA filters did not function, if they degraded, which would the dose increase is the question I have in my mind? What contribution is it to that whole dose for the main control room?

MR. LEE: Usually the filter, there are activities that build up inside the computer at the time the filter has a shine to the MCR worker. So we considered is that the filter it takes to shine.

MEMBER REMPE: Okay. So how much is the contribution of shine from the filter? Is it a large fraction?

MR. LEE: Can I check the DCD?

MEMBER REMPE: Just a rough idea. Is it a large contribution to the main control room dose?

MEMBER POWERS: Typically they're --

MEMBER REMPE: No, it's not, okay. Dana

has assured me that it isn't. So you don't worry about degradation?

MR. LEE: The total dose rate is at the millisievert, 46.9 millisievert total dose. But in that total dose I think the shine is at 12.9.

MEMBER REMPE: Okay.

MR. LEE: And so the design with the filter is located office door of the MCR. So the shining is a little bit higher than the other plants.

MEMBER REMPE: Okay. And if it, would it ever degrade? I don't know enough about how they operate that you would be concerned that it could increase.

MEMBER POWERS: Typically particulate is not a biggie. Your big dose is coming, as he says, you get a certain component from shine. But most of it comes from the noble gases or the iodine.

MEMBER REMPE: Okay.

MEMBER POWERS: Particulate, just because of the way the source term is arranged it's not huge for this 24 hour period. Now you get into the longer terms and it's a problem. But the HEPAs usually have a roughing filter in front of them and so they don't overload.

MR. LEE: Thank you, sir.

MEMBER REMPE: Thank you too.

MR. LEE: I would like to provide you a summary. APR1400 Transient and Accident Analyses of Chapter 15 demonstrate to comply with the requirements of the federal regulations and NRC regulatory documents.

There are 12 open items in total for Chapter 15 as described in next slides. These are the open items. Thank you for your attention.

MR. CHO: This is Sung Ju Cho. May I add some explanation about the SI RCP? In Tech Specs to one RCP or one shut down coolant pump for share in operating shutdown condition. And also allow operation without any RCP running for up to one hour.

So we assume that this event occur during this period over time because to maximize the primary to secondary side temperature difference maximized to maximize the primary and secondary side temperature difference. And also in Tech Spec, boron dilution operation is not allowed during this condition.

So we assumed the homogeneous boron concentration inside the ICS during SI event. So this is my explanation.

MEMBER MARCH-LEUBA: Okay, thank you.

MR. CHO: You're welcome.

CHAIR BALLINGER: Any other questions from the Members? We're a half hour behind but we should probably be able to make that up. So we are in recess until 1:00 p.m.

(Whereupon, the above-entitled matter went off the record at 12:01 p.m. and resumed at 1:00 p.m.)

CHAIR BALLINGER: We're back in session and the floor is the Staff's.

MR. STECKEL: Thank you very much. Thank you. My name is Jim Steckel. I am currently the Chapter PM for Chapter 15, and I've worked with the group that's presenting today since inception of the review.

Before we begin further, I would like to be assured that two contractor staff that we have that will be calling in as part of the presentation, that they are on the phone. Mr. Jim Servacious, are you there?

MR. SERVACIOUS: Yes, I'm here.

MR. STECKEL: And Doug Barber, are you there?

MR. BARBER: Yes, I'm here.

MR. STECKEL: Great. Thank you very much. We'll let you know when you're needed. I am moving to Slide number 2. These are the reviewers that have worked on this chapter for the SER. And you can see our consultants at the bottom.

We have two contractor support personnel here today that will be part of the review in person, and of course the two who are on the phone right now.

Here are sections we will present and the presenter names. And if there are no questions or other comments, I think we can begin. I'm going to turn it over to Mr. Shanlai Lu.

MR. LU: Great. Shanlai Lu from staff, reactor system, and good afternoon. As Jim mentioned that we had a lot of reviewer and then consultants as part of our team. And as a result probably today you're going to hear about 17 people presenting plus two on the phone.

So the one of the major difference between this design certification comparing with any other large reactor design certification is we have much shorter schedule. So therefore, that early on and that during the Phase I and Phase II, we put a lot of, we have a burst of resource spending.

So that's reason and this chapter becomes one of the most heavily -- chapter by mostly applicant and the staff. So before we get into the details section by section, I want to use two slides to give you staff's perspective about this design certification and also the approach we took to tackle specifically for the reactor system part of Chapter 15.

Okay, all right. So back to one and a half year before the DCD was documented it was docketed. And we had a QA inspection about to check that and see what does the application looks like.

And the Staff's first impression was oh, okay. It's called an APR1400, it's a design similar to the previously approved reactor design. It's actually a design evolved from System 80+ which was certified 23 years ago.

So it's really, by itself it's a mature design in terms of System 80+, for example, Palo Verde, the System 80 has been running. And then in addition, the KHNP has already completed construction of core unit three which is also the APR1400 and which has already been in, you know, operation.

So throughout this review, and it's further confirmed that we found we found that many system designs are either similar or even identical to those I would say System 80+. And the DCAD application is also similar to that of say System 80+ itself.

So however, give the credit to the KHMP, they did add more safety margins. We already covered for example accumulator and then additional ECCS trainings, and then the reactor vessel injection there.

So that's our perspective throughout the, during the past almost three and a half year review. So next slide. Okay. So let me talk about high level. What's the approach we took?

Okay, based on this initial assessment of what exactly this application is about, we decided to focus on the change that we have mandated or we're required to cover every single sections, every single design features.

But our focus of the resources is on the, has been on the changes that are implemented into the APR1400 design. And then we decided to also, because after 1995 we identified a lot of generic safety issues.

And then so we decided to also spend resources on the in-depth review on those safety issues identified after 1995. And then of course we have to then again provide overall coverage with the assistance of Staff confirmatory analysis on select areas. We needed to cover every area.

But we also asked our, assistance from Office of Research to perform confirmatory analysis on select areas like the, for example yesterday 9.1.1 was we were doing the criticality analysis based on the Staff's in-house calculation and the additional analysis performed by Oak Ridge.

So for today's presentation you are going to see three steps for confirmatory analysis by Office of Research. And then we, as part of the review we decided that since we have such shorter schedule comparing with any other previous design review, so we decided to identify the potential issues as early as possible.

For example, some of the issues were identified even during the before the submittal. We told them hey, solve this problem GSI-191. And the thermal conductivity degradation and a few seismic.

They paid up and then did take time for both sides to converge the specific solutions and the approaches. And then I think with that effort, we should be able to finish this application on time. All right, and we conducted quite a number of audits and on-site inspections, so we are going to talk about that one too. And then with those on-site inspections and audits, we can zoom in and focus on our RAIs. So we issue less number of RAIs comparing with any other design certification, at least from that perspective.

But it has been focused on, based on the audit inspection and the confirmatory analysis. So that's the overall approach in the review and the Staff perspective. Any particular questions for these first two slides, summaries?

Okay, so stepping into the details of 15.0 section, it's the title is Transient Accident Analysis. And the first part is about classification of the events. And as it is, because the similarity to the existing design and System 80+ and then so there is really not a whole lot of change there.

And then the Staff concluded that there was no additional issues, or any issues related to the classification. And plant characteristics, initial condition assumed in the active analysis which was spread into the different sections. And as part of summary 15.0 we found it's acceptable.

Trip system, engineered safety feature systems and analytical limit and delay times the same thing, was carried out by each specific sections too. The component failures, non-safety related system, operator actions considered in a safety analysis, especially the single failure, has been considered throughout by different scenario, different event.

Loss of offsite power and long term core cooling methodology for determining uncertainties and the thermal conductivity degradation. Those are the parts we did have some issues, so that's really I want to talk about it, even as part of it point zero.

So Staff finding in the, except the two open items we identify at this point, and that the entire section of 15.0 satisfied the relevant three requirements. So next page, we're going to talk about the two open items here.

Okay, fuel pellet thermal conductivity degradation, and we talked about that already. And I think it was first presented by our 4.2 in the, we identified as an RAI. And I think I do agree with Dr. Schultz' comments.

And then for new reactors, you've got me. So it's not a new phenomenon. We Staff and the industry knew that for a long time. Why still, you know, takes that long to fix it and then why the fuel reactors, I think that's really good comments there.

And you know, in addition to that actually the Staff identified this issue as part of Regulation 50.46. It's required. So when we identify this issue back to it, it's not came from KHNP because they just pulled it from the Westinghouse exporters Phase 3D methodology and it's part of technology transfer of three 80+ to them.

So it has been done there for a long time.

And then so based on what we learned at this point is okay, we actually initially identified this issue not from a Westinghouse or KHNP, from another fuel vendor.

And we found this is the industry wide issue. Not only that one, and our Staff approved that too. So we research the issue during the initial QA inspection and identified this issue. So KHNP actually did a thorough evaluation to figure out what's scope out or what is the impact.

So their initial submittal on the docket identified the fuel center line temperature increased by 550 degrees Fahrenheit for certain burnup level at hot spot. So that's just temperature is so large, and difference so large and then missed by FATES 3B code.

So we really, you know, decided to pay attention to this issue. And then the issue has been on its path to be resolved. So we had a lot of iterations, and you're going to hear from the Staff.

One of the reasons we also identified this issue in addition to 4.2 is because of -- as the TCD does have the impact on the initial condition, the central line temperature stored under, it has the impact on specific transient.

So when we identified this issue broadly at right at the beginning of 15.0, hopefully that any sections of transient would be also, we don't have to repeat the same open item, although you may see the SER mention about a particular open item.

So we can talk about that one in detail, you know, section by section and then why it's a still open item. But their actual resolution we plan to talk to you guys in December timeframe.

Okay, so if you have specific questions, hold on that one. We did have this one. We have not converged with the applicant yet.

All right, another open item we identified as part of the 15.0 is a boron dilution during a LOCA long-term cooling phase. And this is the standard GSI 185 issue. And this is required by 1.206, the regulatory guide for all the new reactors, they need to address this.

So they did address this, but we were looking into the next tier of information. There was no analysis performed. So we identified this issue as an RAI and then became an open item because at the time we finished SER this, the information had not been submitted.

But right now we are going to give you a preview of how we, and the issue is being resolved. Actually, KHMP have already presented and I think this issue by itself is technically speaking it's no longer an issue. But the documentation from the logistics perspective, we have not put in to SER.

Okay, so those are two open items related to 15.0. Any questions about this, comments? Then we are getting to the 15.0.2. Although this may be like a standard section. But this section actually covers the review of transient accident analysis methods.

The entire Chapter 15 used a lot of computer codes. A lot of them have been approved as part of the SEA system 80+, and it totals 17 of them. So that's the reason we spent quite a lot of resources, and then our consultant, Mr. Jim Servacious is on the phone. He actually did a lot of digging.

And then part of our review was to check although it's approved, we have the limitations on each specific computer code, what's the application range of the parameter whether it's applicable to this design. So that's the part to go through that part takes a lot of time.

And then Jim did his job. And the methodology wise then includes a non-LOCA safety analysis method, a large LOCA, what we wanted to mention about it is we are going to have a one day presentation on large LOCA topic in December I think. So are not going to talk about that one in detail.

So our LOCA, large LOCA side of DCD presentation is going to be one slide because really we are going to talk about it in December in details.

Small LOCA evaluation methodology, we also reviewed that one and it's applicable, the current license and basis of the methodology is applicable to APR1400. Post LOCA long term cooling evaluation including in-vessel downstream effects and we found overall it's the, you know, the plan is acceptable except that we have issues of boron precipitation, boron dilution, and then the resolution of GSM 191 has not been completed yet, although technically we don't see a problem.

So that's the reason we have reviewed applicability of this system responses, original approval, limitations, and the -- because all those codes have been made and they were developed, you know, 40, 30, at least 30 or 40 years ago.

So they were running on different QA system, the computer system back then. So we launched a specific inspection audit to check the QA record and to check to see whether the Windows, latest Windows version still worked. So we dive into that part, too. All right, next slide.

That's quick, all right. So we launched quite extensive audits to check the calculation reports and the QA records. And we did ask quite a few RAIs, and so far all of the RAIs at this point have been resolved.

The examples are provided here. HERMITE, radioactive term and then limiting pressure, RCOS code benchmark. I remember we talked about the ENDIVE4 with ENDIVE, compared with ENDIVE7 and why, how the issue was really, you know, we issued RAI and they came back with how they really resolved those issue by doing the benchmark against actually plant data.

So that's the part that we did ask the question. It's not something we let it go with different ENDIVE cross section library. But we found later that the cross section library has been used, even right now. The older version has been used by the current plant operating fleet too.

But with the benchmark, we found that it's acceptable for ROCS codes to be applied into APR1400. For COAST, the code friction and form loss coefficients, CESEC-III cold-edge enthalpy definition, we were just trying to understand some of documentation back to 20 or 30 years ago was not clear. Our consultant wants to know what's exactly the definition there.

Okay, STRIKIN-II, fuel temperature during steam line break event. So we also asked questions. So of those issues, we identified, we also identified REIs related to RELAP5/MOD3 to large LOCA. Those are application issue related to the use of the topic, approve the topic, oh, and the topical report to be approved.

MEMBER CORRADINI: May I ask a question?

MR. STECKEL: Yes.

MEMBER CORRADINI: This is Corradini. So I don't understand the last sentence where it says the concluded is pending review. So what we're looking at here today, your initial calculations on a methodology that you're still evaluating?

MR. STECKEL: That's correct. And as it is right now, because of the TCD issue and also the resolution of the REIs related to the large LOCA topic, KHNP is running another round of final set of the analysis.

MEMBER CORRADINI: So let me just ask a question. So it sounds like we're putting the cart before the horse. Are we going to see now a whole different set of quantitative numbers that we're going to have to look at again?

MR. STECKEL: That's correct. That's the

reason why I'm saying that conclusion is pending on the review of large LOCA topic. That's also the reason today we do not plan to give you a final conclusion of Staff's review on large LOCA analysis.

However, we do want to show you the confirmatory analysis we've performed to support the development of the REIs we had related to the large LOCA analysis. But you are correct that I think that as it is right now, if you look at SER or DCD, the numbers are subject to change.

MEMBER CORRADINI: Okay. So then let me ask the question differently. In all I've looked ahead of it, and I see a lot of the TRACE, what we'll call confirmatory or audit calculations.

MR. STECKEL: Right, right.

MEMBER CORRADINI: Which all indicate much lower values. Much is maybe, I won't put a qualifier. Lower values on peak center line temperature and associated figures of merit. So it's not your intention to try to understand why this is different, it's a matter of just making sure that you see a bounding number at this point and then approve their methodology to do a final set of numbers?

MR. STECKEL: That's correct. And not only that one. When we ran our initial phase of the trace confirmatory analysis at that time, it was weighted, that particular one we just want to make sure we have almost identical physical properties as a real RELAP5 input, whatever the use so that we can have a head on, an apple-to-apple comparison.

But however, after we identify the PCT issue and then we actually as part of the TRACE analysis, it's as part of the I think the backup slide. You can see there's a, we actually did the additional analysis to cover the TRACE analysis with the TCD addressed too. So that one, the PCT goes up much higher.

However, even as it is right now with whatever we calculated, even we performed, we try to use the TRACE to perform the bounding analysis to try to bound the TCD with that, and the PCT still way below 2,200. So it's really an issue of how it will be clearly stated and documented for Staff to approve.

MEMBER CORRADINI: Okay, thank you.

MR. STECKEL: Okay. Next slide. Any question for 15.0.2, and Jim is on the phone. And he generated all those REIs and reviewed this. If nothing, then we'll turn to Michelle.

MS. HART: Good afternoon. I'm Michelle Hart, I'm with the Radiation Protection and Accident Consequences Branch, and I'll be talking about radiological consequences of design basis accidents.

KHNP put that at the end of the presentation, we put it at the beginning.

MEMBER POWERS: Because it's the business end.

MS. HART: For whatever reason we put it here. So in SRP 15.0.3 is where we have the facts on doing these analyses. As Shanlai had said, this design is very similar to the designs we already have. So there's no real differences between the types of design basis accidents they'll look at.

And so I did review the source terms, transport and release of fission products. Core isotopic inventory I did do a confirmatory analysis to see if I believed the core inventory that they gave me.

I also looked at the coolant activity concentrations including the conversion to tech spec, dose equivalents. I had some help from somebody in the chemical branch on the post accident containment water chemistry management or the pH control in the water in the containment.

And then I also looked at the evaluation of fission product removal. We did talk about that some in the ACRS meeting for Chapter 6. And then of course we're evaluating the offsite doses at the EAB and the

LPZ and also in the control room, and the TSC radiological habitability. Next slide, please.

So the applicable guidance is given in the SRP 15.0.3 and also in Reg guide 1.183 which is also sometimes called the alternative source term. I did perform independent confirmatory analysis for all the design basis accidents that helped me develop if there were any RAIs, and like I said, for the core isotopic inventory.

Right now the version of the SER that you have has confirmatory items based on the REIs that I had asked. I cannot make any final final conclusions until it shows up in the design certification document revision. I have preliminarily reviewed the revision to the design certification document and all of those confirmatory items go away except for the steam generator tube rupture.

So those problems resolved. I do find that the offsite dose results are within the regulatory dose criteria for all the design basis accidents, and therefore it's acceptable. So those are resolved.

The control room in TSC results are less than five rem, however there is an open item, 15.0.3-1 which remains unresolved. So I cannot make a final finding on control room and TSC habitability until that open item is resolved. And I will talk about that on the next slide.

But what it was is one of my colleagues was asking some questions about the radiation monitors and the intakes for Chapter 14 for ITAAC. And we discovered in their response to that question that the control logic automatically reopens the intakes on a periodic basis during the event to redetermine which intake has the lower radioactivity, and then close that other intake so that you always have the lower contaminated intake open.

It's not something that I had ever seen before, and it wasn't really described in the other sections of the DCD, so I wasn't really expecting it. So I had some questions about whether the design basis dose analyses did cover for this small period of time when both intakes were open.

So that RAI question, it's in Chapter 14, 14.0.3.08-14 sub-question 6.b remains under review. But we have had discussions with the applicant and it looks like they're on track to get the response satisfactorily to where resolved.

The question under, they have provided some scoping type analyses to show that there is sufficient margin in their current analysis as it stands to account for this reopening of the intakes. It's just a matter of the thing that we're still talking about is the COL items, the COL applicant will have to make sure --

MEMBER MARCH-LEUBA: Just out of curiosity, this reopening of the intakes, is it once a minute, once an hour, once a week?

MS. HART: Well, okay, so that was the question that we had because it was unclear to us. And what the original COL item was is that the COL applicant would choose those times, both the intervals and the amount of time that the intakes are open.

And I said well we need to understand what you, the designer, think is an appropriate thing so that --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: What are you built in the analysis.

MS. HART: Right. And so the COL item would be you, the COL applicant, still pick this, but if it's outside the bounds of what would be covered by the dose analysis, you may have to re-do your dose analysis.

So their scoping analysis, or their sensitivity analysis assumed that it would be open on an hourly basis for a minute at a time. Now both the intakes are open, you know, does it really increase the amount of radioactivity? I mean, it's still being drawn through the filters.

There's some, you know, it's not really clear that it would have much of an effect on the dose analysis, but I think it was just mainly it was unclear what the assumption should be and how the system would really be operating, what the basic assumption for the design was.

So we're trying to clarify that we don't have the final resolution, we don't have the final RAI response. So we're just waiting on that.

MEMBER SCHULTZ: It's good to have it addressed because as I mentioned earlier, the reg guide was developed and it had particular parameters that were allowable if one had different types of intake systems.

MS. HART: Right.

MEMBER SCHULTZ: But because this one in particular hadn't, wasn't available at the time the discussions were held, there wasn't a lot of attention paid to how it would be implemented.

MS. HART: Right, and I was involved with that. So yes, I remember that. I think we were expecting that a system like this, because a system like this has been discussed in SRP 6.4 for quite some time.

It's theoretically possible that somebody out there could have one and I'm just not aware of it.

I think the thought was is that it would automatically isolate and then you would just stay with the configuration for the rest of the event. And so this idea that you would check periodically to make sure you were still at the lower intake concentration was not something that I think any of us thought of.

MEMBER SCHULTZ: I think original it was, it would be implemented automatically and then that gave particular tread.

MS. HART: Right.

MEMBER SCHULTZ: But the switching back and forth, I agree, is something that needs attention to provide some reassurance.

MS. HART: Right. And so, yes, we're just trying to make sure that we have a box around what the COL applicant needs to consider when they make those choices.

MEMBER MARCH-LEUBA: Right, but just a Member's opinion, don't be too hard on them because I'm looking at it from the dirty side, one minute an hour versus I've been there from the dirty side all the time, it's a penalty I'm willing to take.

MS. HART: Right. And I think it's clear

that they did have a reduction. They took some of the credit for that margin that instead of taking a factor of ten for the automatic isolation, they took a factor of eight.

So that, you know, is 20 percent of margin over top of what we would allow normally. So it's well covered by that. So I'm not as concerned about that. I think it's just making sure that the COL applicant knows what's going on and know what they need to do.

MEMBER SCHULTZ: Michelle, other aspects of the control room dose calculation, the question came up related to the chi/q. You said that you had reviewed those. Did you do any audit calculations related to that?

MS. HART: I am not the atmospheric disbursion analyst. We did have, we did evaluate of course the control room chi/q's and had determined that they had followed the guidance that, you know, is out there.

I did not do any particular sensitivities or anything like that around chi/q's. I think the, you know, we'll see the real answer when the COL applicant comes in and uses their real chi/q's to compare. These are kind of site parameters for lack of a better term.

So you just want them to be as reasonable as possible. They don't have to be specifically, you know, really, really correct or anything like that because we do have that opportunity when you actually have a site. But those must be the answer.

So they did have chi/q's for the intake and also for the unfiltered in-leakage. And the unfiltered in-leakage is, you know, 100 CFM total and the intake rate is I guess like 3,700 CFM. So a lot more is coming in through the intake. And the intake is filtered, and there's also a resurface filter.

MEMBER SCHULTZ: Thank you.

MS. HART: Are there any more questions? Well since there are about 17 of us, I have been told I need to get out of the way. I'll still be here, though.

MR. DRZEWIECKI: I am Tim Drzewieki. I'm in the Systems Branch of the NRO. I reviewed 15.1. And so this involved four events that were AOOs, among postulated accident, and steam line break.

So the event which is highlighted, which is the inadvertent opening of the steam generator ADV, this was identified as elevating AOO. These were evaluated using CESEC in order to get the NSSS response, and then CETOP-D using the KCE1 CHF correlation in order to evaluate thermal margin.

Staff did their own calculations in order to, well first evaluate all the parameters as well as trace calculations in order to support a finding that they had identified the remaining event. Next slide, please.

MEMBER MARCH-LEUBA: Not so fast. This question is probably higher than your pay grade, but I personally am so used to working with SAFDLs, specified acceptable fuel design limit, that I'm not sure where they come from. What we've seen is that the applicant is using this 20 kilowatt per foot LHGR SAFDL. How does that get reviewed, approved, accepted?

MR. DRZEWIECKI: Well, I do know where that's at. And okay, so where that value resides now is that they have a TCD tech port which they had evaluated the impact of thermal connectivity degradation on several aspects of their DCD.

And in there, that was where they came up with a value of I guess it's 20. So that value I believe --

MEMBER MARCH-LEUBA: Based on center line temperature melt?

MR. DRZEWIECKI: Yes, that's right.

MEMBER MARCH-LEUBA: Okay, so 20 is a surrogate for center line temperature melt?

MR. DRZEWIECKI: Yes, yes.

MEMBER MARCH-LEUBA: Because that's what I thought I've heard from people that 20 would melt the core. And indeed it does.

MR. DRZEWIECKI: Yes.

MEMBER MARCH-LEUBA: And is there a process, maybe it's more -- is there a process for accepting that?

MR. LU: Yes. I think as it is right now, 20 kilowatts per foot, the number at this point staff has not taken a position yet. It's open, it's still remain to be part, an open item as a part of a few topical approval review. And so we are still not there yet.

MEMBER MARCH-LEUBA: But the fuel topical will have an SER revisit 20?

MR. LU: Yes.

MEMBER MARCH-LEUBA: Okay.

MR. LU: Right. And then see whether it's --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Is that like a setpoint or something like that?

MR. LU: The reason they are using 20, I think they are using them as they are somewhat design, you know, they lean on that power density based on core center lines less than 20 kilowatts per foot, they're fine simply because they think that based on that number that it satisfy SAFDL actually.

MEMBER MARCH-LEUBA: And just, this would be a SAFDL for transient power? I mean, you have a 20 kilowatt per foot in a steady state? It won't be this close from it in a steady state? Will it apply only for transients?

MR. LU: Right, right.

MEMBER MARCH-LEUBA: Peak power.

MR. LU: Oh, of course, of course. I don't think that they are trying to design a core with 20 kilowatts per foot steady state power. They're far, far below that one, right? My understanding, right? It's like less than 13 kilowatts per foot.

MEMBER MARCH-LEUBA: Twelve, thirteen is normal.

MR. LU: Right. But for example the rod ejection case, they may have higher. That's one of the reasons that when the TCD, when we're talking about the rod ejection case, that particular part still has, remain to be an open item.

We are still working with them because they have not provided to us what's exact the final analysis based on TCD, what's the center line temperature will be.

MEMBER MARCH-LEUBA: Since you mentioned the control rod ejection, I'll save you time later --

MR. LU: Yes, that would be a confirmed analysis.

MEMBER MARCH-LEUBA: They mention 230 calories per gram. Am I mistaking that with high rise formation that there is a limit that is a function of burn up?

MR. LU: That's right, yes.

MEMBER MARCH-LEUBA: Or this doesn't apply.

MR. DRZEWIECKI: Well, okay, so a value still applies in terms of the fuel disbursal. Okay, but in terms of if you get a fuel failure, that's a much lower limit. That's burn up dependent.

MEMBER MARCH-LEUBA: They're applying it correctly?

MR. DRZEWIECKI: Yes.

MEMBER MARCH-LEUBA: They just put a simple number in the slide? I mean, in the slide they just said 230. They didn't say anything else.

MR. DRZEWIECKI: Two thirty, yes, that's right. Actually they were below, I believe, 60 which would be a high burn up limit.

MEMBER MARCH-LEUBA: Okay, thank you.

MEMBER SKILLMAN: Tim, I've got a question. The limiting event here is the either main steam safety valve or relief valve lifting. What consideration was given to a relief valve on your first or third stage feedwater heater, your high pressure feedwater heater where you get not only a great amount of heat removal from secondary due to the loss of the steam, but you also get a temperature reduction from your design for your final feedwater temperature.

As a consequence, the feedwater is cooler going into the tube bundle, and as a consequence, the moderator temperature coefficient, you can be adding reactivity.

So my question is is this really the limiting condition, or could there be a feedwater heater relief valve failure that could result in a slightly different and perhaps more serious transient?

MR. DRZEWIECKI: Well, I can say that this event, 15.1.4 was not the one that had the most amount of over cooling. It was the one they had identified as giving the least amount of thermal margin.

15.1.1, actually that was the event that resulted in the largest amount of reactivity insertion. However, it was enough of a reactivity insertion to actually cause a trip of the reactor.

And so this event, there was a rise to about 113 percent and it just stayed there. So when it got to manual trip, that was when they saw it eliminated the DNBR.

However, I believe our incinerator, as you described, would result in likely more reactivity insertion. However, I believe that could be bounded by the 15.1.1 event which has shown to be less limiting in terms of DNBR.

MEMBER SKILLMAN: Thank you.

MR. DRZEWIECKI: Sure. Okay, next slide.

MEMBER MARCH-LEUBA: So just on 15.1.1, you postulate a drop in fuel temperature. You're going to assume once you do it's a failure, right?

MR. DRZEWIECKI: Yes.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: They always overestimate how much the temperature drops.

MR. DRZEWIECKI: Yes. They had assumed that it would drop by about 100 F, and we had calculated that if you lost one train it would probably be more, you know, about 50 degrees or something like that, 40-some degrees. That was our calculation by hand. Yes, it was bounding.

MEMBER SKILLMAN: Thank you.

MR. DRZEWIECKI: Sure. Okay, so in evaluating these events, there's a table on the staff's MSER which we tried to go through find all the inputs, trips, all of that, find the value for that and state what the basis is for that value, why it's conservative.

The applicant did assume a bounding relief capacity for this valve that would bound the opening of any valve on the secondary side. They ran this with and without a LOOP or loss of offsite power, but the one that was bounding should have a LOOP.

And of course there was a feedwater cutback trip, but that was not taken credit for. There was only credit taken for operator action in order to cause a trip of 30 minutes into this event. And the results showed that they didn't violate SAFDLs, they didn't violate any of the pressure limits.

Okay, next slide. So now we're on the steamline break. They had two separate calculations. One was to maximize the amount of other activity insertion, and you would get post trip. And then a second calculation or series of calculations in order to try to minimize the amount of thermal margin that they had.

They used CESEC again. Again, they used a cold edge temperature for the feedback. And what this is, this is a way in order to calculate the effective moderator temperature such that you get a bounding low value for the feedback, or low value, the temperature used in the feedback such to get a bounding positive value for the reactivity insertion.

And of course they used CETOP-D for the thermal margin. And again, there's a table in the Staff's SCR in order to look at all of the inputs and the basis for those values.

MEMBER MARCH-LEUBA: And in this event, what's the consequence of the return to power, other than we've scared the operators to death in the control room, you return power to two percent, five percent power?

MR. DRZEWIECKI: Well, it would be short-term if it did happen. If it did not happen, this event, what happens eventually is that safety injection is going to put the boron into the core, and that's going to shut you down.

So usually what happens is what I've seen at other plants, you know, like at Palo Verde, there could be a rise in power up to maybe even a size, you know, ten percent.

Now for them, they would have to use a

separate code in order to calculate that because the pressures are so low that they're outside the range of the KCE 1CHF correlation. That's where that H-RISE code would come into play. But for them, but I didn't see that for any of their calculations.

MEMBER MARCH-LEUBA: Well, the primaries are still at full pressure, no? Is the primary --

MR. DRZEWIECKI: It's going to drop. It's going to drop, I believe the pressure drops because of the overcooling on the shrinkage on the primary side, pressure drops, like, 1,600 psia, maybe even less than that.

MEMBER MARCH-LEUBA: And I guess just, like, KC01?

MR. DRZEWIECKI: Yes, that's out of the range of KC01.

MEMBER CORRADINI: So this is Corradini. I guess I had a question just so I understand what you meant. So pressure would fall and I would be at modest power. So why would I be worried about CHF? Is it just simply that the correlation is not in its applicable range?

MR. DRZEWIECKI: So okay, yes. So you're worried about CHF if the power comes back. So if it cooled down to the point, you know, where you're critical again, then you want to make sure that your power is so low such that you don't violate CHF.

MEMBER CORRADINI: Right, but --

MR. DRZEWIECKI: Now for them -- yes?

MEMBER CORRADINI: But if I just might, let me just make sure I've got this qualitatively. But at these pressures, as pressure goes down, CHF rises. As it cools down, CHF rises. So and I'm at partial power. And we say it's as much as ten percent of full power. So I would be, the CHF ratio would be much larger under these conditions.

MR. DRZEWIECKI: Okay, all right. So it depends on the correlation. Now for this correlation and for these flow rates actually, if you lower the pressure in the system, you're going to get a larger critical, I'm sorry, you're going to get a lower critical heat flux. So it's adverse to your thermal margin.

MEMBER CORRADINI: Okay, all right. I guess I want to think about that because I thought, so is the flow decreasing significantly?

MR. DRZEWIECKI: Yes. Yes it is. Especially if you have a LOOP, or loss of outside power because your pumps are going to trip.

MEMBER CORRADINI: Okay, all right.

Thank you very much.

MR. DRZEWIECKI: Sure. Okay, so where was I? Okay, so there are two trips that are credited for this event. That depends if you have a LOOP or not. So if you don't have a loss of offsite power, then you take credit for the variable of over power trip.

If you do have a loss of offsite power, that causes the RCPs to be lost, and then take credit to the low flow trip. They take the single failure of diesel generator which then results in the loss of two trains of safety injection, and that's conservative because that is credited for keeping you shut down.

So if you have less safety injection, it's more challenging from the return to power standpoint.

And then again, there's credit taken of operator action of 30 minutes in. The results of this is that there was no post trip return to power, it stayed shut down. We also checked pressure temperature limits because this was a rapid cooldown, those were not violated, and they didn't violate SAFDLs.

MEMBER MARCH-LEUBA: Can you describe to me in 15 seconds what the variable of a power trip is?

MR. DRZEWIECKI: Yes. And this is described probably better in the SC. But what it has to do is that you have a trip based on your current power level. So if you are at a lower power, you will have a lower trip. Now --

MEMBER MARCH-LEUBA: So this is delta over the initial? The simplified way?

MR. DRZEWIECKI: Yes, yes. Now in terms of how it's treated in the safety analysis, it's really just at a fixed value. Now the way it's done in safety analysis is that, because if you're at full power, it's really just a set value.

So the nominal value is actually 109.6, but there's uncertainty on that. And that band goes down to -6.1 or the 103.5, and it will go as high as 6.9 or the 116.5. And that's why you see different values used in different analyses pending on what's conservative.

MEMBER MARCH-LEUBA: Thank you.

MR. DRZEWIECKI: Yes.

MR. HERNANDEZ: Yes, good afternoon. My name is Raul Hernandez, and I'll be presenting the decreasing heat removal by the secondary side. The Staff evaluated this seven events in order to understand the progression had identified which is the most limiting of them.

The analysis of this event must confirm that the primary, the RCS and the main steam pressure remain below 110 percent. And the fuel cladding integrity is maintained.

The Staff found that the most limited event is the loss of condenser vacuum, like the applicant stated in their design. Also, the Staff found that the applicant used conservative assumptions in order to demonstrate that the RCS and the main steam pressure remains below 110.

Also, the DNBR remains above the minimum DNBR limit, ensuring fuel Cladding integrity.

MEMBER MARCH-LEUBA: Raul, this 110 percent, shouldn't that be linked to some ASME service level, is it linked to a service level?

MR. DRZEWIECKI: It is. It's actually service level B, upset conditions.

MEMBER MARCH-LEUBA: Okay.

MR. HERNANDEZ: Any other questions in this section? Okay. For the feedwater pipe break, the Staff evaluated the feed line break event following the guidance of SRP 15.2.8.

The Staff evaluated the applicant description of the event and agreed that the most limiting event would be a large break between the feedwater line check valve and the steam generator.

The Staff found that using that approved methodology which we already discussed in section 1502, and using conservative assumptions, the applicant report will yield conservative results.

The applicant report analysis demonstrate that the NBR remains above the minimum value and ensures fuel cladding integrity, and also shows that the RCS and the main steam pressure remain below 120.

MEMBER MARCH-LEUBA: Okay, 110 or 120?

MR. HERNANDEZ: In this case it's 120.

MEMBER MARCH-LEUBA: Which is service level B and a half.

MR. DRZEWIECKI: It doesn't correspond to ASME service levels, I'll tell you that.

(Simultaneous speaking.)

MR. THURSTON: Yes, the feedwater line break even is a postulated accident. So it's 120 percent.

MEMBER MARCH-LEUBA: Your name?

MR. THURSTON: Carl Thurston, Reactor Assistance Branch.

MEMBER MARCH-LEUBA: Can you say again? Can you repeat that again?

MR. THURSTON: Yes. The feedwater line break is a postulated accident.

MEMBER MARCH-LEUBA: So that's --

(Simultaneous speaking.)

MR. THURSTON: So it's 120 percent of the design pressure versus 110 percent for AOOs.

MEMBER MARCH-LEUBA: Okay, thank you.

MR. HERNANDEZ: A preview of this section. Other question? That's it for this section.

MR. LU: 15.0 to 15.2.

MR. HERNANDEZ: Yes.

MR. LU: So we have next group coming in to talk about 15.3 and then 15.4.

MR. VAN WERT: Thank you and good afternoon. My name is Chris Van Wert and I'm with Reactors Systems Branch from the NRO. And I'm going to be presenting the Staff's review of the APR1400 analysis of the loss of force flow accidents.

So the Staff's review of the APR1400 evaluation model confirmed that the analyses were based on approved codes, that the codes were appropriate for the APR1400 design, and that the inputs that were used were appropriate in bounding of the plant conditions.

To assist the staff in this review, the Office of Research performed TRACE/PARCS confirmatory calculations, and very shortly, Dr. Yarsky will present the results of the confirmatory runs. But first you have to listen to me present the regulatory findings.

So as part of the review, the Staff made the following regulatory findings for the loss of force flow event. As mentioned in the previous slide, we found that the codes and methods were appropriate and that the inputs were appropriate, and that the analyses resulting from the use of these codes and methods demonstrated that no SAFDLs were violated, therefore demonstrating compliance to GSE 10.

The analysis further demonstrated that the RCS and the auxiliaries were not breached, therefore demonstrating compliance with GDC 15, and that the loss of offsite power on startup of BDGs demonstrated compliance of GDC 17.

And reactivity changes are reliably controlled, so SAFDLs are not exceeded, thereby demonstrating compliance with GDC 26. So without further ado, I'll turn it over to Dr. Yarsky here.

DR. YARSKY: Thank you. Good afternoon, I'm Dr. Yarsky from the Office of Research and I'll be presenting -- am I on? Sorry about that. I'll be presenting the results of our TRACE/PARCS confirmatory analysis for the loss of flow event.

The loss of flow event was selected for confirmatory analysis because it is the event that produces the change in minimum DNBR.

MEMBER REMPE: Excuse me. Before you get

into the results, could you talk a little bit about how you got the model to be similar to the APR1400 and the MELCOR calculations that were done for Chapter 19?

There had been some issues where erroneous geometry inputs, et cetera, were assumed at first and had to, did you start with the System 80 or how did you generate this model?

DR. YARSKY: I'm not able to talk about the MELCOR model, but I can talk about the TRACE/PARCS model.

MEMBER REMPE: But I want to know, yes.

DR. YARSKY: So the model that we're using for these calculations was evolved from the model that we developed to do confirmatory analysis for LOCA, which I believe as a starting point relied on the system 80+ and then incorporated information that we got from the applicant's own analysis.

So using the input doc that the applicant used to develop their models is a source of information for developing our TRACE systems model.

We can get into some detail about it, but something that's really interesting and more relevant to where we get to the part where we talk about rod ejection is that our TRACE/PARCS model is really interesting for APR1400 because we are doing a true 3-D calculation and that we're representing the core not only in three dimensions with our PARCS neutronics code, but we're also simulating each node and fuel assembly within the core with explicit Cartesian vessel component.

So there is that evolution on top of the LOCA model, and then we put a lot of detail into the core modeling to support the application to the rod ejection analysis.

MEMBER REMPE: Okay, thank you.

MEMBER MARCH-LEUBA: So the TRACE model is a 241 channel?

DR. YARSKY: The TRACE model is 241 assemblies, and then that's divided up axially. Yes. So the vessel component where we have a radial node for every assembly in the core.

The TRACE vessel model includes a radial node like one for each assembly in the core, so 241 radial --

MEMBER MARCH-LEUBA: And not only in the bypass, or what's it modeling?

DR. YARSKY: The --

MEMBER MARCH-LEUBA: The vessel model --

DR. YARSKY: The bypass I believe is modeled with six types. I'll double check that.

MEMBER MARCH-LEUBA: We can talk offline.

DR. YARSKY: Yes, the Cartesian vessel representing the core only represents the core, and we don't have a Cartesian vessel component representing the bypass. The bypass is represented with pipe --

MEMBER MARCH-LEUBA: Okay.

MEMBER CORRADINI: So, this is Corradini. Peter, or I should say Dr. Yarsky, so you've got to this level of detail because you want us to use the same model for the rod ejection, was that the point of this?

DR. YARSKY: Yes, yes. So we're using the same model for AOO as we are using for rod ejection.

MEMBER CORRADINI: Okay. So you got to this level of detail for that where you actually needed this level of detail for the rod ejection?

DR. YARSKY: Well, also one of our models to be able to simulate the feature of the APR1400 that allows a trip based on a sensed DNBR through the CPC, and a 3D model allows us to have that kind of capability.

MEMBER CORRADINI: Can you explain that again? I'm sorry. Can you go slower?

DR. YARSKY: Okay, this may be a bit too much detail because we don't actually use this feature. But in the APR1400, the CPC has a feature that allows for a trip based on DNBR.

So the process computer essentially evaluates a transient DNBR, and there's a feature of APR1400 that can initiate a reactor trip based on that DNBR evaluation.

MEMBER CORRADINI: Oh, okay. So --

DR. YARSKY: So when we developed the model, we wanted to have the capability in the TRACE/PARCS model to simulate that behavior if we needed to do a confirmatory analysis of that feature. However, that ended up not being required for this project.

MEMBER CORRADINI: Okay, all right. I think I get it now. And the last thing, since this is a Cartesian model on a bundle by bundle basis, what is the coupling in the transverse direction between the bundles? Is the coupling like a COBRA TF momentum coupling?

DR. YARSKY: No, it's not that detailed. We're just using the TRACE vessel component. So it's essentially like a loss factor in the transverse direction.

MEMBER CORRADINI: So like a --

DR. YARSKY: So similar to what's done for a multi-sector TRACE vessel model.

MEMBER CORRADINI: Okay, all right.

DR. YARSKY: So don't equate it to, like, a detailed sub-channel model.

MEMBER CORRADINI: Okay, okay, that's fine. I just didn't know what it was. Thank you very much.

DR. YARSKY: No problem. So for the loss of flow event, I wanted to step through the sequence of events before getting into our results. The event is initiated by a loss of outside power. The loss of outside power will cause the simultaneous trip of all the reactor coolant pumps, the turbine, and the reactor.

The reason why this is the limiting event from a DNBR perspective is the coolant flow decreases because of the coast-on of the RCPs. Eventually the DNBR margin is restored because the power decrease following the trip and the primary side reaches a natural circulation condition.

The RCS temperature initially increases because the loss of flow limits the transfer of heat from the primary to secondary side. However, once the auxiliary feedwater system is operating and the main steam safety valves are able to relieve secondary side pressure, the temperature will come down.

So if we go to the next slide, I'll present a series of figures comparing the TRACE/PARCS calculations to the results of the DCD for the loss of flow transient.

This first figure compares the relative core flow rates. For all of these figures, I tried as best as possible to put them on the same scale. So even though the units are different, the figures are sized so that it's pretty much one-to-one for the scale.

What you see here is, you know, essentially the relative core flow rate transient is dictated by the inertia of the pumps which is sort of an input to the analysis. So these match pretty much one-for-one.

If we go to the next slide, this presents results of the relative core nuclear power. These are very similar. The primary difference here is that the TRACE/PARCS calculation shows a more rapid decrease in the power compared to the DCD calculation.

This is attributed to two facts. The first is that the reactor trip is a little bit earlier by about a quarter of a second in the TRACE/PARCS calculation. This is due to in the DCD analysis, the reactor doesn't trip until the RPS is actuated based on low RCP speed.

However, in the TRACE calculation, we initiate the trip based on LOOP, but then delay it by the RPS delay time. So because of that slight difference on assumptions, the DCD being conservative, there's a difference of about a quarter of a second in the trip timing.

Additionally, TRACE/PARCS predicts a stronger negative reactivity insertion from the shutdown bank of control rods relative to what's assumed in the DCD analysis which results in TRACE showing a more rapid decrease in the power.

If we go to the next slide, we also compared the RCS temperature qualitatively. The responses are very similar except TRACE shows that the temperature is reduced a little faster. Oscillations in the temperature response are more long term. So after about two minutes or so, the oscillations seem to have a slightly higher magnitude in the DCD calculation.

This is really due to differences in secondary side pressure. So these temperature differences are related to the PARCS secondary side pressure changes in response to the MSSVs lifting and resetting to relieve secondary side pressure.

If we go to the next slide, this is where we compare the reactivity predicted by TRACE/PARCS to the DCD. As I noted earlier, TRACE and PARCS predict a stronger negative reactivity insertion of about 9.6 percent delta K by K compared to 8 percent delta K by K assumed in the DCD analysis.

So the stronger negative reactivity insertion is one of the reasons why the nuclear power decreases more rapidly in the TRACE/PARCS calculations. If we go to the next slide --

MEMBER MARCH-LEUBA: Now, is the TRACE/PARC reactivity compare favorably with the steady state shutdown margin calculation?

DR. YARSKY: This eight percent value is a value that's an analysis value in the DCD. When we performed our confirmatory calculations in the nuclear design, which I think have already been presented --

PARTICIPANT: That's right.

DR. YARSKY: -- I think the values of the shutdown margin are comparable. What we're showing here is in the 3D calculations we can't really specify the external reactivity applied by the control rods. We can only essentially, like, insert the rods and then PARCS internally evaluates their worth.

MEMBER MARCH-LEUBA: Correct.

DR. YARSKY: So in this calculation, we actually excluded all rods except for rods assigned to a shutdown bank.

MEMBER MARCH-LEUBA: So that was the question. Do you have the most reactive rod out, or in?

DR. YARSKY: This also has the highest

worth rod stuck out. So this is a N-1.

MEMBER MARCH-LEUBA: You're comparing apples to apples?

DR. YARSKY: Yes. So it's shutdown rods minus the strongest worth rod. We'll talk about this a little bit more when we get to the rod ejection. But because we're doing the 3D evaluation, we can't, we don't have the flexibility to pick the worst combination of reactivity factors.

So we selected a point in cycle that was the most limiting point in cycle. But this point in cycle also has with just the shutdown bank with the N-1 worst rod stuck out, still has a stronger amount of total negative reactivity insertion compared to the assumptions made in the Chapter 15 analysis.

So this is indicating a conservatism in their analysis. But I mean, they're kind of close. It's just the TRACE is predicting more reactivity insertion.

MEMBER MARCH-LEUBA: Yes, theirs is probably more situation dependent because you picked --

(Simultaneous speaking.)

DR. YARSKY: I think it's more realistic compared to developing a conservative assumption for the Chapter 15 analysis and the DCD. So this slide here, like I said earlier with the RCS temperature, this is really tied to what's happening on the secondary side in terms of pressure.

TRACE/PARCS predicts similar to the DCD this pressure oscillation. However, we predict a slightly lower pressure and a more frequent lifting and resetting of the MSSVs. We attribute this just to differences in the input assumptions to MSSV lift and reset pressures, and initial pressure on the steam generator. However, the qualitative behavior is very similar, and they're pretty close.

If we go to the next slide, in this instance I wasn't able to put the slides on the same scale, and I want to stress that they're not on the same scale. In TRACE/PARCS, like I've eluded to earlier, we have the capability of evaluating something like a DNBR.

This minimum DNBR is based on evaluating critical heat flux derived from the KCE1 CHF correlation, processing that through a control system and then developing an assembly specific DNBR.

However, this is based on assembly wise thermohydraulic conditions. So we don't' want to say it's exactly comparable to DNBR. However, we think that this approach allows us to compare the trend or the delta and DNBR over the transient.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Go ahead, Mike.

MEMBER CORRADINI: No, I'm sorry. I didn't mean to interrupt you, Jose. If you want, go ahead.

MEMBER MARCH-LEUBA: Okay. The applicability, you put KC1 to TRACE or was it the post processing?

DR. YARSKY: What we have is a control system in our TRACE model that allows us to calculate the critical heat flux according to the KCE1 CHF correlation. So it's being done by a control system. And then based on --

MEMBER MARCH-LEUBA: Did you develop a proprietary version of TRACE --

DR. YARSKY: No.

MEMBER MARCH-LEUBA: No, good.

PARTICIPANT: It's the input.

MEMBER MARCH-LEUBA: It's an input then that you can take away and --

DR. YARSKY: Yes, so the control system can be removed from the deck, and the correlation is not implemented in TRACE.

MEMBER MARCH-LEUBA: Good.

DR. YARSKY: Okay, so but I want to just

kind of sort of put this small caveat on it is we're not doing a sub-channel calculation, even though we have this great detail in the corner able to get assembly by assembly flows and temperatures. We're still not evaluating at the sub-channel level.

So we think that this gives a good indication of the trends in mDMBR, but I wouldn't say that's the value of the mDMBR even though we're using the applicant's correlation and control system.

MEMBER MARCH-LEUBA: Because with that --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Sorry. Your turn. Go for it.

MEMBER CORRADINI: Okay. So I think, I just wanted to, I'm not sure where Jose, but he got two of my three questions. So I'm kind of curious about, though, the general shape because TRACE is imbedded CHF correlation in the lookup table.

I would expect the lookup table, using that to get the same qualitative shape of this. Is that a fair statement?

DR. YARSKY: Yes, I think that is a fair statement.

MEMBER CORRADINI: Okay, and then the second part of my question would be you said sub-channel, but you're modeling each bundle. So you're looking at this on a bundle by bundle basis? That's what I didn't understand, I'm sorry.

DR. YARSKY: Yes, so we are not using a sub-channel method. And because we are not using a sub-channel method, even though I have a figure labeled mDNBR, I want to stress that that's not the predicted mDNBR. This is a figure where we're evaluating the CHF correlation according to assembly wide parameters as opposed to how it should be, or how it's intended to be used which is on a sub-channel basis.

MEMBER CORRADINI: Okay.

DR. YARSKY: Because of that, the absolute value of the mDNBR, I don't feel is accurate. However, we feel that it provides valuable information in terms of the trend, or the change in mDNBR over the transient.

MEMBER CORRADINI: Okay, thanks. Thank you very much.

MEMBER MARCH-LEUBA: So at a minimum did you use peaking factors for this CHF?

DR. YARSKY: Yes. So what we have is in the assembly heat structures representing the core fuel assemblies. We have a supplemental heat rod with a high peaking factor that bounds the peaking factors presented in Chapter 4.3 of the DCD.

MEMBER MARCH-LEUBA: So you used the peak power but the average flow?

DR. YARSKY: Right. It's, we think that it can indicate trends, but because we don't get to the sub-channel level, I don't want to say that our value represents the actual mDNBR.

MEMBER MARCH-LEUBA: And another thing to consider, if I remember correctly. CTOP is a sub-channel model but is very simplified. And it is made to overpredict what TORC does which is the real sub-channel model. I mean, if you were doing it right, CTOP would be conservative because it overestimates, right?

DR. YARSKY: I'll let Tim answer that. I'm familiar with --

(Simultaneous speaking.)

DR. YARSKY: I'm not familiar with what the applicant did.

MR. DRZEWIECKI: Yes, that is right in the sense that it's not accurate to say that, you know, if you're familiar with VIPER or COBRA, it's unaccurate to say that CTOP is like a VIPER or a COBRA. TORC is more like VIPER or COBRA, and then they have a simplified model which actually works inside the CPC as well, which is C top D. And it's a simplified model. It runs faster but it is tuned so that you get a result which is conservative versus TORC.

DR. YARSKY: Yes, so if we -- I just wanted to say here what we show in the TRACE/PARCS calculation is that the initial mDNBR transient is very similar showing about a 0.1 decrease in mDNBR for the first about one and a half seconds.

That's when in the TRACE/PARCS calculation we predict the mDNBR transient turning around and mDNBR increasing. In the DCD analysis, the mDNBR continues to decrease until reaching its minimum of about four seconds.

We can go to the next slide. So found in our confirmatory analysis is that the TRACE/PARCS calculation and the DCD analyses compare very favorably in terms of the major trends and the overall system behavior. We notice some small differences in RCS temperature and steam generator pressure, but we think these are relatively minor.

With our confirmatory analyses, we've shown that the assumptions made for the reactivity insertion from the shutdown banks is conservative relative to a more realistic prediction of the reactivity insertion with shutdown banks.

We also in our confirmatory analysis have shown a milder mDNBR transient compared to the applicant's calculations indicating that there is some conservatism on how they evaluate the mDNBR during the transient.

And lastly, we think that their analysis seems to reasonably predict the system behavior and to conservatively predict the thermal margins. And that's all I have on the confirmatory analysis.

CHAIR BALLINGER: Okay, we're within three minutes of a scheduled break, so I would like to take that. So come back say at 2:27. We're in recess.

(Whereupon, the above-entitled matter went off the record at 2:12 p.m. and resumed at 2:27 p.m.)

MEMBER STETKAR: Okay, we're back in session, as promised. You've got to hit the button and make it green. There you go.

MR. VAN WELT: I told myself I was going to remember that.

Again, this is Chris Van Welt. And I will be presenting now the staff's evaluation of the APR1400 analysis of reactor coolant pump malfunctions. There are two events that are contained in this overall section. The first one is reactor coolant pump seizure, which is the limiting event. And the second one if the reactor cooling pump shaft breaks.

And, okay, again kind of similar to 15.3.1 for the loss of forced flow. The evaluation model as presented here consists of the approved codes. The staff confirmed that they were appropriate for this design. And we also looked at the inputs as presented in DCE Table 15.3.3-2, and also in methodology CENPD-138 -- or 183-A, and confirmed that these assumptions were appropriate for this analysis.

And although it says on the slide here regulatory findings for loss of force flow, that was supposed to be regulatory findings for RCP malfunctions.

And, again, similar as the last section, since we found that the closer methods and inputs for all are appropriate and we found them to be acceptable, the analyses, the resulting analyses demonstrated that loss of offsite power, at least to automatic -- which demonstrates compliance with GDC-17.

GDC-27, the compliance with GDC-27 is demonstrated by showing that the operator can achieve full check during the event.

And compliance with GDC-31 is demonstrated by making the RCS pressure within 110 percent of the design pressure.

Because of fuel failures from this possible accident are bounded by the assumed values and dose consequence analysis that was presented previously by Michelle.

If there are no further questions -- if there are no questions, that will be the end for 15.3.

MEMBER MARCH-LEUBA: Just out of curiosity, don't we have a SAFDL on DNBR which presents the last -- I mean there won't be any failures for RCP malfunctions; right?

MR. VAN WELT: Well, there is a SAFDL for the operation -- well, yeah, 1.429.

MEMBER MARCH-LEUBA: So as long as, I mean this is run as an AOO for every load.

MR. VAN WELT: Right.

MEMBER MARCH-LEUBA: And we don't expect fuel failures. And, of course, you meet 10 CFR 100.

MR. VAN WELT: Right. Well, this one is a postulated accident. And per the DC --

MEMBER MARCH-LEUBA: Oh, this is a PA?

MR. VAN WELT: Yeah. This one is a PA. There is -- let me confirm that this is a non-prop number here. But there is a number presented in the analysis, yes, less than 7 percent failures. And that's less than the assumptions used in the dose consequence analysis.

MR. LU: Yeah. We did not conclude on all of those numbers. You see the numbers from this slide because they're non-proprietary.

MR. VAN WELT: Just to confirm, that was a non-prop number that I just mentioned.

MR. LU: Oh, really? Oh, okay. All right.

MR. VAN WELT: Just in case you were curious.

Any further questions or?

MR. LU: Okay, I will cover 15.4.1 to 15.4.3

This section was -- these three sections were reviewed by Matt Thomas. He is not here to give the presentation. He was not here yesterday either. Fortunately, we have our senior consultant Doug Barber on the phone.

Doug, are you on the phone?

MR. BARBER: Yes, Shanlai, I am here.

MR. LU: Okay. All right. So Doug provide the technical evaluation report to the staff and then Matt finished this part of SER.

Okay. So these three subsections covered uncontrolled CEAE withdrawal from subcritical or low power startup condition; uncontrolled CEAE withdrawal at power; control element assembly misoperation.

All right, let's go to the next slide.

So, I think the staff and the consultant reviewed the causes of those three particularly possible events. And then we confirmed that the applicant considered all the process of event and consistent with typical large PWRs for all the scenarios described within those three sections.

And the methodology, you know, were reviewed by 15.0.2 and then they used the approved methodology to analyze this one. For all three subsections we did issue RAIs. And then at this point it's closed, we are closed now.

And we find that the limiting event progression and the sequence were identified and analyzed. The conditions, under conditions core parameters are adequate. Proper conservatisms were applied.

The input assumptions under these three subsections crossed all single failures, loss of power, trip delays. And also staff found the consequence of that's reading the SAFDL criteria. Those are the terms right now. And system response are acceptable.

And DNBR, heat generation rate meets the SRP acceptance criteria at the margins. That's what staff found; right?

RCS pressure design limits are met with adequate margin, too.

Next slide.

So the results here is and the arguments about and based on we actually audited once through the electronic reading room and Doug and also Matt found that the approved methods were properly used. And the parameters and applicability were confirmed.

Except one item, still open item is related to the PCB. And the analysis right now found in support of those all three subsections and Matt considered the thermal conductivity degradation.

And the KHNP is addressing this issue to evaluate what's the impact on those, all the conclusions of those subsections.

At this point what we observed is for all the events described at 15.4, section 4.1 through 4.3, SAFDLs are not exceeded using conservative assumptions with considered uncertainties. And the general design criteria is 10, 13, 17, 20, and 25 are met.

I think that's the subsections, three subsections Matt was supposed to present. And Doug's here if you guys have any questions about these three sections. They're confirmed to analyze it down to in support of those three sections. Presented this one right after another two, one or two slides.

Any questions for now for Doug? Doug is on the phone.

MEMBER MARCH-LEUBA: My brain works well right after the break.

What I was asking Chris about was the flow and SAFDLs.

MR. LU: Okay.

MEMBER MARCH-LEUBA: I didn't realize that you were talking about also offsite power with loss of flow. So what situations do you have that event under LOOP, our generic loss of offsite power? Because when you lose power you also lose flow.

MR. VAN WELT: Right. The question, well, when you have the CEAs you get lower flow, get a trip. And that leads to the turbo trip and then the assumption of this is offsite power at that point from then on.

MEMBER MARCH-LEUBA: So if something is a trip, causes a disturbance in this yard and --

MR. VAN WELT: That's right.

MEMBER MARCH-LEUBA: So that's why that's a bad, bad event.

MR. VAN WELT: Right.

MEMBER MARCH-LEUBA: Is the loss of offsite power, not the loss of flow.

MR. VAN WELT: Right.

MEMBER MARCH-LEUBA: Thank you.

MR. LU: Okay. So if there's no other questions for 15.4.1 to 15.4.3, then we'll move on to the next slide.

Who's -- You are covering this one; right?

MR. DRZEWIECKI: Yes.

15.4.4. on the startup of inactive RCP. This is -- so there's no real code calculations here, just simple hand calculation. Industry had bounding values through the ITCs or the actual thermal temperature coefficients up to bounding values for temperature difference between the primary and secondary side with positive and negative. And they would calculate, had the assumption if you started the pump there would be a change in the temperature for the primary side to the secondary side instantaneously.

Verified that with conductivity assertions that you would get that you'd stay subcritical, that you had, you know, a shutdown margin.

These calculations were only done in Modes 3 through 6 because in Modes 1 and 2 you have to have all of your pumps running.

Any questions?

MEMBER SKILLMAN: Yes, I do.

What I find curious about this -- and of course KHNP used the same words you used -- you can't have, you can't be in Modes 1 and 2 with this condition. Well, actually you can.

The way this tech spec is written, it's two RC loops shall be operable in an operation with two coolant pumps in each loop. I've got that, I've got four pumps, I've got two loops.

But if I drop a pump for whatever reason -- motor stops, I've got an electrical something -- this tech spec doesn't preclude me from restarting that motor. It tells me what the operability conditions are supposed to be and it gives me six hours to be in Mode 3. So if I'm the shift supervisor I've got 360 minutes to get back to Mode 3 in which I can do something.

So my question is, per the assumptions could there be a situation where the reactor operator, the panel operator says, I'm going to try to restart that motor? He's got three pumps running. He's got one loop running backwards, partially backwards, through the cold leg loop.

MR. DRZEWIECKI: Yeah.

MEMBER SKILLMAN: So couldn't he actually, or she, go ahead and try to restart a motor?

MR. DRZEWIECKI: I would expect that after operating at full power and he lost a pump that you would probably trip out.

MEMBER SKILLMAN: You'd probably go out in flux flow.

MR. DRZEWIECKI: Yeah.

MEMBER SKILLMAN: I'd agree with that.

But supposed you didn't. Supposing you were in startup and you're 35 percent power, you're coming up slowly, you're escalating, --

MR. DRZEWIECKI: Yeah.

MEMBER SKILLMAN: -- and you drop a pump, for whatever reason?

MR. DRZEWIECKI: The statement that I can say there is that you'd be outside the bounds of your safety analysis. So you'd be in a condition that you haven't analyzed yet. If they have sufficient, you know, of analyses to show they could do that, it might be all right.

But, but based on the analyses that they presented here, Chapter 15, I couldn't, I couldn't state --

MEMBER SKILLMAN: Yeah, I can't either. That's why I'm asking the question.

MR. DRZEWIECKI: Yeah.

MEMBER SKILLMAN: The strength of this

argument seems to rest on this tech spec 3.4.4.

MR. DRZEWIECKI: Yeah.

MEMBER SKILLMAN: But 3.4.4 does not communicate don't start. It says, be in Mode 3 in six hours. And I find that interesting because, as an ex-operator, I could say I could start that motor. Nothing prevents me from starting it.

MR. LU: Let me just, let me try to address this one.

I was over the plant, too. And I looked over one of the -- at Vermont Yankee actually.

MEMBER SKILLMAN: Operating what?

MR. LU: I was over Vermont Yankee and I worked in support of the operation of the plant.

MEMBER SKILLMAN: Oh, okay.

MR. LU: During the startup one of our reactor pump suddenly stops, that's a big event.

MEMBER SKILLMAN: Oh, absolutely.

MR. LU: It's so big I would say that it would be unlikely for the operator to say, okay, let's proceed and to start up and keep going and restart the pump. There is something, you know, very unlikely. But I think this information also --

MEMBER SKILLMAN: I concur with you, with your on-the-fly assessment.

MR. LU: Right.

MEMBER SKILLMAN: But what I'm trying to communicate is the strength of the argument rested on this tech spec 3.4.4. And I'm saying 3.4.4 does not stop me from starting an idle motor.

MR. LU: Okay.

MEMBER SKILLMAN: It drives me to be in Mode 3 within six hours. Those are two different things.

MR. LU: Okay, I got you.

MEMBER SKILLMAN: See what I'm saying?

MR. LU: We'll take a look at that one.

MR. DRZEWIECKI: Yeah, the one thing you have to look at is that if I'm, you know, if I'm in Mode 1 or 2 and I lost a pump, I would expect that I'd get a trip. But we need to verify that. Maybe add some of that later to the SE to help strengthen the finding.

MEMBER SKILLMAN: Thank you. Thanks.

MEMBER STETKAR: According to the reactor trip log, if any one of the four pumps have a speed less than what the speed setpoint trips to --

MEMBER MARCH-LEUBA: Any one?

MEMBER STETKAR: Any one of the four.

MEMBER SKILLMAN: Results in a trip.

MEMBER STETKAR: Reactor trip, yeah.

MEMBER MARCH-LEUBA: I was noting there that you are two, that has extended situational loops and they were doing maintenance. And there was one that was shut down. And the guy was supposed to go work on that one and he went to work on the other one that was working and he pulled the card from the working one. So he says, Oh, crap, the moment he did it because he heard -- No, it was not a trip. There was no trip and they restarted the pump.

They did think that he was on an inspection team which I was part of.

MR. LU: That's a good comment. We'll take a look.

MEMBER MARCH-LEUBA: Thank you.

MR. THURSTON: Good afternoon. My name is Carl Thurston. I'm going to present the staff's evaluation of 15.4.6, inadvertent decrease in boron concentration.

So we had two open items remaining. Item 1, questions our conservation dilution times used to predict a complete mixing for Modes 4 and 5, with one shutdown cooling pump in operation. The applicant has provided mark-ups of tech spec LCL changes that include locking off the pump, locking off boron sources of the pump via a valve, valve closure.

And we found their response to be acceptable, acceptable for that, but they propose to close is in a non-safety system. So we're in the process of working with DNRAL to write a response for them that we will send to KHNP.

So, next item --

MEMBER SKILLMAN: And what are you going to tell them to do?

MR. THURSTON: Well, we're going to advise them that the non-safety sys -- the non-safety valve is not going to be acceptable. So we will have to find another mechanism to lock off --

MEMBER SKILLMAN: Thank you.

MR. THURSTON: -- the boron sources.

MEMBER SKILLMAN: Thank you, Carl.

So the second open item, questions that CEA withdrawal event which credits the VOP, variable overpower trip, down to slow event like boric dilution. So that RAI went out.

And we did get a response back from the applicant in August. And the response, so they propose to use a CPCS DNBR trip or other CPCS auxiliary trip like cold temperature range or primary pressure range which would protect the DNBR for the boron dilution event.

So the latest response that they sent us seems to be acceptable, except the staff has not finalized our, our evaluation. So, we're still --

MEMBER MARCH-LEUBA: So the event here is that you're slowly affecting your activity, so you're slowly lifting the power?

MR. THURSTON: Right.

MEMBER MARCH-LEUBA: But you're on very low power, but you are operating on very low power so you're not, you're not tripping a high power.

MR. THURSTON: Right.

MEMBER MARCH-LEUBA: And the VOPT is resetting this average as it moves, so it doesn't trip either; right?

MR. THURSTON: That's what we're -- that's our opinion. So they're proposing to use --

MEMBER MARCH-LEUBA: So what were the backups?

MR. THURSTON: Temperature range.

MEMBER MARCH-LEUBA: Why would that change?

MR. THURSTON: Because you're heating up.

MEMBER MARCH-LEUBA: But you would heat a scram? Will -- I mean is there simulation or calculation that the scram will happen?

MR. THURSTON: Yeah, well, it should be a slow, a slow warm-up for the RCS that this trip will --

MEMBER MARCH-LEUBA: All right.

MR. THURSTON: So we think that it's better than VOP trip.

MEMBER MARCH-LEUBA: Okay.

MR. LU: I think that's the part I mentioned yesterday as part of 9.3.4, the chemical and the volume control system.

MEMBER SKILLMAN: CVCS.

MR. LU: Yes. If for this their final resolution of the first open item, really the conclusion becomes that the source valve needs to be isolated during Mode 4 to 5. And the tech spec should be changed and then corresponding that valve needs to be monitored, controlled through the control room. So whether there's going to be safety, safety grade of the valve or not safety grade of the valve, so that the safety needs are still pending. And that just captured what we talked about yesterday.

So that's why we want to add a response here.

The issue is not really comes from that we have a problem already. It's really they analyze this boron dilution, potential boron dilution event in Mode 4 and 5. And if the fresh water source is provided and then there's potential to manage the contaminant.

And the analysis provided at this point was based on the analysis of CFD. But we felt that maybe the input may not be conservative. I mean that the final results will be still acceptable or not, so but we just changed the input. And then we found that to get it to a very accurate or, you know, high confidence, conservative result becomes difficult because of the system itself. And we assume that the complete mixing of the fresh water with the entire RCS loop before it gets into the core. And that just becomes unrealistic.

And does not mean that you really turn on the valve and then we have the boron dilution, may still survive. But right now it's to have this issue completely resolved within this time frame. And we felt that maybe the way to go is just to isolate that one. Secure that valve, there is no boron dilution event. What you, you know, by test facts.

So, so that's where we are. It's an open item at this point.

Okay? All right, thank you.

MR. THURSTON: Any questions?

MR. LU: Next item.

Okay, I will cover that one. You are the lead and now you haven't covered anything.

Okay, 15.4.7, inadvertent loading and operation of a fuel assembly in an improper position. And at this point the application has procedure controls, including fuel assembly I.D. verification process once the core is complete. That's part related with the fuel loading.

The application surveillance plan, and then we have the peaking factor occurred taking the specification 3.2.2 every 31 EFPD. I don't know who's -- Okay.

The piece on the technical misloading peaking factor increase is bounded by the peaking factor, increased from the CEA drop event analysis. So at this point the applicant claimed that the DNBR limit is not violated. And the staff at this point agreed with this.

All right. So that's just one slide I think somebody -- it's probably this one.

Now we go to the actual another confirmed three analyses.

MR. DRZEWIECKI: No. This is actually the review of the CEA.

MR. LU: Okay. That's your section.

MR. DRZEWIECKI: Yes. CEA ejections, there's actually three separate analyses here. Okay, the peak RCS pressure, looking at the CHF and DNBR analysis, did a technical issue at all looking the fuel enthalpy.

This evaluation model is actually, it's a methodology that dates back to 1976, the CE methodology. Some things have changed, of course, because I think they used a different tool back then. It was from PDQ, now it's ROCS. But it's a combination of using the 3D kinetics just to really get, or core design codes to get your pre-and post-peaking factors as well as you activity insertion. And then you would use your point kinetics, your CESEC-III and your STRIKIN calculations.

So and then in order to calculate the number of failed fuel pins that you get, it's not a simple case of if you violate the SAFDL and you failed fuel. They have, they have a method which goes back to the actual loss of flow methodology for CE in which they determine a response surface such that if your DNBR is a certain value, then you have a certain probability of seeing fuel, failed fuel. And they add that up by doing a pin census.

MEMBER MARCH-LEUBA: Is it like the kind of temperature limits? I mean you can violate DNBR for a microsecond and it will never burn. So there has to be some time involved; right?

MR. DRZEWIECKI: They don't include time. It's really based on the uncertain -- the largest driving factor here is the measured-to-predicted values on the CHF correlation. So taking that you're taking the values of, you know, other uncertainties associated with CETOP and all the response circuits by running like 200 or 300 cases like that and trying to vary these things, trying to determine that, you know, if my -- even though my staff felt it was 1.29, if on that 1.29 that's my, you know, mean value, or something like that for I only have a certain probability of having a failure.

And that was found acceptable in part by what you just said because it was, it was deemed to be conservative back when they approved this in the early '80s because of what they had learned. There's a NUREG-0562 in which we took rods and had them NCHFed and saw how long they had to be there in order to actually get fuel failures.

So that's kind of the basis for the approval then. And it's carried forward here.

Next slide, please.

So, again, staff created a series of tables to go through and look at the parameters for these series of calculations and to determine the basis for those. They're suitably conservative.

They do take credit on the RPS trip on the variable overpower trip. But that trip also includes, it includes a set point of -- there's a penalty because, of course, if you eject your rod you power shape is going to shift. And so your detectors have to include a decalibration factor.

And so they calculated what the decalibration factor was going to be. And they took the response from the worst responding exploited piece.

We also considered a loss of offsite power and if operator action is taken at 30 minutes. The results they presented so far showed that all the fuel failures were associated with a violation of DNBR. There were no fuel failures so far that associated with a violation of fuel enthalpy limits. However, that's the one which is on, which is on the most sensitive connectivity degradation. And that's why it remains an open item.

In terms of the peak pressure they showed they stayed below I believe 120 percent, which was, which is a limit for this event. And we don't expect that to change as a result of connectivity degradation.

Any other questions on CEA ejection?

Okay, so Pete's going to talk about some of the work that Russ has done to help us in this area.

DR. YARSKY: Thank you. I'll be presenting work that was primarily done by Dr. Andrew Bielen from the Office of Research who couldn't be here today. Hopefully I can cover it competently.

The first slide here, I want to spend some time talking about the methods that we used for our confirmatory analysis for rod ejection. We touched on these topics earlier during our discussion of the loss of flow AOO. But they're more relevant here for the analysis of rod ejection.

The first is, as I'm sure you're all familiar with TRACE/PARCH is that PARCH allows us to do a 3-dimensional kinetics calculation which is very relevant to a rod ejection if it's the highly-localized event. However, in this model we also have a detailed 3-D core thermohydraulics model using a vessel-in-vessel capability in traits to model the core, as we said, with 241 thermohydraulic channels, also coupled to the 3-D PARCH calculation.

This is I would say the most realistic picture of what occurs during a postulated rod ejection accident because we have a significant amount of detail and electronics modeling as well as our thermohydraulics modeling.

If we go to the -- I also wanted to mention here before we go to the next slide that we analyzed two specific points in the cycle, in the beginning of cycle and end of cycle. Because these were two potentially limiting points based on the combination of delayed neutron fraction and the feedback parameters.

So we go to the next slide.

Before getting to the results, I wanted to take some time to do a quick comparison between what is represented in the applicant's analysis versus what's represented in the staff's analysis. The applicant's method is based on a point kinetics representation. And this methodology allows the applicant to bias a number of parameters.

And to give just one example, for instance, moderator temperature coefficient has its smallest magnitude at the beginning of cycle. So that's you most limiting point in cycle from the standpoint of moderator temperature coefficient. However, delayed neutron fraction tends to be smallest at end of cycle. So your most limiting point in cycle from the standpoint of delayed neutron fraction is end of cycle.

Using TRACE/PARCH, which is a realistic methodology doesn't allow us to mix and match those conditions. We can either analyze the beginning of cycle or analyze the end of cycle, even though the particular parameters are not the minimum or maximum most conservative for overall.

The applicant, however, can mix and match. Take, for instance, the beginning of cycle moderator temperature coefficient and combine it with the end of cycle delayed neutron fraction. Using that kind of general approach for all of the kinetic feedback parameters on our point kinetics model allowed the applicant to develop what we looked at as a significantly conservative methodology for the evaluation of rod ejection.

And looking at these factors, items that are conservative relative to TRACE/PARCH for a more realistic confirmatory calculation include the input from the worth of the ejected rod, the reactivity feedback coefficients in terms of moderator temperature and Doppler feedback, as well as delayed neutron fraction.

The resulting power transient from the point kinetics calculation is then fed into a thermohydraulics analysis which, as Tim mentioned earlier, uses what's called these post-rod ejected peaking factors, which is another conservatism in the applicant's method.

In TRACE/PARCH this is, again as I

mentioned, a realistic 3-D calculation which means that the ejected rod depends on the thermohydraulic conditions and point in cycle where we actually eject the rod in our simulation, and doesn't allow us to combine sort of the worst case in terms of all of these different variable reactivity feedback parameters.

So with that understanding, if we go to the next slide I can talk about some of the conclusions from our confirmatory analysis.

We found that the applicant's analysis includes significant conservatism when compared to a realistic analysis. We talked about this in terms of the ejected rod worth, the delayed neutron fraction, and the reactivity feedback parameters.

To list an example, for the hot zero power, the applicant assumes an ejected rod worth of $1.08, which is over $1.00 which can leave to a prompt reactivity feedback and a significant increase in power level.

However, using a TRACE/PARCH realistic analyses at both beginning of cycle and end of cycle and ejecting a population of potentially limiting rods based on the power-dependent insertion limits, we found that the maximum ejected worth using realistic assumptions is about $0.26. As a result of this, there's a significant difference between the staff's confirmatory analysis and the applicant's reference analysis where we showed tremendous margins.

Our TRACE/PARCH calculations show the power increasing by a factor of three because of the low reactivity insertion, compared to the applicant's analysis showing the power increase back eight orders of magnitude.

MEMBER MARCH-LEUBA: A factor of three versus ten to the eighth?

DR. YARSKY: Yes. From zero power.

MEMBER MARCH-LEUBA: Sure. But still.

DR. YARSKY: Yeah. It's really a different -- the analyses diverged significantly because the applicant can force a reactivity insertion over a dollar based on their method.

MEMBER MARCH-LEUBA: Yeah.

DR. YARSKY: So even in TRACE/PARCH, trying to find the worst rod to be ejected at the beginning of cycle or end of cycle, if we adhere to the power-dependent insertion limits, there isn't a rod that can be ejected that inserts a dollar of reactivity.

MEMBER MARCH-LEUBA: Okay, now this philosophy, but we can do conservative calculations, it's perfectly acceptable, if you know where did we do the conservatism. But if the applicant believes that the hot rod is worth 1.08 and you calculated .26, as we usually say, one of the two is an error, and probably both.

And being there by a factor of four, I can see a ten to the eighth because it's critical.

DR. YARSKY: Right.

MEMBER MARCH-LEUBA: You're going from critical and that is that.

But miscalculated overall by a factor of four, that deserves a little scrutiny on the nuclear numbers.

DR. YARSKY: Just to be clear, this isn't that the applicant miscalculated the worth of the rod. This, these parameters that go into -- essentially parameters of their safety analysis, are generally selected so that they're bounding of any potential future cycle or core design. So what you can do is you can say from my nuclear design calculations performed in Chapter 4.3 for the initial core or the equilibrium core, I may have this is a maximum ejected rod worth based on those core designs. But go to Chapter 15 and then increase that worth so that your safety analysis has additional margin.

And what this would allow is for future

cycle designs. If the rod worth were slightly greater than what's represented in 4.3, this safety analysis would still be bounding.

MEMBER MARCH-LEUBA: But 400 percent margin on your rod worth deserves calling the guys out on 4.3, which are at the table anyway, and do that, make sure that somebody's not messing up royally. 4 percent on the rod worth is too much.

MR. DRZEWIECKI: One thing I wanted to add is that for this evaluation, you know, over the $1.00 reactivity insertion, they didn't calculate that they only get $1.00 reactivity insertion, it's that if you don't insert $1.00 and you're at hazard power, not much happens. And so you boosted it up, you know, just artificially in order to get a prompt pulse.

MEMBER MARCH-LEUBA: It is okay to do conservative calculations when you say, hey, my results were completely relevant and I can handle even 1.08. But my best estimate was .26. But as long as they did that, but if they calculate one way, then their nuclear methods are suspect.

DR. YARSKY: No, the 1.08 is a conservative bounding input assumption.

MEMBER MARCH-LEUBA: It's worthwhile to go back to 4.3 and see what numbers they actually calculate, to make sure. Because this is really -- I mean, as you said, maybe the 1.08 is not a calculated rod worth, but they run a sensitivity, 1.06, 1.07, 1.08. At 1.08 they decide to stop.

DR. YARSKY: Yeah.

MR. DRZEWIECKI: So, we did calculations as well that were looking at 4.3, you know, which is our results. And that was the best estimate. And we were must closer. We were, you know, pretty on the line there.

I don't have the exact values in front of me. That was presented, I believe, I'm not sure, I believe at Chapter 4 ACRS. But that was presented there and I thought it was a lot closer.

MEMBER MARCH-LEUBA: What method is used? Did you use CASPR to simulate, or what, what did they use for physics, what section?

MR. DRZEWIECKI: Oh. They use ROCS and DIT.

MEMBER MARCH-LEUBA: Oh, okay. So their own, their own system.

MR. DRZEWIECKI: Yeah. It's a CE methodology.

MEMBER MARCH-LEUBA: And we don't -- do we ever get a PATHS model for this, a model for this? We don't have --

DR. YARSKY: Yes.

MEMBER MARCH-LEUBA: We do?

DR. YARSKY: Yes.

MEMBER MARCH-LEUBA: So, you could --

DR. YARSKY: We did this. We presented these results to the subcommittee for 4.3.

MEMBER MARCH-LEUBA: Yeah, I don't remember.

DR. YARSKY: But the, for instance, the shutdown worth comparison and the bank-by-bank worth comparison were in good agreement between PARCS/PATHS and the applicant's analysis in 4.3. So, there was no indication at that point, but the methods were diverging in terms of the prediction of rod worth.

MEMBER MARCH-LEUBA: Yeah. And with the shutdown event you also believe --

DR. YARSKY: Right.

MEMBER MARCH-LEUBA: -- 9.6 was -- So this is an outlier.

DR. YARSKY: The 1.08, as we said, is artificially imposed in their method. There isn't a calculation of an ejected rod that has this worth. They, in their method, imposed in a point kinetics model this degree of reactivity insertion. So there isn't a nuclear design calculation up front that says the rod in location 27, if it were ejected, would have this worth. This is an imposed assumption.

And then this magnitude is such that if you compared it to Section 4.3, all of the worths are bounded by this assumption.

MEMBER MARCH-LEUBA: But the DCD or FSAR, whatever the document, it seems there should be a affirmative calculation that says the number we used is conservative because --

DR. YARSKY: Yes.

MEMBER MARCH-LEUBA: -- they would calculate this.

DR. YARSKY: Exactly.

MEMBER MARCH-LEUBA: Yes. Does that thing exist?

DR. YARSKY: Yeah, it does. It says that they calculated a value of like, you know, \$0.90. But, you know, multiplied by 1.2 so you get up front a critical response.

MEMBER MARCH-LEUBA: So this made .9 as opposed to use .26.

DR. YARSKY: Yeah.

MEMBER MARCH-LEUBA: I mean it merits asking a question.

DR. YARSKY: If there are no additional questions, that's all I have to present on rod ejection.

MR. DRZEWIECKI: So that's all for 15.4. So we move on to 15.5.

Our first event, this is inadvertent ECCS actuation. This was, this was not evaluated in any kind of code in civil hand calculations. Or, actually, it was the qualitative type of evaluation.

They ran three cases. If the RCS is above the SI pump shutoff head, if it's somewhere in modes like 3 or 4, and then also if you're, if you're in a lower mode on the LTOP system.

And of course, you know, staff went through and identified all the inputs from their basis and their values.

Next slide, please.

Okay. So if you're in Modes 1 and Modes 2 you're above the shutoff head of the SI, and so there's no impact up on the RCS.

So, if you're in Modes 3 and 4 and you're cooling down, okay, there's actually two scenarios if you're cooling down and you're heating up. What if you're, if you're outside the range and it should be on the LTOP system, then your pressure limit is actually higher than your shutoff head of the SI pumps. So there's no challenge to your pressure limits there.

If you're in these lower Modes 4, 5, and 6 and you're on the LTOP system, then the relief capacity of the LTOP system is much larger with the inside of ejection capacity. So there's no challenge there.

So 15.5.2, this is a malfunction of the CVCS that can increase the inventory of the RCS. Against that had gone -- Oh, okay. This was evaluated using CSEC-III and classification KCE-1 CHF correlation. I guess staff had gone through and tried to evaluate all the input parameters and to ensure they had a basis that was conservative.

There was an operator action that was taken at 30 minutes. They did assume the head had a loss of offsite power with reactor trip. And it didn't take any credit for pressurizer heaters in order to have -- in order to maximize inventory in the RCS.

What they had shown is there wasn't any challenge to the SAFDLs. DNBR 1.5177. So not very challenging in that perspective. Not very challenging from a peak pressure perspective.

They did not do an overfill analysis of the steam generator. And so this is where staff actually had some questions. And so what they were able to show to us during an inspection is that the POR -- sorry, the

POSRVs all qualified for water and 2-phase mixture passage.

So during inspection we had looked at their design spec for that valve and I was able to verify that. Therefore, we determined it didn't have to actually evaluate these consequences.

So that's actually all for 15.5. That's a very short section.

Move on to 15.6. The first event, 15.6.1. Now, in the SRP this is an AOO. But they evaluated it as a postulated accident. So staff had some questions on that.

Now, it's actually, now this is evaluated by KHNP as part of a small break LOCA. And so we issued an RAI. And they had responded that there was no single operator action or spurious signal that could cause this event because of this tech spec here. If they lock out one of the valves, it would have to open in order to have a spurious signal.

What it is is that POSRVs they have an emergency depressurization function, if you would, to lose the heat sink. And so they would pass a feed and bleed using these. But to have that scenario set up you have to take the action of actually to close a breaker in order to get power to these valves and to actuate that sequence.

Next slide, please.

Okay, the next one is the failure of small lines carrying coolant outside of the containment. This was identified as a letdown line break. This was the largest line. Staff had gone through and verified that that was consistent with table in the BCD 6.2.4.1.

They did evaluate this using CSEC and CETOP. Again, staff had gone through and evaluated all the parameters to verify they were suitably conservative. And the results showed that there was a change in the pressurizer level, but that was about it.

This event, it didn't really have much of an impact on the response of the NSSS system. It just kept going. And so a half an hour was actually taken in order to trip the reactor.

Next slide, please.

Okay, steam generator tube rupture. This again has two separate analyses to look at the thermal margin to see if you would fail any fuel at all, and then to look at the radiological consequences.

This was ran with CSEC and CETOP again. Again, staff had gone through, cleared a table. Had identified parameters and made sure they had a basis. They ran it with and without a loss of offsite power.

But the one with the loop is more conservative. They assume that the NSRVs are going to close instantaneously on a high steam generator level. And that is done so that the safeties are going to open up so you get more of a steam release, which is worse with those consequences standpoint.

Next slide, please.

So, the results are that there is no violation of the SAFDLs, so no fuel failures. No challenge to the peak pressure. There was no steam generator overfill that occurred during this event. And that the mass leak through the break is consistent with the value used in the dose consequence analysis for steam generator tube rupture.

Staff had determined that that leak was suitably conservative to use in dose consequences calculations.

MEMBER STETKAR: Tim, why was there no steam generator overfill? I was looking at something else. The operator action stops it?

MR. DRZEWIECKI: I believe so, yes.

MEMBER STETKAR: That's the only way --

MR. DRZEWIECKI: It took a half hour, yeah.

MEMBER STETKAR: Auxiliary feedwater on this plant is not isolated on high steam generator level.

MR. DRZEWIECKI: Yeah. Yeah, so a half an hour they have to identify it, one, and then --

MEMBER STETKAR: Yes.

MR. DRZEWIECKI: That's it.

MR. LU: Thank you. Before we jump into the LOCA section. I had LOCA, you know, and in the LOCA section we have a lot of slides, about 30 slides. So there is one slide about ATWS.

Jim, do you want to come over. This is supposed to be 6.8. It's after the LOCA but I, since there is only one slide, if Jim will cover that one then we can get into lots of LOCA slides, too, because they're still pending for review. And then we are going to -- it's going to be under these contractors will give a one-day presentation about the LOCA.

So maybe just go through those few slides then if we want to take a break we can take a break. Then we'll finish the remaining of the LOCA set.

CHAIR BALLINGER: Let's just keep going.

MR. GILMER: Good afternoon. I'm Jim Gilmer with Reactor Systems.

We touched briefly this morning with Member Stetkar's questions on the diverse protection system in Chapter 7 conclusion that satisfies the ATWS Rule

50.62.

I should mention that the DPS is virtually identical to that in the system 80-plus plants. So obviously approved in an upgrade.

The Section 15.8 is a technical report which did a detailed evaluation basically re-running all the cases submitted by Combustion Engineering in CENPD 158, which was provided to the agency back during the ATWS rulemaking. And it led to the special requirements for Combustion Engineering plants. So that's part of the 50.62.

The applicant basically concluded the same thing that Combustion Engineering did, that the limiting event was found to be the loss of normal feed without a turbine trip. Even the Combustion Engineering tests resulted in a reactor coolant system overpressure which exceeded the service level C. And that's what led to the special ATWS requirements.

For this section, because the Chapter 7 reviewers concluded that the protection system meets the intent of the ATWS rule, and the Chapter 19 actually does the Level 1 PRA for all the ATWS events. So there's more discussion in the beyond design basis section in Chapter 19 for ATWS.

So, therefore, for this section staff

concluded that it's acceptable.

Any questions on this?

MEMBER MARCH-LEUBA: In this event in this plan boron terminates the event? Is boron assumed to work or this also fails?

MR. GILMER: There was no, no mention of credit.

MEMBER MARCH-LEUBA: So the event never terminates? I mean, how do you get out of ATWS?

MR. GILMER: What turns it around? We haven't dug in detail into the analysis but I presume it's because --

MEMBER MARCH-LEUBA: Well, what does the protection system -- I mean in a PWR your rod still can serve and then boron terminates the out. Here, what event will you use enough negative radioactivity so you are out of the, the woods?

MR. GILMER: Well, I assume it's just the negative --

MEMBER MARCH-LEUBA: Yeah, and you continue to operate at the same power forever? That's, is that the assumption? Which is probably okay, I mean as long as you don't lose anything else.

MEMBER STETKAR: Does the diverse protection system work successfully here?

MR. GILMER: In the analysis or?

MEMBER STETKAR: In the analysis.

MR. GILMER: Yes, sir. I believe it does.

MEMBER STETKAR: The diverse protection system stops it.

MEMBER MARCH-LEUBA: Okay. So you use control rods and boron.

MEMBER STETKAR: Yes. It opens the reactor breakers alternately.

MR. GILMER: They did make a conservative -- they made a conservative assumption that there is a mechanical failure of the rods to drop. So your diverse protection system --

MEMBER MARCH-LEUBA: So then you still have --

MR. GILMER: Yeah.

MEMBER STETKAR: Jim, I didn't read the ATWS analysis. Let's be clear. In their analysis does the diverse protection system shut down the reactor or not?

MR. GILMER: I believe that it does.

MEMBER STETKAR: And how does it do that? Does it drop the rods or does it not drop the rods?

MR. GILMER: It initiates the signal to drop the rods. But there's certain, many rods do not drop. If the rods do not drop because of some mechanical part

MEMBER STETKAR: the reactor does not ever get shut down. Their injection pumps will not inject at the higher pressures that you're going to get. You establish some sort of meta-stable state with auxiliary feedwater flow and some reactor power that is controlled by temperature and Doppler and secondary heat removal.

I mean I, I didn't look at their ATWS analysis, so I don't -- Can't pull it up quickly here.

MR. GILMER: Well, and we didn't dig in detail into it because basically they were re-bench marking the analysis that was previously done by Combustion during the rulemaking.

MEMBER STETKAR: You can't get boron into the pressure down because they don't have high pressure injection pumps that will dead-head against the safety valves.

MEMBER MARCH-LEUBA: Yes. But you can't get boron but you can still get some regular boron control.

MEMBER STETKAR: From where?

MEMBER MARCH-LEUBA: I don't know. They need boron control.

MEMBER STETKAR: Right. You can't get

boron in.

MEMBER MARCH-LEUBA: Yeah, it would be worthwhile to find out what happens or it's supposed to stay there for the next 60 years. Eventually you will burn up the U-235; right?

MR. GILMER: Right. That was definitely a good question. And we'll take it back and I think we should address it from there.

MEMBER MARCH-LEUBA: Assume the contingency there was some way what you do.

MR. LU: Let me add another point. I don't think that we need to take action at this point.

They have performed this analysis, actually they, of course, did not assume that the unit's diverse scram system would work. Okay. So actually if the power goes on and then, you know, the pressure goes on. Then so actually has really reached the pressure limit. But that's the beyond event basis.

For ATWS scenario by itself seems this, because of this analysis demonstrated there is a need for scram. So scram is required. And then because of the requirement there's no need to worry about the continuous power of the reactor for 60 years. That's not possible. This is number one.

Number two, the APR1400 design, we just

endorsed the same, the diverse scram system. And the ATWS rule, and they do have like a diverse scram system. And they confirmed that in Chapter 7 the Division of Engineering I&C Group, that's its function, and acceptable. So they can take credit of having a diverse scram system.

If that satisfies the requirement from the ATWS rule and also that's the same level of safety requirement we have imposed to the Combustion Engineering plant operating fleet. So, therefore, staff does not believe there is anything we need to pursue in terms of this section.

MEMBER SKILLMAN: I need to ask. I'm looking at the safety evaluation.

MR. LU: Yes.

MEMBER SKILLMAN: Section 15.8.6 on Chapter 15-239. And the conclusion is because of the open items the staff cannot yet conclude that the APR1400 design meets the requirements of the ATWS rule.

And that's contrary to what is on that slide. It's on your SER, page --

MR. GILMER: Yeah, those open items were Chapter 7 open items. Thermohydraulically we don't have any open issues on this. So it's just being contingent on the final conclusion for the Chapter 7.

MEMBER SKILLMAN: Well, that may be. But I'm reading out of your Chapter 15 conclusion. And your Chapter 15 conclusion is that the staff cannot yet conclude that the APR1400 design meets the requirements of the ATWS rule. That's on page 15-239 of your safety evaluation.

So I think that there's a difference between what's in your safety evaluation and what's on your slide.

MR. GILMER: You have a good point. That really is open until we finalize the Chapter 7.

MEMBER SKILLMAN: So I think your slide 52, the last bullet is premature.

MR. GILMER: I would agree, yeah. A good catch.

MEMBER SKILLMAN: Thank you.

MR. GILMER: Any other questions on this section?

Okay. Okay, back to 15.6.5, large break LOCA. As you heard earlier, we're going to have a separate session on the topical report review of the realistic evaluation model.

One thing I was going to point out is that I believe will be a Thermohydraulic Subcommittee meeting, so logistically we may need to work out the like a joint APR1400 site works committee meeting. Because typically these topical reports would be presented to the TH Subcommittee.

MEMBER CORRADINI: This is Corradini. Can I ask when that is scheduled? I was under the impression --

MEMBER STETKAR: Why don't we do that offline because we don't need the staff, the NRC staff to tell the ACRS how to organize subcommittee meetings.

MR. GILMER: Yeah, I think it won't be until November. It's not currently scheduled, sir.

MEMBER CORRADINI: That's all I wanted to know. Thank you.

MR. GILMER: Okay. So because we're going to get into more details in a separate session, the only things we're going to talk about here are issues that were specifically identified for DCD Revision 0.

And one is -- well, these are sort of generic questions. So the first one was on loss of offsite power, single failure, and the limiting single-failure assumptions. And under that, then whether or not the reactor coolant pump trips were -- we were not exactly clear which resulted in the limiting peak clad temperature. Ultimately that was just resolved by a sensitivity study in response to RAI.

Similar question on control element assembly insertion, whether with insertion or without resulted in peak clad temperature. And the sensitivity study demonstrated that no insertion resulted in what's sought by PCT.

And the third question was regarding the safety injection tank check valves, whether active or passive failure. And KHNP took the position that they did not need to consider a check valve failure. But the staff's concern was, in particular, stuck open -- or stuck closed, rather, check valve preventing injection.

And we sort of answered our own question in investigating that, number one, their design with regard to the safety injection check valves is not that much different than conventional PWR, Westinghouse, or Combustion. And we typically have not required them to analyze their reasons why. And one is related to the SECY paper 94-084, that there were certain exceptions where designs similar to the safety injection systems were postulated failure of the check valve was not required. But there were certain other stipulations that the failure probability was less than 10 to the minus 5.

And at the time of the question we didn't have the Level 1 PRA completed, but now do. And the probability is certainly less than that value of failure.

Another reason is the surveillance requirements for the check valves for KHNP it's 120 times over the design life of the plant, 60 years life. So that would be essentially twice per year.

So given the probability type and the surveillance requirements, staff concluded that this was not an issue here.

Another item --

MEMBER SKILLMAN: Jim, let me ask, is there any other place where you deem a check valve an active failure?

MR. GILMER: For LOCA or surveillance?

MEMBER SKILLMAN: I'm scratching my head here because for my many years check valve was always considered to be passive device that you could depend on.

MR. GILMER: That didn't fail.

MEMBER SKILLMAN: That's in the B&W design, the Westinghouse design, the Combustion design, and the GE design.

MR. GILMER: Right.

MEMBER SKILLMAN: So perhaps you've said enough that this is not an issue now. But I'm just curious why it ever would have become an issue?

MR. GILMER: Well, it's not an issue because of the extremely low failure probability. But --

MEMBER SKILLMAN: Well, Hallelujah, you've got these passive valves everywhere. I mean it seems like a precipitous direction to go given 100 plants in the country that basically use check valves.

MEMBER STETKAR: I can say this because San Onofre Unit 1 is shut down now. They have five of them fail to open. Check valves can fail occasionally.

MEMBER SKILLMAN: It's not very often. And it is not often.

MEMBER STETKAR: It is not often but they do occasionally fail. So, you know, whether they're passive or active, they do occasionally fail.

MR. GILMER: Yes, that's a good question. And I, I don't know the status of this as a generic issue here. I think it was in the past. And I'm not sure exactly how the agency closes that generic issue. We'll have to look into that.

MEMBER SKILLMAN: I am not suggesting any further action. Thank you.

MR. LU: I think the conclusion on this point is that a check valve, active or passive, is not considered as part of what's needed for large LOCA analysis. And the issue resolved.

MR. GILMER: Okay. Continuing then. The members may have observed when looking at the DCD is that there is a surprising result for one of the curves in particular. 15.6.5-13 there is an unrealistic power spike which is totally unexpected for a light water pressurized water reactor. So the staff questioned that as an RAI.

And KHNP basically concluded that there is an error in the input of moderator temperature coefficient reactivity table in the RELAP codes that's being used for large break LOCA.

So this would apply not only to the ON6 double-ended guillotine case that was presented in that figure, it would apply to all of the spectrum of breaks. And that is being corrected. And the runs that are currently being done to find a resolution of all of these, so there will be new curves generated for the TCD.

Finally, because of the review is not complete, mainly because of the thermal conductivity issue and some other issues that were identified during the review that are being corrected, the talks are currently ongoing, so that's why we won't be ready to have the detailed presentation until November.

The other issue besides the thermal conductivity that we had a lot of discussion was on the number of statistical random samples that would satisfy the 95/95 criteria. The original Rev. 0 they used 124. But because they were reporting the third highest PCT, we determined that they really needed to run 181 cases to meet the 95/95. So that's also being addressed in the revision.

So it will result in another revision to the DCD which I guess will be Revision 2.

MEMBER CORRADINI: So this is Corradini.

I'm not exactly sure then why we're reviewing any quantitative calculations at this point, given what you just said.

MR. GILMER: Yes. We are not reviewing quantitative at this point either. So we would not expect the members to do any detailed look at it at this point.

MEMBER CORRADINI: Okay, fine.

MR. GILMER: Any other questions on this?

Okay, moving on then.

MR. LU: Shall we move on? Or do you want to move on to LOCA? I think we will have quite a lot of presentations here, including a confirmed analysis. So loop seal formation clearing and it is a big section.

So we have Syed and -- Dr. Syed Haider and also Dave Caraher and maybe Dean Prelewicz.

CHAIR BALLINGER: We're scheduled for a break at 3:45.

MR. LU: 3:45. So you want to do that now?

CHAIR BALLINGER: So, why don't we take that now.

MR. LU: Right.

CHAIR BALLINGER: And come back at five till 3:00.

(Whereupon, the above-entitled matter went off the record at 3:40 p.m. and resumed at 3:45 p.m.)

CHAIRMAN BALLINGER: All right, we are back in session. I don't know who's next, but whoever it is.

MR. HAIDER: Good afternoon. My name is Syed Haider. I'm a reviewer at NRO for the APR program and the small break loss of coolant accident methodology.

We also have with us David Caraher from ISL who is with the contractor.

Today I will present the status of the staff review regarding the evaluation of the small break LOCA as it relates to Section 15.6.5 of the APR1400 DCD.

The applicant also submitted a technical report to describe the APR1400 small break LOCA evaluation model details. Experience shows that for pressurized water reactors the most challenging transient for peak cladding temperature is always in limiting small break LOCA due to loop seal formation and the potential core uncovery.

I would like to emphasize that the present review was confined to the short-term thermohydraulic response of the reactor systems during the initial phase of the small break LOCA until the peak cladding temperature has occurred and the core has recovered and is covered with 2-phase mixture and the loop seals have been closed.

For the chill core reheat phenomenon due to later reformation of the loop seal will recover and the post-LOCA long-term cooling for both small and large break LOCAs, that will be presented later today by Dr. Shanlai Lu.

The present slide highlights the four aspects of the APR1400 small break LOCA safety evaluation that the staff focused on. The objective was to ensure that the APR1400 design complies with the 10 C.F.R. 50.46 acceptance criteria for emergency core cooling systems for light water reactors. That essentially means that even in case of the most challenging small break LOCA, the peak cladding temperature would not exceed the 10 C.F.R. mandated 2,200 degree Fahrenheit safety limit.

The staff first assessed the application of the Supplement 1 small break LOCA methodology and its four computer codes used for the APR1400 design. The SM-1 methodology and the use of the computer codes were approved by the NRC in 1977 for the Combustion Engineering ABB design. Continued use of S1-M methodology was also approved by NRC in 1986 for meeting the TMI requirements.

The applicant also showed that the Supplement 1 methodology used for APR1400 predicts more conservative PCTs than its Supplement 2 variant that was approved later by NRC in 1998.

So, this methodology has been reviewed about three times.

The S1-M methodology uses two computer codes CELASH 4AS and Compact 2 to model the system's hydraulic response for blow-down and reflood (phonetic) cases, while STRIKIN-II and PARCH are used to model the hot rod cladding temperature.In this backdrop, the staff also reviewed various modeling assumptions used for the small break LOCA analysis.

The staff also spent considerable effort on reviewing the applicant's modeling of the safety certification involved in small break LOCA, such as the initial loop seal formation and clearing, core cooling, the peak cladding temperature, or PCT.

My presentation will also cover the acceptability of the applicant's small break LOCA spectrum analysis.

Next slide.

This slide illustrates APR1400's conceptual design of each loop that can also help explain one of these main small break LOCA concerns in this review about the loop seal formation at the coolant pump section.

Starting from the right and going to the left of the slide, the zone of concern includes the cold leg initiating from the coolant pump start, the pressure vessel, the hot leg, the steam generator, and the intermediate leg that runs from the steam generator outlet to the coolant pump inlet.

The tube part of the intermediate leg between the steam generator and the coolant pump is called the loop seal. In case of a limiting small break LOCA, the loop seal may get flooded due to the accumulation of a significant amount of water. Unless the flooded loop seal is cleared of the water, the entire steam continuum running from the intermediate leg to the reactor pressure vessel to the steam generator will get pressurized.

Due to the so-called double manometer effect, the deeper the loop seal geometry, the higher the steam pressure needed in the core to clear the flooded loop seal by overcoming its static pressure head. With the bottom elevation of the APR1400 loop seal being closed to the neck point of its core height, the steam pressure on top of the core needs to rise up to the point of overcoming the static head of the deep loop seal before it is cleared.

The resulting pressurization could leave to a depressed water level in the core and, hence, to a temporary core uncovery and peak cladding temperature offense.

MEMBER MARCH-LEUBA: Okay. So if I understand correctly, if you have a seal, a loop seal like the one you're showing there, you will have lower water level in the core than in the downcomer because of this high pressure; correct?

MR. HAIDER: Yes.

MEMBER MARCH-LEUBA: Yes. If once you clear the seal, then the steam can go through there and you will still have a little higher pressure in the upper plenum because you have some pressure drops with the steam related, but it won't be as much; correct?

MR. HAIDER: Yes.

MEMBER MARCH-LEUBA: So the numbers we were looking at there from the DCD showed 6 meters of delta p, just water, which later we found out that is not as much because it was seal reference, but there's still a significant result.

Are you going to show us some continuous on this?

MR. HAIDER: Yes.

MEMBER MARCH-LEUBA: Maybe I'll wait.

MEMBER CORRADINI: This is Corradini.

I think, Jose, you're looking at a large break calculation when you quote 6 meters.

MEMBER MARCH-LEUBA: You may be right.

MEMBER CORRADINI: So, I had a question. I had a question I want to make sure just about quantitative analysis. You've got a bunch of axial distances labeled here. What is the bottom of the loop seal V sub -- I can't read it exactly -- V sub LF compared to Z core, what's the difference in height?

MR. HAIDER: The difference in height I believe is about -- it's about 18 percent if I remember correctly.

MEMBER CORRADINI: 18 what?

MR. HAIDER: 18 percent of the core height. The top of the core is 20.67 foot. While, while the bottom of the loop seal is 15.909 foot. And the diameter is about 2 foot.

MEMBER CORRADINI: Okay. So just repeating it back just so I got it right. The delta, the delta difference between the core Z and V sub 3 is what, how many feet?

MR. HAIDER: I was referring to ZLS because Z sub 3 is, Z sub 3 is a transient. I think you must be referring to either ZLS or the bottom of the crossover piping.

MEMBER CORRADINI: Either one. I'm just trying to --

MR. HAIDER: Okay.

MEMBER CORRADINI: -- I'm just trying to get the difference in height.

MR. HAIDER: Okay. So difference in height, if you look at the top of the core and the top of the crossover piping, the difference is about 2.27 foot.

MEMBER CORRADINI: Okay.

MR. HAIDER: Which is about 18.2 percent of the core height.

MEMBER CORRADINI: Okay. All.

MEMBER REMPE: Say that number again? What was that?

MR. CHO: This is Sung Ju Cho from KEPCO Nuclear Fuel. The difference between the tops of the loop seal height and the active core top is 2.3 feet.

MR. HAIDER: Yes. And you can see from the applicant's slide number 42 from this morning, they have a nice drawing of those relationships.

MEMBER CORRADINI: Okay. Thank you.

MR. HAIDER: Okay. Can we go to the next slide, please.

MEMBER CORRADINI: The reason I'm asking the question, then I'll be quiet, is that it's not the collapsed water valve but it's the 2-phase liquid level. And I'm eventually going to ask what's the calculated 2-phase liquids level, because with just a modest amount of void I still would cover the top of active fuel.

MEMBER MARCH-LEUBA: If I look at the applicant's slide number 42 I see that the maximum column height that they can have on the loop seal is 6.2 feet, which is 3.9 plus 2.3, of water. And I said 6 meter, really it's 6 feet.

So the maximum you could have on the

collapse level in the core would be 6 feet. And with a little boiling you will cover that.

MEMBER CORRADINI: So that, okay, you're thinking what I'm thinking. Thank you.

MEMBER MARCH-LEUBA: Right. So 6 foot of 12, I mean it will go, it can go as far as half, half the core. I mean if there is no pressure drops anywhere else, it still will be.

So the maximum you could possibly go was half, of the collapsed level will be half, half core.

MR. HAIDER: Yeah, that is correct. I mean, to be exact it's about 38 percent. To be exact.

MEMBER MARCH-LEUBA: Yeah. It would be nice to have a background calculation of how much boils you need to cover the core, to have 50 percent more.

Oh, you're back here. Do you guys have a Path 1, the bypass to the downcomer. On the top of the slide, Path 1.

MR. LU: Bypass through the core region or the bypass --

MEMBER MARCH-LEUBA: Yeah, bypass from the upper plenum to the downcomer. Are we generating so much steam that that bypass becomes irrelevant?

MR. LU: Okay.

MEMBER MARCH-LEUBA: I would like to see

for the RELAP or TRACE calculation the Path 1 bypass is it 1 percent of the steam flow or is it 99 percent of the steam flow?

MR. LU: David, do you want to take a shot at that?

MR. CARAHER: Yes. I'm David Caraher.

It's closer to 1 percent. It's very small compared to what's going around the loops.

MEMBER MARCH-LEUBA: But we've been told that during normal pressure with 100 percent water flow, liquid flow, you have .5 percent of the liquid going through there.

MR. CARAHER: No. You have .5 percent of the liquid flow goes up to downcomer to upper head.

MEMBER MARCH-LEUBA: Right. So now you --

MR. CARAHER: Point 5.

MEMBER MARCH-LEUBA: -- don't have -- you have, you have 2 percent power because you are shut down. So the mass flow rate of steam flow is 2 percent of what you used to have in liquid. And before you were able to put .5 percent of the liquid through there. And now it's steam, so it cannot be 1 percent.

MR. LU: But I think about giving the low pressure there, especially for this manometer type, the pressure difference between the upper plenum of the vessel and the downcomer, that part is very small. So with that small, even let's say, let's assume right there so you have a .5 percent of the opening or flow, the nominal operation with the pumps running through which has much higher DP, and then so the leakage flow for this case it becomes very small.

And then it's so small and then I think the current assumption is that let's assume it does not exist or that we need them to evaluate the loop seal, which may be more conservative.

MEMBER MARCH-LEUBA: My initial question was what does TRACE or RELAP predict?

MR. LU: Okay.

MEMBER MARCH-LEUBA: I would get the 100 percent power, 100 percent flow, adjust the KM till I get .5 percent, and then see what happens in this condition.

MR. CARAHER: We have done that.

MEMBER MARCH-LEUBA: You have done that? And you say it's 1 percent? Okay.

And you've done that decisively, I mean you're sure?

MR. LU: I'm sure.

MEMBER MARCH-LEUBA: You certainly agree with conviction.

MR. LU: Yes. Because that's on the analyses.

MEMBER MARCH-LEUBA: So you sound credible. Because if that part was open then the seal will never clear. But if it's only 1 percent it's okay.

MR. LU: That's right. But --

MR. CARAHER: This is David again. If it, if did allow all that steam to go there you wouldn't need to clear the loop seals.

MR. LU: There is no issue.

MR. CARAHER: There's no issues now. The loop seals don't need to clear because you're getting all the steam short circuited to, directly to the downcomer.

MEMBER MARCH-LEUBA: Yeah. And you don't have a difference in the --

MR. LU: And then you don't have depression off the -- to this level in the core either.

MEMBER MARCH-LEUBA: No, you will still have a delta P.

MR. LU: But the DP is very small. And if it is further reduced so you can have a --

MEMBER MARCH-LEUBA: The DP is the one that -- yeah, okay. If you don't clear the seal you'll never have sufficient pressure to depress the core. All right.

MR. LU: That's right.

MEMBER MARCH-LEUBA: It would be nice to have a bounding calculation saying, ignoring everything else, 6.2 feet of delta P between downcomer and upper plenum, you need 50 percent boils to cover the core.

MR. CARAHER: This is David Caraher again.

The TRACE calculation is around. You could actually look at those numbers. And I have informal RELAP5 calculations that could also show you those numbers.

MEMBER MARCH-LEUBA: Okay.

MR. LU: I think both we have, they're in both codes, yeah. So I think right after this, several slides later and we're going to show the slide with TRACE confirm 10 on the slide.

MR. HAIDER: So this is Syed Haider again.

Based on the docketed information in RAI responses the staff were able to identify several conservatisms used in loop seal modeling that are built into the S1M methodology and the computer codes used in the APR1400 small break LOCA evaluation model.

This slide captures a summary of those conservatisms.

First, the applicant described that the

loop seal clearing is delayed in the model until the water level reaches the bottom, and not just the top, of the horizontal segment of the crossover piping of the loop seal. The staff considered such a biasing of the loop seal downward to be conservative with respect to the loop seal clearing.

Modeling the loop seals 2.5 feet deeper than they typically are would delay their clearing and would allow for longer core uncovery period and, thus, a higher PCP.

It's also worth mentioning that the S1M SBLOCA methodology is based on the Appendix K and uses conservatisms like 1.2 multiplier for decay heat curve and a partly skewed axial power shape that would promote core uncovery by biasing the axial PCP to peak near the top of the core. So, the hardest part is keeping it moderate, somewhere 15 percent to below the top of the core conservatively.

Another feature of the S1M methodology is that if you lump two of the four seals for intact loop cold legs into a single equivalent loop seal. For licensing basis simulation results they have provided, the staff established that the lumping two loop seals into a single loop seal was conservative with respect to loop seal cleaning for delimiting case of small break

LOCA.

From detailed spectrum analysis it was evident that the lumped loop seal was the last one of the three loop seals to clear for delimiting the small break LOCA. Even for the two break sizes analyzed around delimiting SBLOCA case, the lumped loop seal was not the first one to clear.

This supports the staff's conclusion about the conservatism of the lumped loop seal modeling.

The staff also found that CEFLASH-4AS licensing basis calculations were more conservative than the applicant's supporting RELAP5 and the staff's TRACE confirmatory calculations. So the applicant also submitted their RELAP5 calculations. But they are no the licensing basis calculations, they are just working calculations.

MEMBER CORRADINI: This is Corradini.

I don't understand what that bullet means compared to the second bullet or Appendix K. So you're telling me then on top of using Appendix K assumptions there were additional conservatisms in the CEFLASH calculations?

MR. HAIDER: That is right.

MEMBER CORRADINI: And you know what they are or do you just know they're somewhere buried in the calculations?

MR. HAIDER: I'm identifying them on this slide.

MEMBER CORRADINI: Okay.

MR. HAIDER: Like the loop seal clearing delaying it until --

MEMBER CORRADINI: Yeah. Yeah, I understand the loop seal clearing. I understand Appendix K methodology.

I'm asking what are the additional things referenced in the fourth bullet?

MR. HAIDER: Okay. The fourth bullet is emphasizing that a CEFLASH-4AS licensing basis calculation were found to be more conservative than the applicant's RELAP5 and the staff's TRACE confirmatory calculations.

So, yeah, the fourth bullet doesn't have an additional conservatism but it builds confidence that CEFLASH-4AS licensing basis calculations are more conservative than RELAP5 and TRACE confirmatory calculations.

MEMBER CORRADINI: Okay. But just asking my question again, the root cause for the difference is unknown? In other words, you're telling me the number if bigger but you don't know why the number if bigger, you just know it's bigger?

MR. CARAHER: It's bigger due to the bullets ahead of it.

MEMBER CORRADINI: Well, I don't think that, I don't think that's what Syed was saying to me. The bullets ahead of it, you could do a one-to-one comparison between TRACE and CEFLASH with the Appendix K methodology and lumping. And what I thought the fourth bullet was saying, even if you did that it has additional conservatisms embedded in it.

Am I misunderstanding the bullet?

MR. HAIDER: What I am really trying to say here is that because of the three earlier bullets, CEFLASH licensing basis calculations are more conservative.

MEMBER CORRADINI: Okay, fine.

MR. HAIDER: And it has been demonstrated by the applicant and the staff.

MEMBER CORRADINI: So it's the -- okay, but it's the things above that you mean? You don't mean additional things in addition to those above?

MR. HAIDER: No, no, no.

MEMBER CORRADINI: Okay. All right, thank you.

So another aspect is that the staff also found that the applicant's CEFLASH-4AS -- yeah, I mean I've gone over that, so.

MEMBER REMPE: Well, let's not just skip by that. This is a good place for you to answer my question from this morning. Tell me how you validated CEFLASH using data from Semi scale and why it's validated for a CE plant?

MR. HAIDER: Yeah, I'm going to do that.

MEMBER REMPE: Oh. Oh, okay. I'm sorry.

MR. HAIDER: No, no, I'm not skipping that.

MEMBER REMPE: I thought you were going to skip it.

MR. HAIDER: No, no, I'm not skipping that.

Okay. The applicant also documented that CEFLASH-4AS computer code used in the S1M methodology had been relegated for the loop seal clearing phenomena in Semiscale test SUT-8.

The SUT-8 test was designed to assess CEFLASH-4AS computer code for the CE plant features that are also relevant to the APR1400 design, such as core uncovery, water hold-up, and the loop seal formation. The test was designed to induce an extended core water level depression prior to loop seal clearing.

MEMBER REMPE: So was the geometry for a CE System 80 type of configuration or an APR1400 configuration? Or was it a generic PWR that was probably looking more like a Westinghouse plant?

MR. HAIDER: It's a generic PWR --

MEMBER REMPE: Okay. So you're aware of the issue that was raised about last time you talked about the PRA, that there's a difference with the CE geometries and the Westinghouse geometries. And so what I was trying to get to earlier this morning was do we have confidence without going through a process such as was done with NUREG-2121 -- am I giving the right report number? No, I should know it now. But do we have confidence that this code is appropriately or is it appropriate for CE geometry?

Am I saying that clearly enough? You know where I'm coming from? I think someone --

MR. R. LEE: This is Robert Lee from Westinghouse.

And I'd like to try to answer your question on behalf of -- well, right now it's staff but really it's KHNP. Earlier this morning, I think it's this morning, right, you raised the question which you just asked of us that the scale geometry is really modeled on Westinghouse design compared to a CE design, so what's the difference?

Well, if there is a difference between CE design and Westinghouse design how can this system test the reactor for CE design application? I think that is really key of your question; right?

MEMBER REMPE: Yes.

MR. R. LEE: Okay. That, I think most of us, for me it was Semiscale testing, so I know what to describe what the Semiscale test is. One thing, Semiscale test, a lot of criticism in that scale. One thing in my view Semiscale, the best thing was the way it maintained the elevation of the system. And so that in this design, the Semiscale design especially for this clearing behavior, because this is an manometric effect between this loop seal side and core side, and that geometry is basically the same between Westinghouse and this redesign, again APR1400 design.

The only difference, only difference is that the difference between the top of the core and the loop seal bottom is much shallower for APR1400 design. So in terms of the model, that's going to affect the pressure depressions and the core region before the core -- it's clear that physics is the same, so.

MEMBER REMPE: So it's an elevation, it won't be the actual volume of water, --

MR. R. LEE: Exactly. Yes.

MEMBER REMPE: -- the water height.

MR. R. LEE: Right.

MEMBER REMPE: It's the case parameters is what you're telling me.

MR. R. LEE: Yes.

MEMBER REMPE: Okay, thank you.

MR. R. LEE: Thanks.

MR. HAIDER: This is Syed Haider again. So

based on the overall review of the staff, the staff concluded that the S1M methodology and computer codes conservatively characterized the safety significant phenomena of the loop seal formation and clearing and peak cladding temperature during the most limiting small break LOCA. Therefore, there is sufficient overall conservatism in the S1M as the LOCA methodology as it is applied to the APR1400 design.

MEMBER MARCH-LEUBA: Have we run a sensitivity analysis to let's assume we're all wrong and the seals never clear, what happens? I mean are we relying on the seal clearing to survive the event? Or is the seals all blocked, all four are always blocked, does this survive the event?

Because in my mind there is sufficient physics support and volubility that there is some probability -- I don't know how high -- that none of the seals clear.

MR. CARAHER: Not possible. David Caraher. Not possible.

They have to clear. They have to clear. One has to clear. Because otherwise you just build up steam forever in the upper, upper part.

MEMBER MARCH-LEUBA: You don't have enough relief from the pressurizer? Are you --

MR. CARAHER: No. No. It's not possible to have a small break LOCA without clearing a loop seal, at least one. But often you'll clear all four and end the slide.

MR. LU: Another analogy is you put a ball on top of the hill and you can always assume the ball may stay at the top of the hill, but in reality something's going to happen and it's going to tilt and so either it's not a stable condition. You have a continuous loop seal build-up. At the same time you have the depression of the core, of the lining of the core. That's part of the manometer, the banners if it's already unstable situation. So either way it will be, it will be done.

And then based on actually that's one of the reasons it's possible to catch that phenomena. That's the reason we asked the KHNP initially. So they were doing the best estimating analogies, running RELAP5.

We were able to predict the core, the depressed level into the core region, it's almost half of the core, and then it comes back. And the loop seal clears. And then so does the TRACE.

So we have not seen the situation, at least right now we don't have evidence to show that the loop seal would stay forever at this point.

MEMBER MARCH-LEUBA: The loop seal you have the left side of the seal when you when you try and push it down you have 6 feet of water.

MR. LU: Yes. Right.

MEMBER MARCH-LEUBA: And there's steam behind it.

MR. LU: Right.

MEMBER MARCH-LEUBA: And this steam has to push 6 feet of water out. It pushes like the minute it starts boiling. But and then releases pressure but never, never removes the column of liquid.

MEMBER CORRADINI: This is Corradini.

I'm not, I'm not sure that's how it works in a plumbing sense. It doesn't have to clear all of it, it has to bubble through it.

MR. LU: Right.

MEMBER MARCH-LEUBA: That releases a little pressure but only at the critical pressure, which is the 6 feet of water.

MR. LU: Yes. But that's a manometer type of pressure boundaries. But continuously you could have steam generated through the core, getting the steam generator aside. And inside the steam generator it will be the tube gun going downwards. There is a continuously condensation going on if you have steam going there. So you are going to have additional water supply to get into the loop seal.

And that's what --

MEMBER MARCH-LEUBA: You keep presenting it.

MR. LU: Yeah, exactly. And then at certain point that all depends on the pressure upon it. So far we have seen quite many like PKL tests, in additional to Semiscale, and then PKL test is the latest one, they all observe loop seal clearing if you have deep loop seal. And then none of them would show it would stay there. And only periodically for a limited amount of time you capture that one.

MEMBER MARCH-LEUBA: My point is --

MR. LU: Yes.

MEMBER MARCH-LEUBA: -- it would be very nice if we could say that the loop seal remains intact, my core is fine. I'm not relying on --

MR. LU: I agree with you. I agree with you. We then can argue for -- KHNP can demonstrate that one. That will be very safe for us to say everything's fine.

MR. CARAHER: This is David Caraher.

You could, the only way you could prevent the loop seals from clearing and still have this reactor survive would be feasibly to bleed off, go with the secondary and keep the connections clear.

But that would, then you wouldn't need loop seals to clear in that case, of course.

MEMBER CORRADINI: Yeah, you would equalize pressure by --

MR. CARAHER: Yes. That's right.

MEMBER CORRADINI: -- by condensing.

MR. LU: Yeah. Secondary side or, you know, keep it going.

CHAIRMAN BALLINGER: Can we kind of move on.

MR. LU: Let's keep going.

CHAIRMAN BALLINGER: Not just kind of move on. Let's move on.

MR. HAIDER: So we already submitted DCD and technical report presented a small break LOCA spectrum analysis results for two types of breaks that included direct vessel injection line or DVI line breaks and the pump discharge or PD cold leg breaks.

The applicant presented four DVI line breaks and four cold leg breaks. The staff determined that the submitted small break LOCA spectrum had measure gaps in the analyzed break sizes that would not satisfy the SRP guidance that interior diameter break sizes may not be sufficient to identify delimiting SBLOCA break size with the highest peak cladding temperature.

Therefore, the staff requested KHNP to perform a final small break LOCA break spectrum analysis with no major gaps. And the applicant submitted a revised spectrum analysis of 15 DVI line breaks and 17 cold leg breaks in half-inch break size increments.

As asked, the applicant provided details for each of the targeted breaks analyzed, including peak cladding temperature, loops seals clearing order, and the core cool phase mix 11, white fraction, mass flow rate, safety injection flow rate, core pressure, and break flow rate.

The staff accepts the licensing basis calculations and found that a 5-inch diameter DVI line break is identified as the limiting small break LOCA or the highest peak cladding temperature of 1,683 degree Fahrenheit. This still has about a 517 Fahrenheit margin to the 2,200 degree Fahrenheit regulatory limit.

The staff's TRACE confirmatory calculations show the maximum peak cladding temperature of 1,265 degree Fahrenheit that has 935 degree margin to the limit of 2,200. Which shows that the licensing basis calculations are significantly more conservative than the staff's confirmatory calculations.

MEMBER CORRADINI: Syed, this is Corradini. So let me make sure I understand.

This margin or this difference is due to the Appendix K methodology and the location of the assumed loop seal clearing? Or what is -- are those the two major reasons?

MR. HAIDER: And also the fact that they depressed the loop seal clearance by 2.54.

MEMBER CORRADINI: Okay. But what I guess I should have asked just precisely, when you do the TRACE calculations you're not making the same assumptions for Appendix K as the applicant?

MR. HAIDER: Yes.

MEMBER CORRADINI: Okay, thank you.

MR. HAIDER: So this is -- This is Syed Haider again. How this issue is still being read as a confirmatory item as the applicant needs to update the DCD and the technical report to reflect the revised break spectrum analysis as shown by the mark-ups that have been submitted with the RAI response.

Now, so based on the overall review activity, the staff concludes that the applicant was able to demonstrate sufficient conservatism in the APR1400 small break LOCA analysis using the Combustion Engineering Supplement 1 methodology to meet the regulatory requirements for light water reactors, as given in 10 C.F.R. 50 46 and Appendix K to 10 C.F.R. Part 50.

The SER still has one confirmatory item and one open item. However, they are mainly documentation issues with no outstanding safety concerns about the APR1400 short-term response with small break LOCA.

For the confirmatory item, as I described, even though the detail SBLOCA analyses are acceptable to the staff, the region has still yet to be reflected in the DCD.

Similarly for the open item, the staff is still expected a docketed RAI response from the applicant to justify the upper bound on the small break LOCA break sizes analyzed. We will do it as an open item. This is not a safety concern but a completeness issue as delimiting PCP was proven to take place for the 127 centimeter square DVI line break.

This concludes my presentation for the short-term response of SBLOCA. Now I would like to ask if the committee still has any questions left?

MR. LU: So we will go into the TRACE large and small break LOCA confirmatory analysis.

And as we mentioned at the beginning, we asked for Office of Research to develop the TRACE input deck based on the APR1400 RELAP5 deck. And they have done great work to support our regulatory review.

So we will have Dr. Staudenmeier from Office of Research to talk about and give a presentation. The actual work was done by Bill Krotiuk. He was NRC Research staff.

MR. STAUDENMEIER: As Shanlai said, I'm Joe Staudenmeier. Bill Krotiuk did the calculation. I was involved in some small break LOCA calculations last summer if we had some issues that came up before the final reports Bill did before he left. He's out on a cruise now, enjoying -- having a more enjoyable day.

So the presentation provides a description of the TRACE model. I'm not going to go into too much. I need some more details about the modeling to some of the backup slides that are at the end of the presentation.

We looked at large break LOCA analysis and small break LOCA analysis to compare the DCD results and also some results for thermal conductivity degradation.

Next slide.

This is just kind of a noding diagram view from SNAP showing that we modeled the whole system in TRACE. It's a fairly big model as far as TRACE models go for LOCA. Highly detailed.

Next slide.

The way we run TRACE, we run steady state calculations to reach full plant operating conditions. We initial the system at 102 percent power to cover the uncertainty in power range instrumentation. WE have steady state system conditions that compare well to the conditions in the DCD. And also have set points and delays from the DCD.

Next slide.

Okay. For the limiting break it was a double-ended guillotine break in the primary system cold leg. Failure to safety injection system pumps, consistent with the DCD, and all safety injection tank flows.

Can see the summary, the results. We get answers that are pretty close to what they are calculating in the DCD, and in terms of both temperatures and locations of the limiting temperature or limiting location.

CHAIRMAN BALLINGER: So, to the uninitiated you're assuming that the APR1400 DCD, their calculation did not include burn-up dependent thermal conductivity? That's in TRACE?

MR. STAUDENMEIER: Not in this calculation. In later calculations we have some.

MR. LU: We did a spectrum of them.

CHAIRMAN BALLINGER: Okay.

MR. LU: So initially it was comparing head to head to see what's the difference.

MR. STAUDENMEIER: Yeah. There's a slide that comes later that shows how much difference there is.

Okay, next slide.

This is just kind of traces of curves of the temperature versus time compared to the licensee calculation. There's two different probes there for TRACE, the TRHMAX, that's maximum temperature looking around the whole core. So that position of that temperature can change during time.

The other curve is the temperature at the same position as the licensing -- licensee limiting curve. So it moves in a little earlier because in the TRHMAX that's not quenching until the whole core quenches, so the top of the core will quench later in lower elevations where the peak temperature is.

MEMBER CORRADINI: Can you repeat that? This is Corradini. I don't understand the difference between the red and the black. Why is it quenching later?

MR. STAUDENMEIER: TRHMAX is -- looks core-wide for the limiting temperature. So that won't quench until everything in the core is quenched.

The other curve is at a fixed location in the core where the peak temperature occurs, which is lower down in the core, so that will quench earlier than the rest of the core quenches.

MEMBER CORRADINI: Oh, I understand that. But I'm looking at the difference between the black line which is the DCD --

MR. STAUDENMEIER: Oh, the DCD? Okay. The DCD is a licensee -- okay, they're quenching, if you look at their quench model, they're quenching from a higher temperature than TRACE is. So they're entering in the transition blowing at a higher temperature than TRACE does.

Our t-min model is based on stainless steel data. RELAP5 doesn't really have a t-min model. I don't, actually I don't if the KHNP model does. But our

RELAP5 doesn't have a t-min model. What it looks at, it has a transition boiling correlation and film boiling correlation, and it takes the highest t-transfer coefficient of the two, so.

MEMBER CORRADINI: Okay. That explains it. Thank you, Joe.

MR. STAUDENMEIER: So it's, yeah, it's just entering in the transition boiling. That's the way we do our TRACE.

Okay, next slide.

Small break LOCA, we also performed some small break LOCA calculations for a spectrum of break sizes with the same safety system features and failures, consistent with the DCD. There was a break spectrum. We can see the different SI pump operation and SIT availability, a variety of calculations.

Next slide.

Yeah, actually unfortunately I don't have a PCT versus time curve for this.

MEMBER STETKAR: This actually isn't APR1400, is it? It's the old CE plant, used to have two pumps on one diesel.

MR. STAUDENMEIER: It's the assumptions, same assumptions as in the DCD, in the DCD calculations, so.

MEMBER STETKAR: It's the old Combustion Engineering plant that had two SI pumps on one diesel. You can't fail two SI pumps on this plant with failure of a single diesel.

MR. STAUDENMEIER: Well, what they assume is the SI in the broken line doesn't inject it, it all spills at the break.

MEMBER STETKAR: Okay.

MR. STAUDENMEIER: So that's the difference between cold leg breaks and --

MEMBER STETKAR: Got it.

MR. STAUDENMEIER: -- direct vessel injection line breaks.

MEMBER STETKAR: Got it.

MR. STAUDENMEIER: Okay?

MEMBER STETKAR: Okay.

MR. STAUDENMEIER: All right. This is a result of the break spectrum. One thing here is you should ignore the numbers with the kinetics on because that was using a faulty feedback curve that was discussed during the large beak LOCA staff presentation before. So, so the temperatures, yeah, there isn't the power feedback that would have gone on in these calculations, isn't real. It was from, this report was done quite a while ago and it would have to be re-done with the more recent feedback.

But the actual, the non-feedback curves are a good way to really model small break LOCAs, which is constant power until you get a reactor trip and you go on the KP curve and that's --

MEMBER STETKAR: I'm still confused. And I hate, hate to belabor this because I'm not a thermohydraulics person, I just know how plants work.

If you look at the DVI break on this slide that you have here, not on the slide that you have up here, the number that I'm on, 65.

MR. STAUDENMEIER: Okay.

MEMBER STETKAR: It says one SI pump operates. Two SI pumps fail due to emergency diesel failure, and one SI flows out the break.

I challenge anyone to figure out how on this plant two SI pumps fail because --

MR. CARAHER: This is Caraher. One is out for maintenance, the other fails due to the diesel generator failure.

MEMBER STETKAR: Ah, okay. Thank you.

MR. STAUDENMEIER: Next slide.

MEMBER SCHULTZ: Joe, on 66 can you tell us what we're not supposed to look at here? I didn't find it --

MR. STAUDENMEIER: Okay. There's a, in the, yeah, next-to-last column there's two temperatures. And one's for no feedback and one's with feedback.

MEMBER SCHULTZ: Got you.

MR. STAUDENMEIER: The ones with feedback you should disregard. So the ones before the slash are ones with no feedback. So they're more representative of what the safety analysis would see.

MEMBER SCHULTZ: Thank you.

MEMBER CORRADINI: So I'm -- whoever asked that question, I'm just as confused. So I ignore the N/As?

MR. STAUDENMEIER: No. Not the --

MEMBER CORRADINI: We're on slide 66 I thought.

MR. STAUDENMEIER: We are. Now, the PCT column, the next-to-last column, TRACE analyses, no feedback/feedback, there's two temperatures under PCT and also two ECRs. No feedback and feedback. So the no feedback ones are the ones you should pay attention to.

The reactivity feedback curve using a feedback calculation was an obsolete curve that is not good.

MEMBER CORRADINI: Okay. All right, sorry. I misunderstood. Sorry.

MR. STAUDENMEIER: Okay. And, yeah, talking about loop seal clearing, loop seal clearing is really not a big deal for small break LOCAs. There's a short transient heat-up you get that isn't the limiting temperature generally. And it's been studied in a lot of test facilities for a long time. There's probably a good description of it in the ECCS compendium if you want to find out more about it. Yeah, lots of test facilities who've looked at it.

And, actually, the bigger the piping, the less the loop seal will fully clear. Like the biggest loop seal test I think were UPTF. They were full-size piping. And you don't clear it fully, you get bubbling up through the water column going up to the cold leg pump in the cold leg.

Something like Semiscale, which is a real thin facility, you actually do sweep out the whole loop seal because of smaller pipe size. You get in a different -- like, a big pipe you can't get slow flow, whereas in a Semiscale you could get slow flow.

MEMBER REMPE: So now you're confusing me. It's not just a water-height effect, it's the geometry of the piping. Is that what you're telling me?

MR. STAUDENMEIER: Some, to some extent it's size of the piping in two-phase flow.

MEMBER REMPE: So now they --

MR. STAUDENMEIER: That can get through it, yeah.

MEMBER REMPE: -- validated their code for Semiscale. Is the sizing of the piping appropriate to --

MR. STAUDENMEIER: Well, it would be actually worse in Semiscale than a full-size plant.

MEMBER REMPE: Well, is it appropriate to --

MR. STAUDENMEIER: But you don't get bubbling through as early as you get in a full-size plant type.

MEMBER REMPE: Then tell me again why it's appropriate to use a code that has been tuned to predict clearing in Semiscale for the APR1400? Why is that a good code to use?

MR. STAUDENMEIER: Yeah, I don't know if I'd call it tuned. I'd call it compared to the results. Semiscale loop seal is deeper than a CE plant loop seal because Westinghouse loop seals are deeper. So, I mean you're just -- and it's really predicting loop seal clearing is a balance of steam flow around the loop.

There's also some water hold-up you can get in steam generator tubes that can also cause an addition with DP. Compared to steam flow in the bypass, like hot leg nozzle or upper T between the upper head and the downcomer, so.

MEMBER REMPE: So you think --

MEMBER CORRADINI: I think, Joy, I think all he's basically saying is that if the length scale of the diameter is on the order of the length scale of the U-tube, then you get bubbling, you don't get a push-through. If the diameter is very small you push it through like a slug of water. That's what he's saying.

MEMBER REMPE: So, then tell me is the L over D appropriate for applying this to the APR1400? Because you've tuned this for matching the Semiscale, did somebody do some sort of analysis and say, yeah, close enough, I can go ahead and apply it to the APR1400?

MR. STAUDENMEIER: Well, I guess back in the '80s there was a decision made that people -- because the only tests back then that existed were Semiscale and LOFT. And aftermath of TMI the vendors were told they had to compare their small break LOCA codes to Semiscale and LOFT. And there were SERs written up on them saying if they compared conservatively to those tests and said they were okay for Appendix K small break LOCA.

So actually, and it's not a big deal. The loop seal clearing isn't going to make a big difference in peak clad temperature anyway. That's not where the peak clad temperature comes from, it's from the longer core uncovery that occurs later, so.

MEMBER REMPE: And you have a lot of margin. There's another thing you could have said that would have made me happier. But, again, I just was, I was looking at this and I said, well, it's validated for using this. I was kind of, in a way it made me wonder. And I looked at the RAI response.

There's other reasons I would buy off and say this is appropriate, but I'm not sure I would say that it's validated and I can get good numbers or very precise numbers with it.

MR. STAUDENMEIER: Yeah. I mean I haven't seen the CE report where they -- back from the '80s where they did the comparison, so I don't, I don't know what that looks like. But I know what TRACE looks like for those things and was able to predict loop seal clearing pretty well over a whole range of tests.

MEMBER REMPE: The staff write-up, your draft SE, they acknowledged the fact people were trying to validate a CFD model because they were saying more advanced codes are being used now. And that was the motivation for asking that question to KHNP. And then they said, and they responded back, and they said, yeah, it's validated or it's appropriate.

And I just am inquiring why the staff decided it was appropriate.

MR. LU: May I just chime in?

So when we talk about Semiscale it's a very skinny, it's a very skinny pipe. As Mike on the phone mentioned, when you have a very skinny pipe and it's easy to accumulate. And it's very hard from a collective point, very hard to clear.

So in comparison, when you, you know, tune the code or validate the code at some point, when you apply it to the APR1400 so that means that you do need some conservatism in terms of timing of the clearing of the slot.

MEMBER REMPE: Okay.

MR. LU: But in reality because it's not a one deep load and they have large diameter, you have 2 feet diameter of the cold leg of the, you know, loop seal, therefore the penetration of this steam going on starts earlier than what your model predicts. So the model, when it's validated at the Semiscale, it tends to be more conservative.

MEMBER REMPE: That I would buy, too. But

I guess I didn't get that from what he said. It may be it's in the words.

MR. LU: Yeah, yeah. I think that I'm trying to explain that point.

MEMBER REMPE: Okay, thank you.

MR. STAUDENMEIER: The big conservatism in Appendix K calculations is the --

MR. LU: Right.

MR. STAUDENMEIER: -- 1.2 multiplier under decay heat. That's where most of it is, and maybe some in the break flow. But the big one is with decay heat.

MEMBER MARCH-LEUBA: Okay. Joe, can I say in plain English what you said? For a small break LOCA whether the seal clears or not makes no difference?

MR. STAUDENMEIER: It makes a difference if it clears. I mean the timing of clearing and things like that don't make much of a difference. It throws some kind of randomness in the calculations. That's why the vendors all try to make deterministic clearing. They bias their calculations in a certain way to get consistent clearing. So that especially if they make a change to a code that they're seeing the difference of that change instead of seeing a difference in how loop seals clear. Because even in testing there's some randomness in testing on how we see this, too, so.

Especially if we have multi-loop, they're balancing and oscillating, and the one that catch it in phase, the right phase, that one will clear first, so.

Okay. Yeah, I was going to say we won't see this same temperature versus at the same break size or limiting temperature. The higher the decay heat, the limiting break size moves to a bigger break size because of the, just because of the physics, a small break LOCA isn't relieving energy through the break. So we'll see in our realistic calculation with lower decay heat, we'll see peak temperatures and a smaller break size in Appendix K calculation there.

Let's go on to the next slide.

Yeah, I was going to say one last thing about loop seals is, like, the big loop seals are B&W plants. And you would never be able to clear them. The bottom of the loop seal is below the bottom of the core. But they have vent valves in B&W plants between the downcomer and the upper plenum so that you don't have the loop seals, you just vent through the vent valves.

Okay, long-term cooling. There was some long-term cooling for both large and small break LOCA to determine if a second PCT occurs due to loop seal refilling and clearing. And to make a long story short, there were calculations were run out for a long time and we didn't see any significant heat-ups.

The longer you go out in time, the less steam is generated and the less loop seal depression you can generate. And eventually at some point you would be able to get where the leakage between internal core and the downcomer would be enough that it could relieve all the steam at some point. But, yeah, we don't see any loop seal depression or significant heat-up. And we bring calculations out long past where the switchover to hot leg injection is, so.

MEMBER STETKAR: That's, recirculation here is hot leg recirculation time?

MR. STAUDENMEIER: Yeah. That recirculation time is where you would have to switch over to actually sump --

MEMBER STETKAR: No, not on APR1400. There is no recirculation. It's the same pot of water.

MR. STAUDENMEIER: Okay.

MEMBER STETKAR: So I'm trying to understand what the significance of sump recirculation and these very precise times are.

MR. LU: I think what Joe means, I think, is they're related to the switchover time.

MEMBER STETKAR: Switchover to what?

MR. LU: From cold leg injection to hot leg injection.

MEMBER STETKAR: Okay. Hot leg I get.

MR. LU: Yeah. Yeah.

MEMBER STETKAR: If that's what we're talking about here.

MR. LU: Yeah, if you have a hot leg injection you don't see the loop seals stop in terms of long-term. And you can see the conclusion around to 5.44 hours or 4.79 hours, assuming there was no switchover. But in reality when they have this one, that's the mandated switchover. And then we to uncover as part of why they, you know, require that 2-hour switchover time.

MEMBER STETKAR: Right. Right. Okay.

Okay, thanks.

MR. STAUDENMEIER: Okay, next slide.

Thermal conductivity degradation. Some calculations were done looking at different times in burn-up. You can see the first column is the primary fuel rod. That's at a nominal peaking factor.

The next three columns are at different burn-ups, and they're a hot rod with higher peaking factor. You can see the difference in peak center line temperature.

As you go to, the column on the furthest right is the lowest burn-up. Supplemental rod one is the highest burn-up, that's 60 gigawatt days per metric ton. You can see the central line temperatures are significantly higher. And down at the bottom, the PCT is significantly higher also.

I guess on thing is kind of unrealistic about this is rods that are at that burn-up, you could never get that peaking factor on them, so. The rods with the highest burn-ups would have lower peaking factors. But this is just kind of bounding everything, showing that if you did have a peaking factor at different burn-ups, the difference that you get. And it's just all because of the stored energy and you can't take as much out during a blowdown cooling. And some of it gets locked into the heat flow, so.

But, yeah, I don't want to -- I know the review isn't finished. But this part has quite a bit of margin in terms of large break and small break LOCA margins. So this isn't something I think that is going to make a significant safety difference in the long run. It's just a matter of doing calculations.

In short we've had the thermal conductivity degradation models in the code for I think about eight years now. They're something we -- I remember looking at this for AP1000 in operating plants at the time the issue was first brought up, quite a while ago.

Okay. Next slide.

Okay, conclusions. Yeah, we have predictions that are similar to or bounded by the AP600 -- or APR1400 DCD. Significant margins to 50.46 limits.

The small break LOCA, I guess the one thing there is about the reactivity feedback. It is significant, but that was with a bad reactivity feedback curve. And long-term cooling calculations show no way heat-ups are things you have to worry about in terms of long-term core uncovery.

Okay, I think that's the last slide.

MR. LU: Go to the next one.

MR. STAUDENMEIER: Okay.

MR. LU: Boron presentation, I think Dan. Yeah, the next one will be Dan Prelewicz.

MR. PRELEWICZ: Thank you. Appreciate it.

I am Dan Prelewicz. I'm going to talk about the boron precipitation. If I find my right slides here.

We're on slide 70. And first of all there's a technical report that's referenced here that covers the boron precipitation analysis.

The methodology is a modification of the

CENPD-254 methodology which dates to the '70s. But in 2005 that methodology was basically unaccepted by the NRC and was called a Waterford Interim Methodology. Replaced it. There were some deficiencies in that methodology. For example, they didn't consider voids. They have a mixing zone which is where the boron concentrates. Half the LOCA plenum was established by some testing and in the core region, and then it was above the core. And then they didn't consider any voids there. So there was too much liquid. They weren't using the 1.2 multiplier in separation.

So that was fixed. And the new methodology is what is being used at this point, the so-called Interim Waterford Methodology. With one exception. Since that time the mixing zone was basically changed by Westinghouse. And I notice Duke is doing the same thing. They don't include the region between the bottom and the top of the hot leg, it's no longer in the mixing zone.

If you think about it, they, once you get liquid up there it's starting to flow and the lines are basically starting to flush. And the methodology also used a pressure drop based on steam flow in going through out the hot leg and through the steam generator. So you would get a higher pressure drop if you had liquid going there and, plus, you're flushing.

So now they're consistent with what's being used for the Westinghouse plants. And I notice Duke uses the same assumption as the lower, the bottom of the hot leg for their analysis of boron precipitation.

So the analysis basically determines when you do the switchover to combined hot leg -- DVI really in hot leg injection above the core injection. And the switchover to hot leg injection by the operator basically starts the flushing process where you flush the core of the concentrated boron in the core.

The methodology uses, going back all the way to the CENPD-254 methodology, four computer codes, BORON, NATFLOW, CEPAC, and CELDA to do various parts of the calculation. The codes that were approved then were basically the same codes used, except that the BORON code is modified very slightly because, I think as somebody mentioned during the previous conversation, you don't have to switch the source. The source is before you had to switch from the refueling water tank to the sump. And now there's an in-containment refueling water storage tank that basically is the sump. So you can do the switchover. So that's a very minor change that was done to the methodology, to the computer code. And we invariably checked that out to make sure that was done right.

There were raised a lot of issues in RAIs. But basically the only one that really -- a couple of them we had some significance. One of them was that switch of the mixing zone from the top of the hot leg to the bottom of the hot leg and the change in the computer code to check the computer code, the BORON code for the switchover.

So then KHNP decreased the size of the mixing zone. And what the consequence was that it changed the switchover time from three hours to two hours. And they modified the DCD to reflect that.

So once those changes were made, the boron precipitation long-term cooling methodology and the plan are acceptable for the APR1400.

Are there any questions regarding boron precipitation?

MR. LU: Just one point about that.

Because of the staff RAIs from the vendors the design changes are made because of the result of our review on RAIs. So the switchover time has been shorted from three hours to two hours.

MR. PRELEWICZ: Okay, next slide.

MR. CARAHER: Yes, this is David Caraher. I'm going to address the review of the long-term cooling boron dilution.

In the current version of the SER there was an RAI and following the SBLOCA which addresses boron dilution due to loop seal clearing or the restart of the pump. And the applicant has responded, providing calculations -- well, in the current version of the SER it's an open item, but it's really been closed since then. The responses were not docketed yet; that's largely why it was open.

So now the boron dilution due to a start-up of an RCP or reestablishment of natural circulation and the applicant has done it simultaneously in all loops. But the PKL test shows this would be conservative because natural circulation basically gets started in one loop, and then another and then another. So that's one big conservatism.

And then we checked the mixing calculations that were used in the boron dilution calculations and they were verified to be conservative.

So the applicant asserted the closure of GSI-185 also applied to the APR1400 because its geometry is essentially the same as the system 80.

And so the minimum calculation, calculated boron concentrations done by the applicant were significantly above the criticality limit. And informally they also verified their calculations, or corroborated them with the CFP calculation which showed that they were significantly above the limit.

MEMBER SCHULTZ: You said informally they, they talked to you about that or they --

MR. CARAHER: They presented it to us in the, it's in the SER.

MEMBER SCHULTZ: Okay, thank you.

MR. LU: The old version of SER is still providing. But, you know, as it is right now we ought to receive their RAI responses. But I don't think it's going to be a problem to close it.

All right, next slide. You want to comment? Long-term loop seal clearing.

MR. CARAHER: Oh yeah. It's getting late.

The long-term loop seal clearing, the analytical methods is, well, there was a supplemental RAI response. We asked the question, well, if you did it does the loop seal reseal later in time in the small break LOCA, plug the system, and now the temperature gets another rise. You know, the slide presentation was only on short-term.

And so we asked that question. And they went off and they, they ran the calculations out 7,200 seconds. And they did a break spectrum and looked at several. And they found that, yes, they did get some, you know, spot loop seal clearing or loop seal refills, and then they would clear slightly and then the refills clear slightly. And so you'd see it going along in the long-term and you'd see little bumps in temperature. But the maximum of all those cases was only 627 degrees Fahrenheit.

So, the question that we asked was that, well, did it remain below 800? And, yes, their analysis showed that it did.

And I think you also heard Joe say that TRACE calculation showed that it never went back up in temperature.

MR. LU: So that loop seal issue by itself has been closed from staff perspective. Consider this acceptable. Okay.

The last piece on long-term cooling in-vessel downstream effects. And it covers three parts, or four parts of the debris source analysis and the available driving head across the debris bed, fuel assembly head loss testing, and LOCA deposition model.

And I did want to point out there was a presence from the regulatory practice perspective. Right now it's being used by the staff, approved the ACRS, too, and it was back too for the WCAP

16793 regarding a clean plant criteria. So the clean plant criteria, as long as you provide the licensee or applicant has demonstrated for domestic plant, not for international, that they have, they can demonstrate that they have a 15 grams per fuel assembly. They now need to do additional analysis, whatever, and then testing.

And but one had the year before the submittal of the DCD. And then we met with the KHNP and we told them, hey, although it might be, you know, 15 grams per assembly clean-plant criteria, however, that was based on the test data for domestic fuel. And then some fuel has not been tested, so it's better to have a test and then they launch their program to do this. And then they perform the full spectrum analysis starting from debris source, available driving head, and the fuel assembly head loss testing plus LOCA at the end. And it's similar to the WCAP methodology.

Next slide.

Okay, related to the debris source they did, they performed a fiber only loss testing. They used the whatever they did for the strainer head-loss testing. And just run that test for the typical testing facility, and they run that fiber only. And that turned out to be very conservative.

And then they also used the in-line filtration system to make sure that they capture all the fibers bypassing the stainer surface. And then the staff actually looks back at that test facility and audited the actual testing. And we found that the testing was conducted following the approved testing protocol.

And then the key number here is 6.8 grams fiber per assembly was predicted by KHNP. And, however, just keep that in mind. For from staff review we have 15 grams is allowed and you don't need to do additional testing analysis. But now they performed the unit bypass testing demonstrating they only have 6.8 grams per assembly fiber. So that's really conservative from the perspective of the fiber debris would get into the core because simply they use this metallic insulation.

All right, next slide.

Available driving head. We audited their initial calculation based on our comments. And they did identify all the limiting core flow condition. They assumed a conservative debris arrival time. And then on top of that they recalculated the driving head is that the available driving head across the core, they actually took very conservative assumption without considering anything in the steam generator side.

And so with that driving head they, based on this driving head they performed the head loss testing.

Next slide.

As KHNP presented this morning, they had a fuel bundle test facility. And then we inspected the test facility early on. And the unit SER did say what they had, we reissued the findings as part of the inspection findings. And then they corrected the design and the manufacture of the flow chamber and introduced additional measures to make sure that the fuel-bundle test facility was designed and then operated properly according to whatever the staff comments through the on-site inspection. We had one week staying there, watching all their tests.

And then so because of that, the results turned out to be much, much, much more margin. And then, on top of that, they were using 15 grams per assembly running the test. They were not using 6.8 or 6.9 grams. So if they used that one, that was probably even lower.

So with that one we, we think that there is sufficient margin in terms of the core DP across the core. And then the debris would not be enough to cause any concern of the blockage in terms of core blockage.

Next slide.

MEMBER SKILLMAN: Shanlai, if we could change back one slide.

MR. LU: Sure.

MEMBER SKILLMAN: Just so you know we're listening, you said full bundle. That was a full-geometry bundle.

MR. LU: A full geometry.

MEMBER SKILLMAN: Half length?

MR. LU: Right. Not the entire core. It's just one bundle.

MEMBER SKILLMAN: Half a bundle.

MR. LU: Yes, half a -- Oh, okay. I thought it was -- Half bundle height? Oh, okay. Yeah, I'm sorry. You're right.

MEMBER SKILLMAN: Making sure we're listening.

MR. LU: Yeah, yeah, yeah, yeah.

MEMBER SKILLMAN: Okay. All right, thank you.

MR. LU: Yeah, yeah. Oh yeah. Well, yeah, okay, yeah, that's right.

So on LOCA DM model and then they followed the PWR Owners' Group and they just follow standard.

They reported the WCAP with the spreadsheet model and which was approved by NRR. And they used that one.

The reason we still have an open item in the SER, not in the slides because of when we wrote it it was part of SER at that time, the final results of RAI responses regarding the LOCA DM model was not coming in yet. And then so we actually state that it's an open item. But as it is today, we don't believe that's a problem anymore.

So we actually asked for their actual spreadsheet model, which is their LOCA deposition model. We performed our confirmatory analysis.

So what it really concluded is that during 30 days on long-term cooling the crud formation shall be less than the thickness limit, which is 50 mills, and the piece, you have the last 800 to get verified. And then the reason we are saying that they are pending on final closure off RAI because right now we still have not really officially told KHNP in writing this is done deal.

So, so based on this one, from our perspective and then as part of Chapter 6 presentation the staff from Containment Branch covered at NPSH, the strainer. And then the entire GSI-191 perspective, from that perspective we believe this plant we don't see a problem anymore. And simply because it has the all metallic installation, plus they are imposing a containment cleanest program to limit the latent debris. And we've got so much margin there and we don't see a problem.

That's the conclusion of the GSI-191, or we call that a post-LOCA long-term cooling in-vessel downstream effects.

Any questions on this part of the presentation?

MEMBER STETKAR: And they do from the Chapter 6 analyses they take credit for 28 pounds of pressure, over-pressure in the containment to maintain that adequate --

MR. LU: Yes. But that's the, that's related to NPSH margin.

MEMBER STETKAR: Right. That's part of the long-term cooling.

MR. LU: Right. That's NPSH margin.

MEMBER STETKAR: That's what you usually have to grapple with.

MR. LU: Yeah, I understand that ACRS has always had issue with people taking the credit of the contained pressure, but I think --

MEMBER STETKAR: Especially for new plants where you can actually --

MR. LU: Right.

MEMBER STETKAR: -- do something about it.

MR. LU: Yeah. I think that that's part of what was presented to you guys at Chapter 6 review; right?

So from the reactor system perspective, this part, the downstream effect, we do not agree with the issue.

That's the conclusion of this part. And did not have the entire whole pad, Chapter 16, as a one conclusion. But I do want to give you just a few words here.

And as I mentioned right at beginning, we had a very, you know, we -- very challenging schedule to finish this one. We conducted this review with initial burst of the spending on resources. And we issued much less number of RAIs. Actually they may still feel the pressure to resolve all those issues. And we understand that. But the number of RAIs is much less than what we average issue to any other new applicant from that system.

We still have about 12 total, 12 including dose, and there are 12 open items there to be closed. And only two of them are still being worked on -- oh, three of them. Dose is one. And then from reactor system there are two. And PCD is being worked on. And then we have RD interactive with KHNP starting from one-and-a-half years before the submittal. And then we start from there.

And then we knew that that's a tough issue for any other certification, we run into similar situation. So we start early on. Now we can see the, probably the end of the tunnel. So it's going to be resolved. And then according to the schedule it's a September time frame. Hopefully, we will get the results and then we can present it to the committee once the staff review, and whether the issues are being closed or not in December or November time frame.

So I think I do want, you know, I do want to say that few words. And those guys have mentioned about that when they are working on the staff's questions being 24 hours because they have the Korean side. And answering our questions, talking to us, and, you know, responding to our RAIs at the end of the day. And then they have a meeting at 8:00 o'clock at night until midnight with us. And we have been doing that almost every two weeks.

And then I think that once you have that group of dedicated people, you can see the issues can be easily resolved sometime.

All right, that's the, that's all the staff's presentation for Chapter 15. Any other comments or, you know, or I understand that you still want to have another session on cooling, so please let us know what exactly specific the issues you want us to talk about. I think that KHNP already presented a testing plan, testing facility. We present our side for the in-vessel downstream evaluation. And also the containment guys finished theirs.

But if you do want to have another session, we definitely will support whatever you need. Okay.

CHAIRMAN BALLINGER: Thank you.

The public line is open? So, as we do usually, are there, is there anybody in the room that would like to make a comment?

Here it comes. Is there anybody on the line that would like to make a comment?

MR. BROWN: There's no one on the line.

CHAIRMAN BALLINGER: No one on the line. Thank you, Theron.

Okay, so then we should go around the table and see if there are members that have to make -- that would like to make or have to make additional comments. Joy?

MEMBER REMPE: I would like, if it's a question on the table, I do think it would be useful to have another meeting to discuss the methodology report. And I'm not sure if you --

CHAIRMAN BALLINGER: What methodology report?

MEMBER REMPE: I don't have the vugraphs in front of me. But the report that you're still the open item about the methodology that you're reviewing the LOCA.

MEMBER MARCH-LEUBA: Large break LOCA?

MEMBER REMPE: Yes. That's what I'm trying to say, large break LOCA. I think we are going to have that; right?

MR. LU: Yes, we are. We are.

MEMBER REMPE: That should be --

CHAIRMAN BALLINGER: I'm not sure when it's scheduled, but I think it's in December or something.

MEMBER MARCH-LEUBA: In December; right?

MR. BROWN: It is scheduled sometime later this year. I'll get it out. I'll send it to you.

MEMBER REMPE: Yeah. And how we're going -- At some point I think I don't know if you're planning to have a full committee meeting and have them present the differences with all the new calculations or how that, where that discussion will occur. That's a question in my mind, too, and what your plan is. But I guess we can discuss that at other places.

I appreciate the presentations from KHNP and the staff. I also wanted to mention that I thought that even though there were some questions asked today, that the SE was well done and that it went through all the assumptions that KHNP made and what the staff had made. And today I appreciated the presentations where they discussed why there were differences in the TRACE calculations versus what the applicant had presented.

And that's helpful to try and understand what's going on. So I wanted to especially thank them for that part. And other than that, I'll turn it over to the next person.

CHAIRMAN BALLINGER: Charlie.

MEMBER BROWN: I will make comments at another opportunity, but not on this subject.

CHAIRMAN BALLINGER: Okay.

MEMBER MARCH-LEUBA: Could you just say "no comment"?

MEMBER BROWN: That was too easy.

CHAIRMAN BALLINGER: Jose.

MEMBER MARCH-LEUBA: Yeah. I'd like to

say that I'm really encouraged that the agency and the staff has the capability of performing all these confirmatory calculations. I mean, being able to put together TRACE models of this complexity in such short period of time is not -- is an achievement that we need to applaud. And I'm really glad that we have that.

I love the fact that for every single AOO, or at least the limiting ones, we don't just at take the word of the applicant, we run a confirmatory and confirm that everything is okay. For the more complex ones like LOCA, we spend more time but we get complete models and complete results, reliable results. So this is a great capability that the agency has.

MEMBER STETKAR: No comment.

MEMBER POWERS: A couple of things. The dispersion of any release that takes place around the main control room seems to me done in an extraordinarily approximate method. While conservatism is built into that and the way they did it, but it's still fundamentally based on a Gaussian plume kind of concept which won't exist in that region.

And so I think I need to understand better how -- that is. And I think maybe the staff needs to assure that a COL applicant validates that analysis for his particular location. Because I mean it's involving an assumption on something that's non-physical. So it may be a point of alerting the COL applicant that he needs to understand how they made this analysis to make sure it's working for his plant, at his location. It will be affected by where he puts the plant as much as how the plant is configured.

So it may just a COL application action item in there.

The sump calculation, sump pH calculation is something I'm going to have to research some more. We did a pretty standard, nice job. They based the pH calculation on a thermodynamic model Stihl gas mix which was a recognized model. Not one of my favorites because it's classic spaghetti code, but that's --

CHAIRMAN BALLINGER: It's also 60 years old.

MEMBER POWERS: Well, so is most of the chemistry. So it's okay to use that old code.

But the problem is you have a dynamic system here where you're getting radiolysis, which is a kinetic effect. They seem to have recognized the radiolysis, the gas phase to create nitric acid. Radiolysis has tables to create hydrochloric acid. They did not mention that they will get radiolytic production of carboxylic acids due to any organics that are in the water. We've certainly seen those in the RTF tests that are done in Canada.

The other issue, of course, is the sump pH is dominated by buffering effects of boric acid trisodium phosphate. And it's not apparent how the applicant took into account the depletion of the buffering capacity as those phosphates and borates reacted with various contaminants that come into the sump as a result of these accidents.

Then there are various arcane features. The molarities of the solution are high enough that both coulombic effects and short-range effects affect the chemical activity of the solutes in making the analysis. And it's not apparent to me exactly how they handled that in the curves.

So I think probably I need to do a little more background work on this to just understand exactly what they did. And I may need help from KHNP to understand exactly what they did. Because there's not this kind of detail in the chapter.

CHAIRMAN BALLINGER: Thank you. Dick.

MEMBER SKILLMAN: Thank you, Ron. I want to thank KHNP and the staff for two very solid days of presentations. And I agree with my colleagues: well done for the confirming analysis that the staff has done to check what KHNP has done.

One comment of specificity. This has to do with Chapter 15.6.5.2, large break LOCA. And this is the figures of the core water level and the downcomer level. And I was kind of taken aback when Jose discovered that the datum weren't the same datum for the portrayal of those levels. And I just hadn't even though of that when I looked at the image.

So that sparks in my mind a need for uber-caution when we see figures, particularly of what may be static or dynamic water levels, to make sure we're using the same zero point for the level. So, I think at a minimum those figures need to be amended. But all of us need to be on guard to make sure that when we see a portrayal like that we understand what the datum, zero datum is.

MR. LU: You are talking about DCD or you're talking about a staff SER?

MEMBER SKILLMAN: Actually it's both.

MR. LU: Okay. All right.

CHAIRMAN BALLINGER: And the slides.

MEMBER SKILLMAN: And the slides. It was in the slides.

MR. LU: All right. So it's the large break LOCA section?

MEMBER SKILLMAN: Yeah.

MR. LU: Okay.

MEMBER SKILLMAN: And it was a slide that showed the core water level compared with the downcomer water level. And it was the 3 meters or 6 meters, 20 feet, .434, that's 10 psi. That's a lot of pressure drop.

CHAIRMAN BALLINGER: it was actually more than one figure.

MEMBER SKILLMAN: Yeah.

CHAIRMAN BALLINGER: I think at least two in each one.

MR. LU: So part of the TRACE analysis or it's really -- or it's a slide; right?

MEMBER SKILLMAN: It's in the slides.

MR. LU: Okay.

MEMBER MARCH-LEUBA: Yeah. Either the slide needs to be corrected or there needs to be a label saying that the zero is 2 meters off.

MEMBER SKILLMAN: It was KHNP slides.

MR. LU: Okay.

MEMBER SKILLMAN: Thank you. That's all I have.

MR. LU: Which page of KHNP slides?

CHAIRMAN BALLINGER: There's two. Two

slides, two slides in a row.

MR. LU: Thirty-three?

MEMBER MARCH-LEUBA: Thirty-three and 34, I think.

MR. LU: Thirty-four. Okay.

MEMBER MARCH-LEUBA: No, not 34. Thirty-three.

CHAIRMAN BALLINGER: Steve.

MEMBER SCHULTZ: Yeah. I just wanted to summarize a couple of things, and certainly that the work by KHNP in developing and documenting the design and the capabilities in these areas, and by the staff in reviewing and confirming the design and its performance or requirements, this is really converging towards a robust design evaluation. Today's presentations were exceptional in that area and demonstrate the work that has been done as was described.

And the open items yet to be resolved, they are worthy, certainly, of additional attention to develop a full understanding and agreement between the KHNP applicant's work and that of the staff. So keep at it is what I would say. And thank you for the presentations.

Just to come back to it one more time, the fuel thermal conductivity degradation. Just a reminder, you know, this affects the steady state fuel temperature, which means it just doesn't affect steady state performance or the transient performance. Isotopic cross sections and physics parameters are affected. That affects then the steady state fuel performance which affects the transient analysis input. The transient analysis and temperatures, the stored energy, and ultimately, realistic source terms.

So it's all, it's all --

MR. LU: All over the place.

MEMBER SCHULTZ: -- engaged. I know we're addressing this in every which way we can. Don't expect a perfect solution by November/December. But just to keep in mind that all of these things are affected by that and ought to be kept in mind as we move forward to the overall evaluation.

Thank you.

MR. LU: Right.

CHAIRMAN BALLINGER: Professor Emeritus Corradini.

MEMBER CORRADINI: Thank you professor soon-to-be emeritus.

CHAIRMAN BALLINGER: Amen to that, brother.

MEMBER CORRADINI: I wanted to thank the

applicant and the staff for their presentations today.

I do think that we have to phase the scheduling of the methodology with TCD being considered as Steve had indicated with the recalculation of some of the key accidents or transients. And if that's going to be in six months, so be it. But I guess I'd leave it to Ron and Chris to decide how you want to phase that, whether it would be the Thermohydraulics Committee or this APR committee, since a lot of us are the same.

But thanks to them. I think this was a good introduction to the whole range of accident analysis that's been done by the applicant.

That's all.

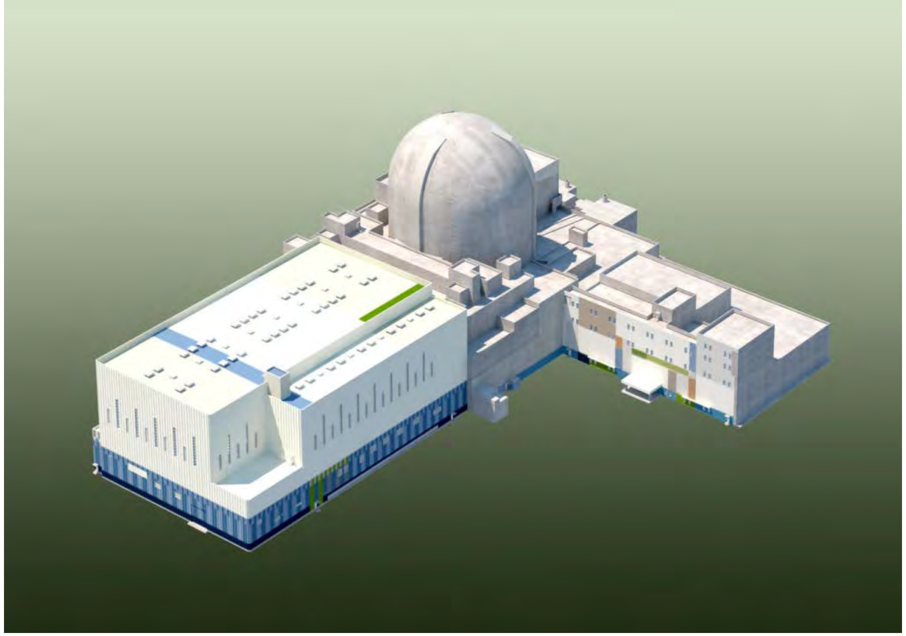
CHAIRMAN BALLINGER: I'd like to express my thanks, too. And it's long -- I can't concentrate.

A great job for the last two days. A long, hard, slow task today and yesterday, but well worth it. So we thank you very much.

And with that we are adjourned.

(Whereupon, the above-entitled matter went off the record at 5:30 p.m.)

APR1400 DCA
Chapter 15: Transient and Accident Analyses



KEPCO/KHNP
May 19, 2017

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Overview of Chapter 15

❑ List of Submitted Documents

Document No.	Title	Revision	Type
APR1400-K-X-FS-14002-P/NP	APR1400 Design Control Document Tier 2: Chapter 15 Transient and Accident Analyses	1	DCD
APR1400-K-X-FS-14001-P/NP	APR1400 Design Control Document Tier 1	1	DCD
APR1400-F-A-TR-12004-P	Realistic Evaluation Methodology for Large-break LOCA of the APR1400	0	ToR
APR1400-F-A-NR-14001-P	Small Break LOCA Evaluation Model	1	TeR
APR1400-Z-A-NR-14006-P	Non-LOCA Safety Analysis Methodology	1	TeR
APR1400-Z-A-NR-14014-P	ATWS Evaluation	0	TeR
APR1400-F-A-NR-16003-P	Loop Seal Reformation	0	TeR
APR1400-F-A-NR-16004-P	Boron Dilution Analysis for APR1400	0	TeR
APR1400-F-A-NR-14003-P	Post-LOCA Long Term Cooling Evaluation Model	1	TeR

15.1 Increase in Heat Removal by the Secondary System

15.1.1 Decrease in Feedwater Temperature (AOO)

15.1.2 Increase in Feedwater Flow (AOO)

15.1.3 Increase in Steam Flow (AOO)

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve (AOO)

15.1.5 Steam System Piping Failure Inside and Outside the Containment (PA)

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve (AOO)

NON-PROPRIETARY

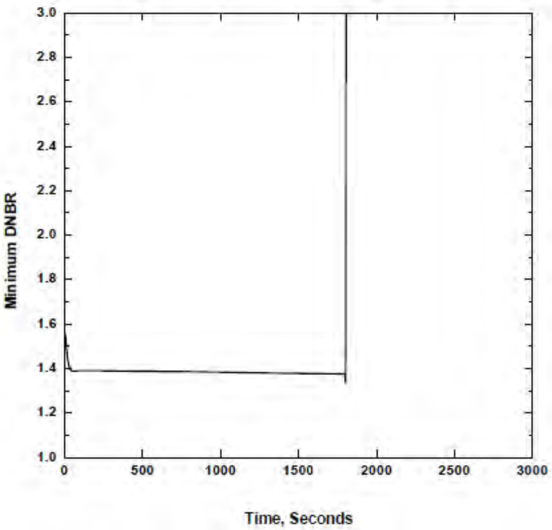
❑ Main steam flow increase

- No more than 11% increase over the nominal full-power steam flow rate
- Resulting in a decrease in core inlet temperature
- Concerning Minimum DNBR

❑ Reactor trip override (RTO) mode failure is assumed as a single failure.

❑ Analysis result

- MDNBR remains above fuel design limit.



15.1.5 Steam System Piping Failure Inside and Outside the Containment (PA)

❑ Excessive RCS cooldown

- Steam line break (SLB)
- Core reactivity increase
- Degradation in fuel cladding performance

❑ SLB analysis cases are chosen

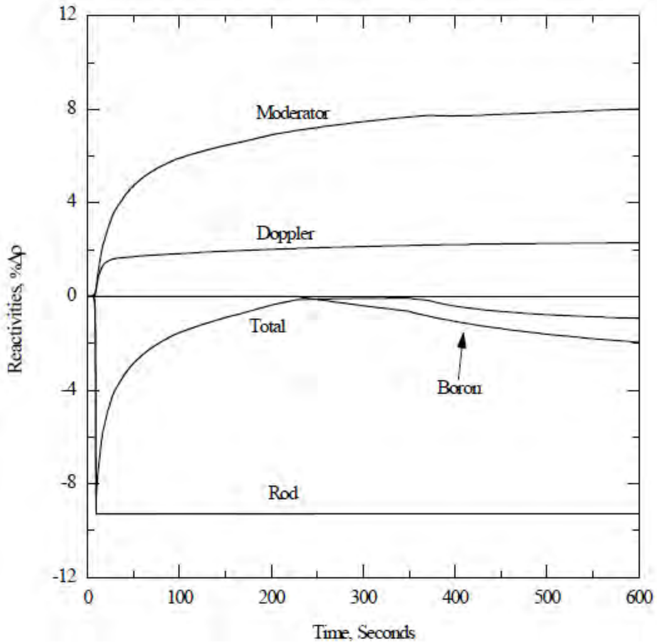
- To maximize potential for a post-trip return to power (RTP)
- To maximize potential for degradation in fuel cladding performance

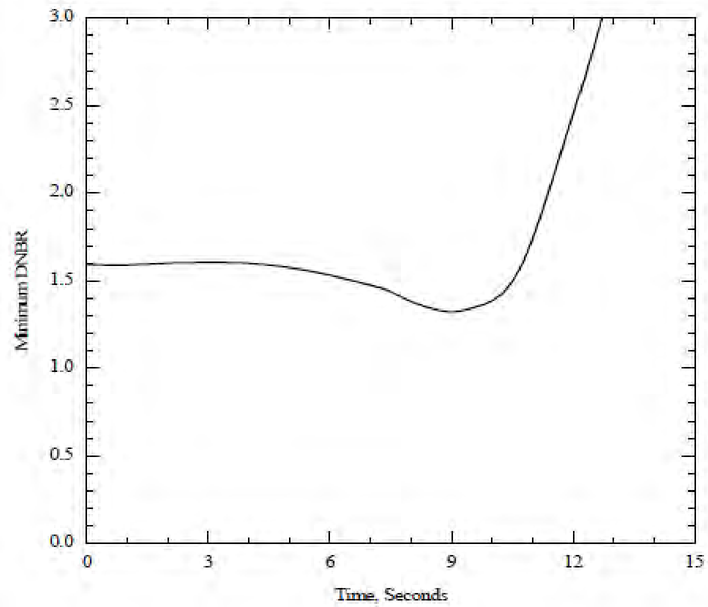
15.1.5 Steam System Piping Failure Inside and Outside the Containment (PA)

NON-PROPRIETARY

- MSIV or SI pump failure are considered as a single failure.

- Analysis result
  - Post-trip RTP does not occur.
  - MDNBR remains above fuel design limit.





15.2 Decrease in Heat Removal by the Secondary System

15.2.1 Loss of external load (AOO)

15.2.2 Turbine trip (AOO)

15.2.3 Loss of condenser vacuum (AOO)

15.2.4 Closure of main steam isolation valve (AOO)

15.2.5 Steam pressure regulator failure (not applicable to the APR1400)

15.2.6 Loss of nonemergency ac power to the station auxiliaries (AOO)

15.2.7 Loss of normal feedwater flow (AOO)

15.2.8 Feedwater system pipe break inside and outside the containment (PA)

15.2.3 Loss of condenser vacuum (AOO)

❑ Loss of condenser vacuum (LOCV) analysis assumes

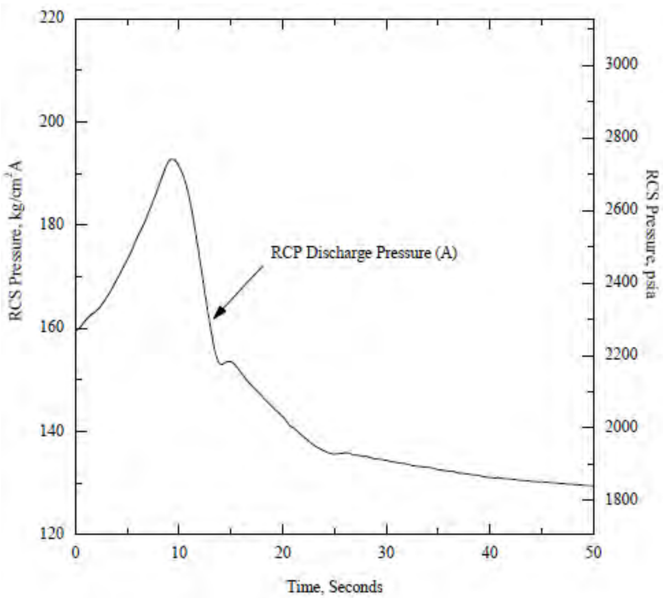
- Immediate cessation of feedwater flow
- Turbine trip immediately coincident with LOCV

❑ Decrease in RCS cooldown

- Increase in temperature and pressure of RCS
- System peak pressure is concerned.

❑ Analysis result

- RCS and main steam system pressures increase but remain below acceptance criteria.

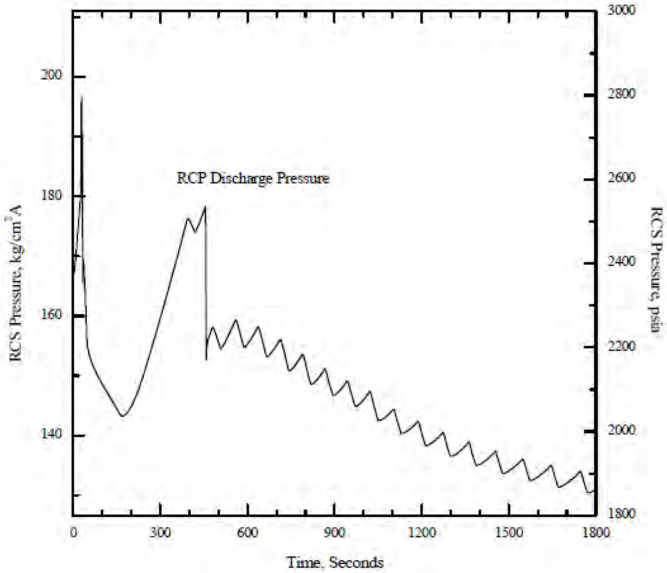


15.2.8 Feedwater system pipe break inside and outside the containment (PA)

- **Rapid depletion of affected SG liquid mass**
  - Reducing heat transfer capability
  - Rapid RCS heat up and pressurization
  - System peak pressure is concerned.

- **Feedwater line break (FLB) at economizer line**
  - Spectrum of break sizes is analyzed to determine limiting break size

- **Analysis result**
  - RCS and main steam system pressures increase but remain below acceptance criteria.



15.3 Decrease in Reactor Coolant System Flow Rate

15.3.1 Loss of Forced Reactor Coolant Flow (AOO)

15.3.2 Flow Controller Malfunctions
(not Applicable to the APR1400)

15.3.3 Reactor Coolant Pump Rotor Seizure (PA)

15.3.4 Reactor Coolant Pump Shaft Break (PA)

15.3.1 Loss of Forced Reactor Coolant Flow

NON-PROPRIETARY

❑ Description

- Complete loss of forced reactor coolant flow event is the most limiting
- Simultaneous loss of electrical power to all RCPs
- Decrease in margin to DNB, increase system pressure

❑ Assumptions

- Simultaneous turbine trip and loss of feedwater flow
- Select the most limiting initial conditions for each aspect

❑ Analysis Code

- COAST, HERMITE, CETOP, CESEC-III ; NRC approved codes

❑ Analysis Results

- Max. RCS and SG pressure < 110% of the design values
- Minimum DNBR > DNBR limit (1.29)

15.3.3-15.3.4 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

NON-PROPRIETARY

❑ Description

- Seizure of an RCP rotor / shaft break
- Decrease in margin to DNB, increase system pressure

❑ Assumptions

- Consequential loss of feedwater flow, coastdown of remaining RCPs with LOOP
- Select the most limiting initial conditions for each aspect

❑ Analysis Code

- COAST, HERMITE, CETOP, TORC, CESEC-III ; NRC approved codes

❑ Analysis Results

- Max. RCS and SG pressure < 110% of the design values
- Doses at the site boundary < their allowable criteria limits

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low-Power Startup Condition (AOO)

15.4.2 Uncontrolled Control Element Assembly Withdrawal at Power (AOO)

15.4.3 Control Element Assembly Misoperation (AOO)

15.4.4 Startup of an Inactive Reactor Coolant Pump (AOO)

15.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate (not applicable to the APR1400)

15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (AOO)

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (AOO)

15.4.8 Spectrum of CEA Ejection Accidents (PA)

15.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low-Power Startup Condition

NON-PROPRIETARY

❑ Description

- Failure in the CEDM, CEDMCS, RRS, or operator error
- Adds reactivity to the reactor core
- Increase core power, core heat flux, reactor coolant temperature, and system pressure

❑ Assumptions

- 0.001% of rated thermal power
- Maximum reactivity insertion rate
- Select the most limiting initial conditions

❑ Analysis Code

- CESEC-III, CETOP ; NRC approved codes

❑ Analysis Results

- Minimum DNBR > DNBR limit (1.29)
- Peak linear heat generation rate < 20 kW/ft

15.4.2 Uncontrolled CEA Withdrawal at Power

❑ Description

- Failure in the CEDM, CEDMCS, RRS, or operator error
- Adds reactivity to the reactor core
- Increase core power, core heat flux, reactor coolant temperature, and system pressure

❑ Assumptions

- 102% of rated thermal power
- Maximum reactivity insertion rate
- Select the most limiting initial conditions

❑ Analysis Code

- CESEC-III, CETOP ; NRC approved codes

❑ Analysis Results

- Minimum DNBR > DNBR limit (1.29)
- Peak linear heat generation rate < 20 kW/ft

15.4.3 Control Element Assembly Misoperation

❑ Description

- Dropped CEA or CEA subgroup / Statically misaligned CEA / Single CEA withdrawal
- 4-Finger CEA drop is the most limiting case
- Increase in the hot pin radial peaking factor

❑ Assumptions

- 102% of rated thermal power
- Maximum radial peak distortion
- Select the most limiting initial conditions

❑ Analysis Code

- CESEC-III, CETOP ; NRC approved codes

❑ Analysis Results

- Minimum DNBR > DNBR limit (1.29)
- Peak linear heat generation rate < 20 kW/ft

15.4.4 Startup of an Inactive Reactor Coolant Pump

❑ Description

- Startup of an Inactive RCP
- Increase or decrease core average coolant temperature
- Increase in core reactivity

❑ Assumptions

- MODE 3 to MODE 6 condition
- Maximum primary to secondary temperature difference
- Most positive or post negative ITC

❑ Analysis Codes

- N/A

❑ Analysis Results

- No return to critical core condition

15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System

NON-PROPRIETARY

Description

- CVCS malfunction or operator error
- Decrease coolant boron concentration
- Increase core reactivity

Assumptions

- Maximum dilution flow rate
- Minimum RCS mixing volume, minimum shutdown margin
- Maximum critical boron concentration, minimum inverse boron worth

Analysis Codes

- N/A

Analysis Results

- More than 30 minutes operator action time is available

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

Description

- Interchange fuel assemblies in a core
- Core power distribution is affected

Assumptions

- Considering of a spectrum of misloading

Analysis Codes

- ROCS ; NRC approved code

Analysis Results

- Peaking factor would not increase more than that assumed in the CEA drop event

15.4.8 Spectrum of Control Element Assembly Ejection

NON-PROPRIETARY

❑ Description

- Mechanical failure of the CEDM housing or its associated nozzle
- Adds reactivity to the reactor core for a short period of time

❑ Assumptions

- Maximum ejected rod worth
- Minimum effective delayed neutron fraction
- Minimum Doppler coefficient

❑ Analysis Codes

- STRIKIN-II, CETOP, and CESEC-III ; NRC approved codes

❑ Analysis Results

- Max. RCS pressure < “Service Limit C” as defined in the ASME Code
- Peak radial average fuel enthalpy < 230 cal/g
- No fuel melting
- Doses at the site boundary < their allowable criteria limits

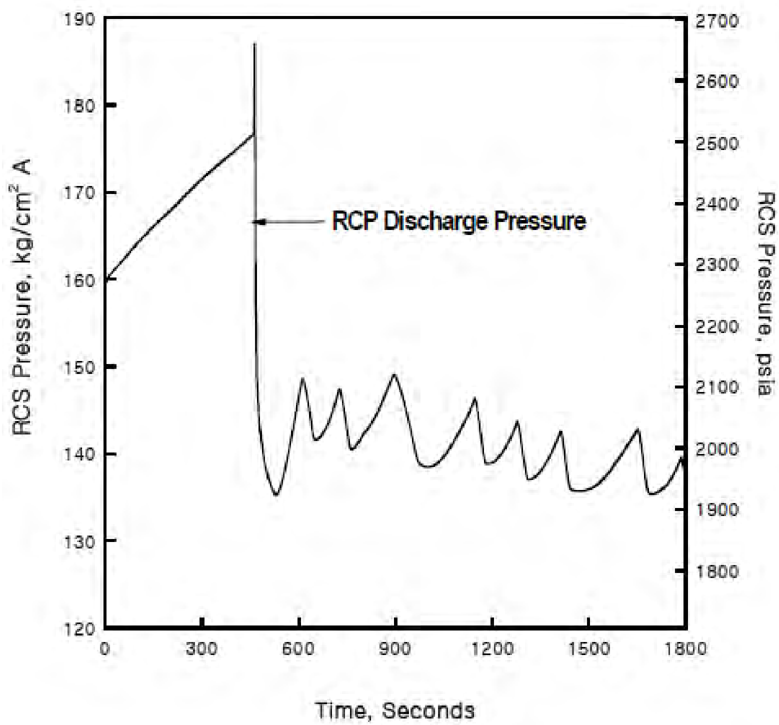
15.5 Increase in Reactor Coolant Inventory

15.5.1 Inadvertent Operation of the Emergency Core Cooling System that Increases the Reactor Coolant Inventory (AOO)

15.5.2 Chemical and Volume Control System Malfunction that Increases the Reactor Coolant Inventory (AOO)

15.5.2 Chemical and Volume Control System Malfunction (AOO)

- **Pressurizer level control system (PLCS) malfunction**
  - Maximum charging flow and minimum letdown flow
  - Pressure transient due to RCS coolant inventory increase, not to thermal expansion
  - No significant power and coolant temperature transient prior to reactor trip
- **System pressure remains below acceptance criteria.**



15.6 Decrease in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve (PA)

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment (AOO)

15.6.3 Steam generator tube failure (PA)

15.6.4 Radiological consequences of main steam line failure outside the containment for a boiling water reactor (not applicable to the APR1400)

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment (AOO)

- ❑ **Double-ended break of the letdown line outside the containment**
  - Results in the largest release of reactor coolant outside the containment
  - Radiological release
  - RCS depressurization

- ❑ **Operator takes action to terminate the primary system fluid loss 30 minutes after initiation of the event**

- ❑ **Analysis result**
  - MDNBR remains above fuel design limit.
  - Radiological acceptance criteria are satisfied.

15.6.3 Steam Generator Tube Failure (PA)

- **Penetration of the barrier between the RCS and the main steam system**
  - Radiological release
  - RCS depressurization

- **Double-ended rupture of a SG U-tube at full-power conditions**
- **Primary-to-secondary leakage and SG release are used as input to dose calculation**

- **Analysis result**
  - MDNBR remains above fuel design limit.
  - Radiological acceptance criteria are satisfied.

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks

NON-PROPRIETARY

❑ Large Break LOCA

- Topical Report, 'Realistic Evaluation Methodology for Large-Break LOCA of the APR1400', (APR1400-F-A-TR-12004)
- CAREM : Code Accuracy based Realistic Evaluation Model
- Revisions of Topical Report and DCD Section 15.6.5 LBLOCA are on going to reflect the Thermal Conductivity Degradation (TCD) issue

❑ Small Break LOCA

- Analysis results confirmed the satisfaction of acceptance criteria

❑ Long-term Cooling

- Analysis results confirmed the satisfaction of acceptance criteria

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

NON-PROPRIETARY

- **Code of Federal Regulations**
  - 10 CFR 50.46
  - Acceptance criteria for ECCS for light water NPR

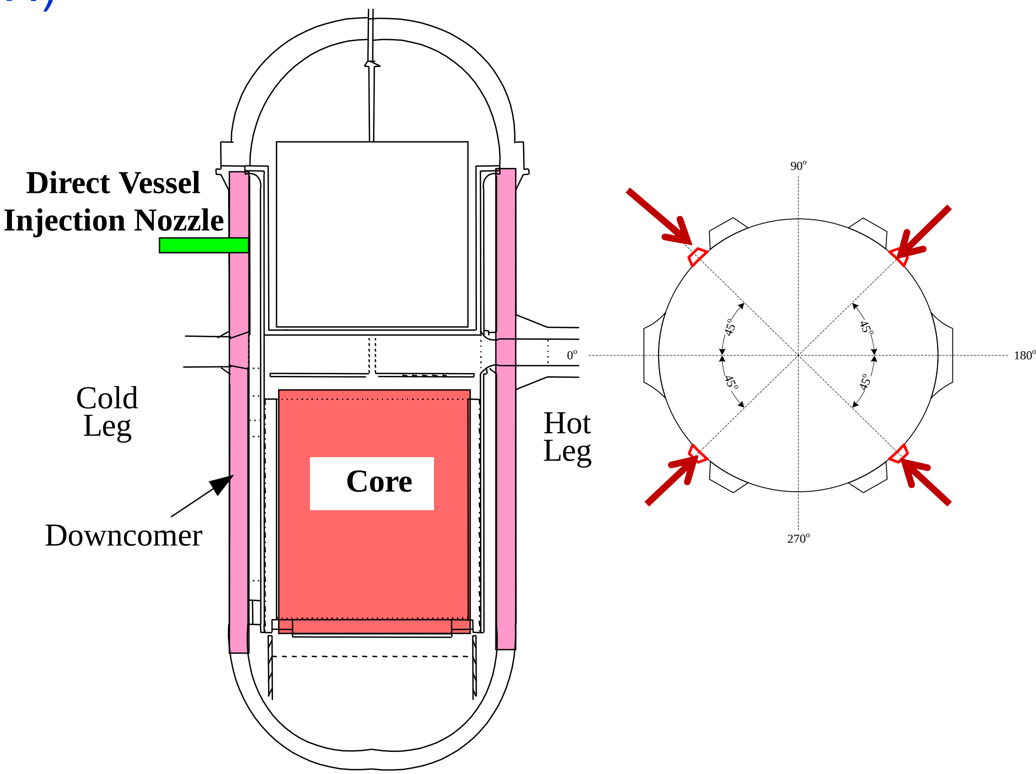
- **Regulatory Bases**
  - RG 1.157, BE calculations of ECCS performance
  - RG 1.206, Combined license applications for NPP
  - NUREG-0800, SRP for the review of safety analysis reports
  - NUREG-1230, Compendium of ECCS research for realistic LOCA analysis
  - NUREG-5249, Quantifying reactor safety margins: application of code scaling, applicability and uncertainty evaluation methodology to a LBLOCA (CSAU)

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.2 Sequence of Events and Systems Operation

❑ Description of LBLOCA

- APR1400 SIS consists of 4 mechanically independent trains
- Direct vessel injection (DVI)
- A safety injection pump and a safety injection tank are installed in each train
- All the ECC water is injected into the upper annulus of reactor pressure vessel



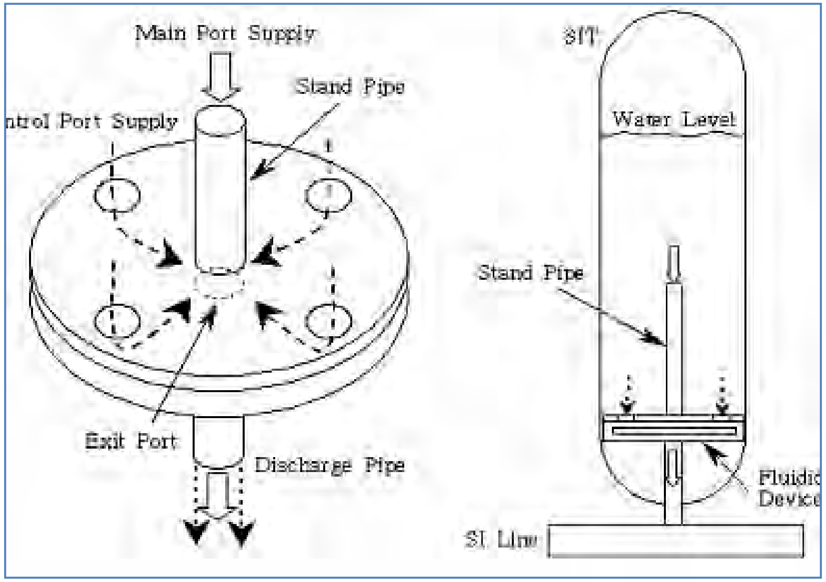
15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

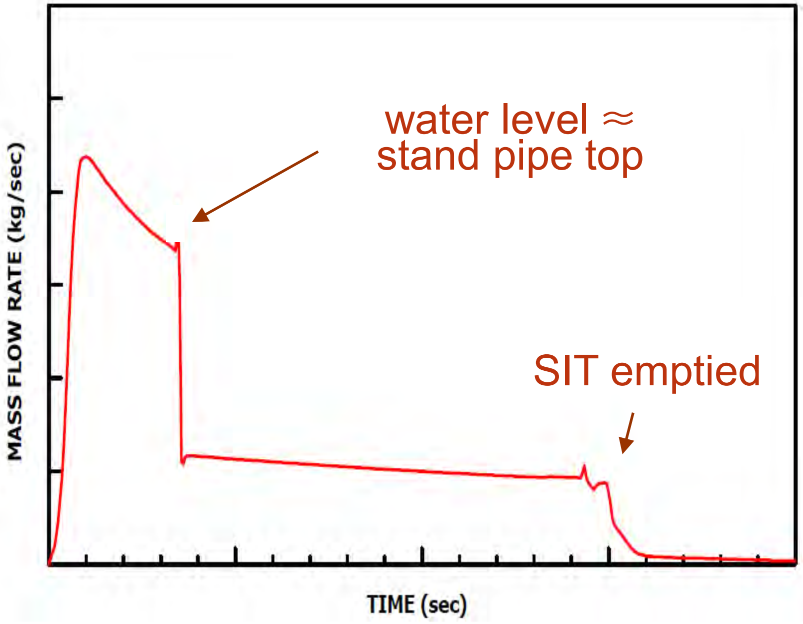
NON-PROPRIETARY

15.6.5.2 Sequence of Events and Systems Operation

❑ Description of LBLOCA (cont'd)

- Fluidic device in SIT regulates the injection flow rate and enhances removal of decay heat in early reflood phase
- Topical Report, 'Fluidic Device Design' (APR1400-Z-M-TR-12003-P-A)





15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

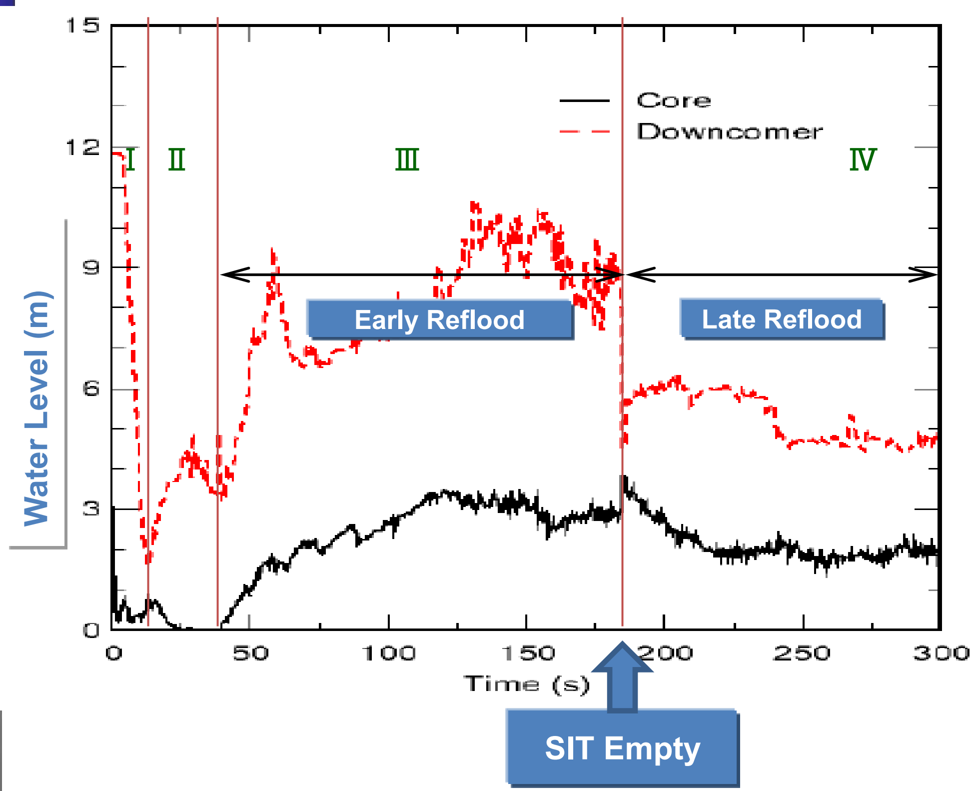
15.6.5.2 Sequence of Events and Systems Operation

- Description of LBLOCA (cont'd)
  - RELAP5/Mod3.3K & CONTEMPT4/Mod5
    - ✓ RELAP5/Mod3.3K: Thermal-hydraulic analysis
    - ✓ CONTEMPT4/mod5: Containment back pressure calculation
    - ✓ Two codes exchange mass/energy and pressure as boundary conditions
  - CAREM developed based on the CSAU (NUREG-5249)
    - ✓ Uncertainties are quantified by non-parametric statistics and SRS calculation
    - ✓ Introduce experimental data covering for confirmation of uncertainty parameters and their ranges & distributions

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.2 Sequence of Events and Systems Operation

- Description of LBLOCA (cont'd)
  - LBLOCA scenario specification for APR1400



I : Blowdown (~ 20 sec)
break open ~ initiation of SIT

II : Refill (~ 35 sec)
until water level is reached to the bottom of active core

III : Early Reflood (~ 190 sec)
until SIT empty

IV : Late Reflood
after SIT empty

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

NON-PROPRIETARY

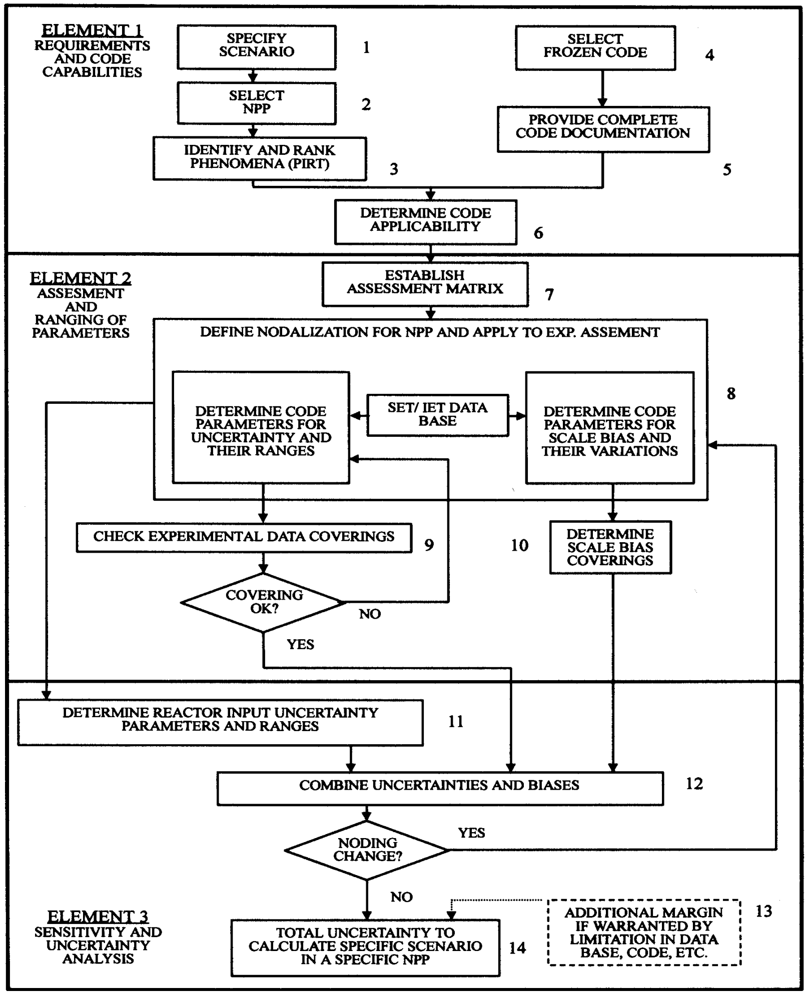
15.6.5.3 Core and System Performance

❑ **LBLOCA methodology: CAREM**

- CAREM consists of 3 elements and 14 steps as in CSAU
- Step 9 checks Experimental Data Covering (EDC) using the uncertainty parameters determined in step 8. If it fails, step 8 repeats until the covering is satisfied
- Non-parametric statistics is used in EDC as well as in plant calculations
- References:

  - Nuclear Tech. V.148, 3, 2004

  - Nuclear Tech. V.158, 2007



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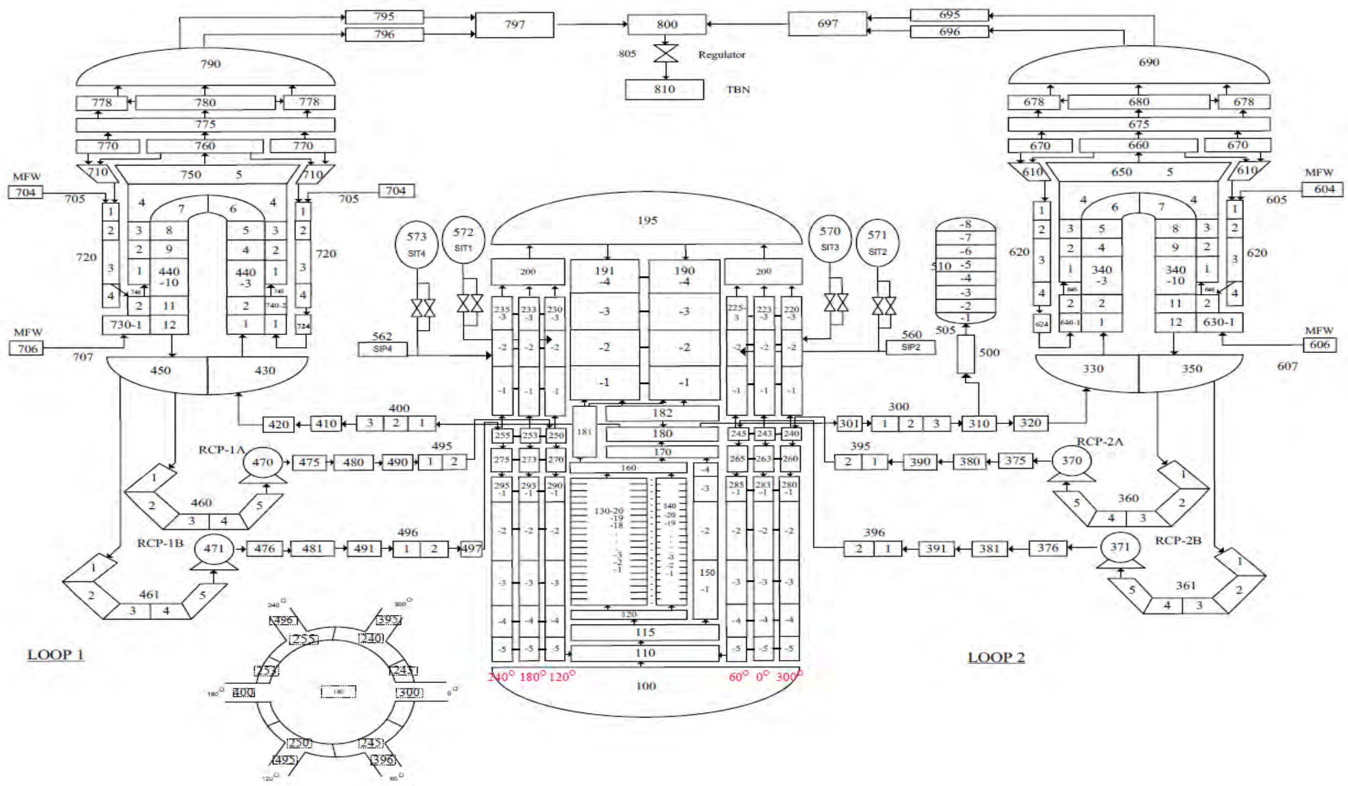
15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.3 Core and System Performance

- **LBLOCA methodology: CAREM**
  - Core is modeled with 2 hydraulic channels and 20 axial nodes
  - Downcomer is modeled with 6 channels and 10 axial nodes

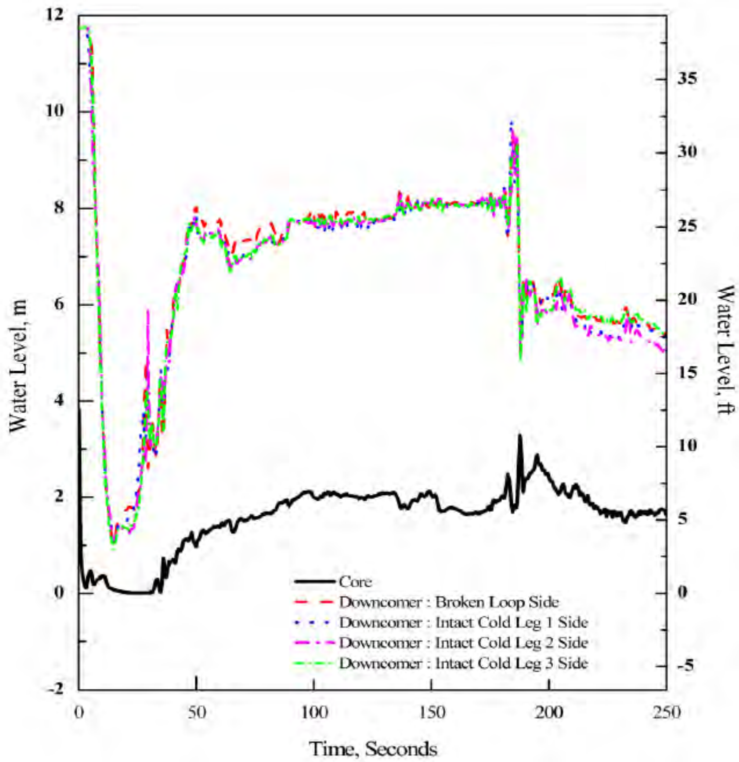


15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

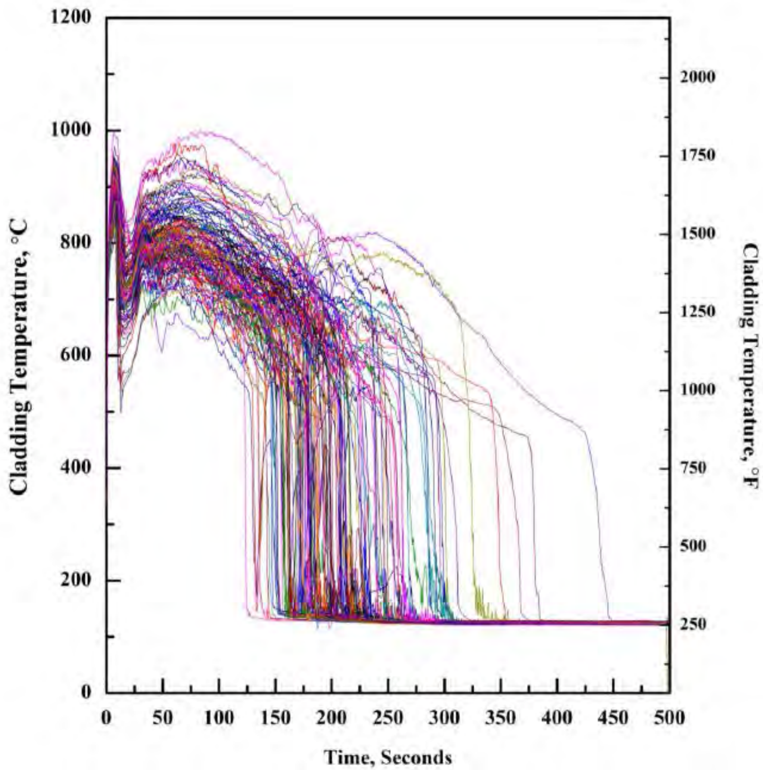
15.6.5.3 Core and System Performance

❑ LBLOCA Results

- 100% double-ended guillotine break in pump discharge leg



Water Levels in Core and Downcomer



SRS Peak Cladding Temperatures

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.3 Core and System Performance

❑ LBLOCA Results

- Licensing PCT
  $= PCT_{95/95} + \Delta PCT_{Bias\ results} + \Delta PCT_{Additional}$ (10 °C)
  < 1,204.4 °C (2,200 °F)

- The satisfaction of acceptance criteria will be confirmed for APR1400 design

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.2 Sequence of Events and Systems Operation

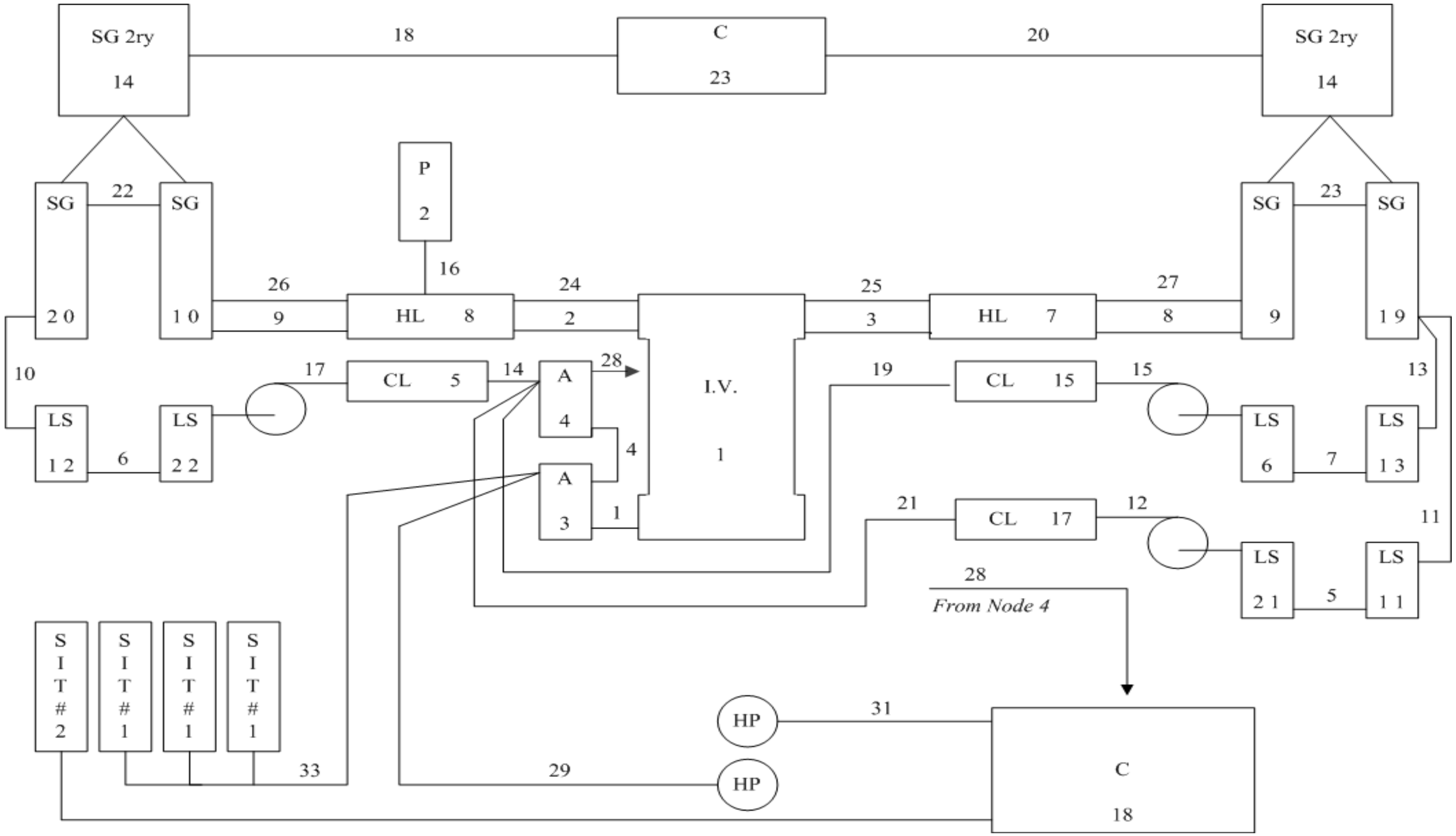
❑ Description of SBLOCA

- CENPD Appendix K Evaluation Model
  - ✓ "Calculative Methods for the C-E Small Break LOCA Evaluation Model," CENPD-137P (1974) and Supplement 1 (1977)
- Multi-Code System
  - ✓ CEFLASH-4AS: Blowdown hydraulics and full transient pressure decay
  - ✓ COMPERC-II: Refill / reflood hydraulics
  - ✓ STRIKIN-II: Hot rod calculation during blowdown period
  - ✓ PARCH EM: Hot rod calculation during pool boiling period
- Used in System80+ CESSAR-DC SBLOCA Analysis
- APR1400 design is the same as System80+ in terms of loop arrangement (2X4) and safety injection system (DVI) design

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.3 Core and System Performance

- SBLOCA methodology: CEFLASH-4AS



15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

NON-PROPRIETARY

15.6.5.3 Core and System Performance

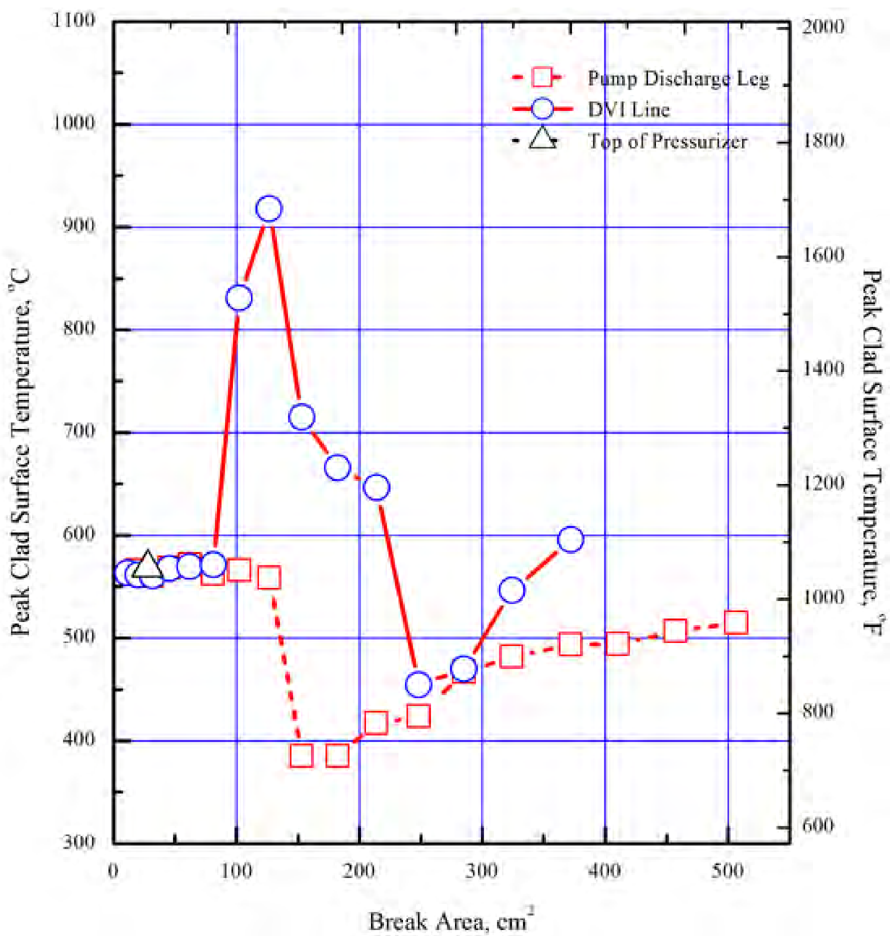
❑ SBLOCA Analysis

- Initial Power = 1.02 x Rated Thermal Power
- LOOP (Loss of offsite power) + Worst Single Failure of ECCS (Emergency Core Cooling System) Equipment
- For the DVI line break, 15 cases were analyzed
- For the Cold Leg break, 17 cases were analyzed
- In addition, one break at the Top of Pressurizer was analyzed and one rupture of In-core Instrument tube was evaluated

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.3 Core and System Performance

❑ SBLOCA Results

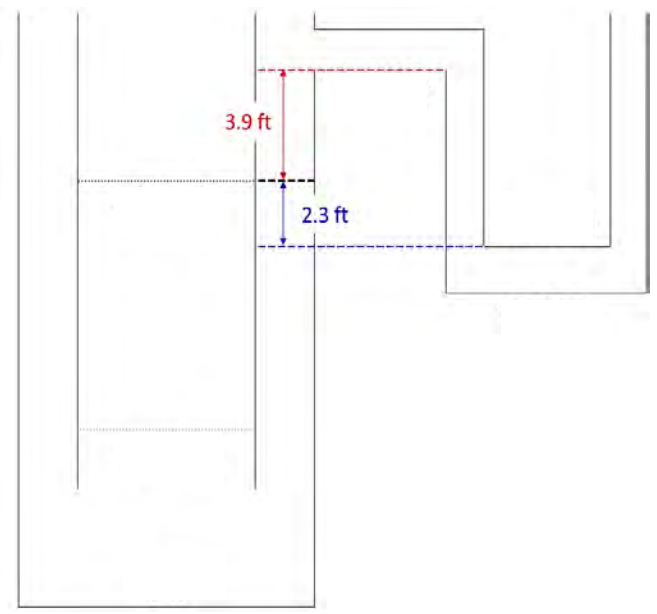


- Limiting PCT : DVI line break
- The result of SBLOCA satisfies acceptance criteria

Loop Seal Clearing and Reformation

❑ Background and Relevant RAI

- Loop Seal Reformation due to ECCS injection during the long term cooling phase of a LOCA can cause suppression of the two-phase mixture level in the reactor core
- If this level drops below the top of the active fuel, cladding heat-up and oxidation can occur
- The distance from top of the core to bottom of loop seal is only about 2 ft
- The Loop Seal Reformation calculation for several break sizes was performed using CENPD SBLOCA methodology



KEPCO KHNP

Loop Seal Clearing and Reformation

❑ Analysis Results

- Loop seal reformation shows slight core uncovery intermittently
- The PCT caused by Loop Seal Reformation remains below 800°F

Post-LOCA Boron Dilution Analysis

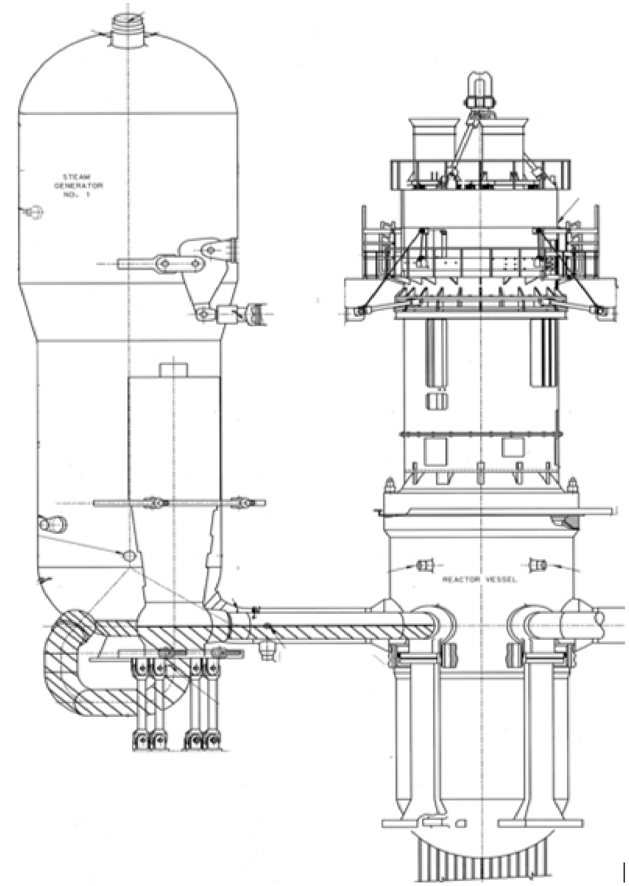
❑ Background and Relevant RAI

- Following a LOCA, a slug of water can be formed in the loop seal by the condensed steam in S/G tubes
- The slug enters the vessel through a cold leg and then travels along the downcomer. Again the slug moves into a lower plenum and it turns upward to enter the core
- During this period, it may cause a reactivity excursion if the water slug is not sufficiently mixed with the borated water in the RCS
- It was requested that the core should not reach a recriticality when the boron dilution accident occurs

Post-LOCA Boron Dilution Analysis

❑ Analysis Results

- Two cases were studied:
  - ✓ Restart of one RCP
  - ✓ Start of natural circulation
- Mixing evaluation shows that the downcomer and lower plenum water mixes well with water from the loop seal
- KHNP has demonstrated that both of the restart of one RCP and the initiation of natural circulation will not cause core recriticality



15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

NON-PROPRIETARY

15.6.5.2 Sequence of Events and Systems Operation

❑ Description of Post LOCA Long-Term Cooling

- The post-LOCA long-term phase
  - ✓ To avoid the precipitation of boric acid
  - ✓ Operator action is needed
- Large break LTC
  - ✓ Heat removal by the safety injection flow
  - ✓ Boron precipitation can occur in the core
  - ✓ Simultaneous injection
- Small break LTC
  - ✓ Heat removal by SG cooldown until shutdown cooling initiation
  - ✓ Boron precipitation concerns are not possible: natural circulation

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.3 Core and System Performance

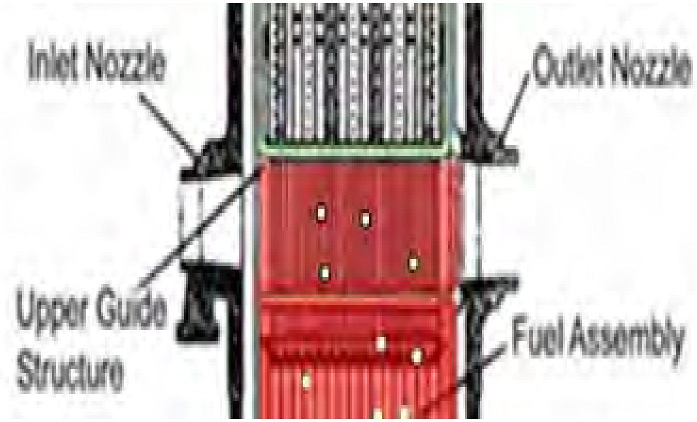
- Post LOCA LTC Evaluation Model
  - CENPD conservative evaluation model
    - ✓ "Post-LOCA Long Term Cooling Evaluation Model," CENPD-254-P-A (1980)
  - Code system
    - ✓ CELDA: Long Term depressurization and refill of the RCS
    - ✓ NATFLOW: Flowrates, pressure and temperature in primary system
    - ✓ CEPAC: S/G cooldown performance
    - ✓ BORON: Transient boric acid concentration in the core
  - NRC approved 'Interim Method' was adopted (Waterford Unit 3, ML050490396)
    - ✓ The interim method provided resolution of issues to CENPD-254

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

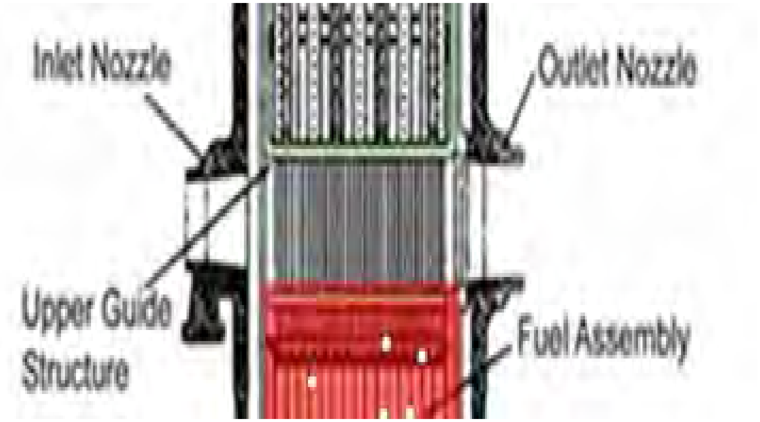
NON-PROPRIETARY

15.6.5.3 Core and System Performance

- Post LOCA LTC Evaluation Model(cont'd)
  - Applying Mixing Volume Change
    - ✓ The limiting mixing region in boron precipitation analysis is changed from top of hot leg to bottom of hot leg



Top of Hot-leg



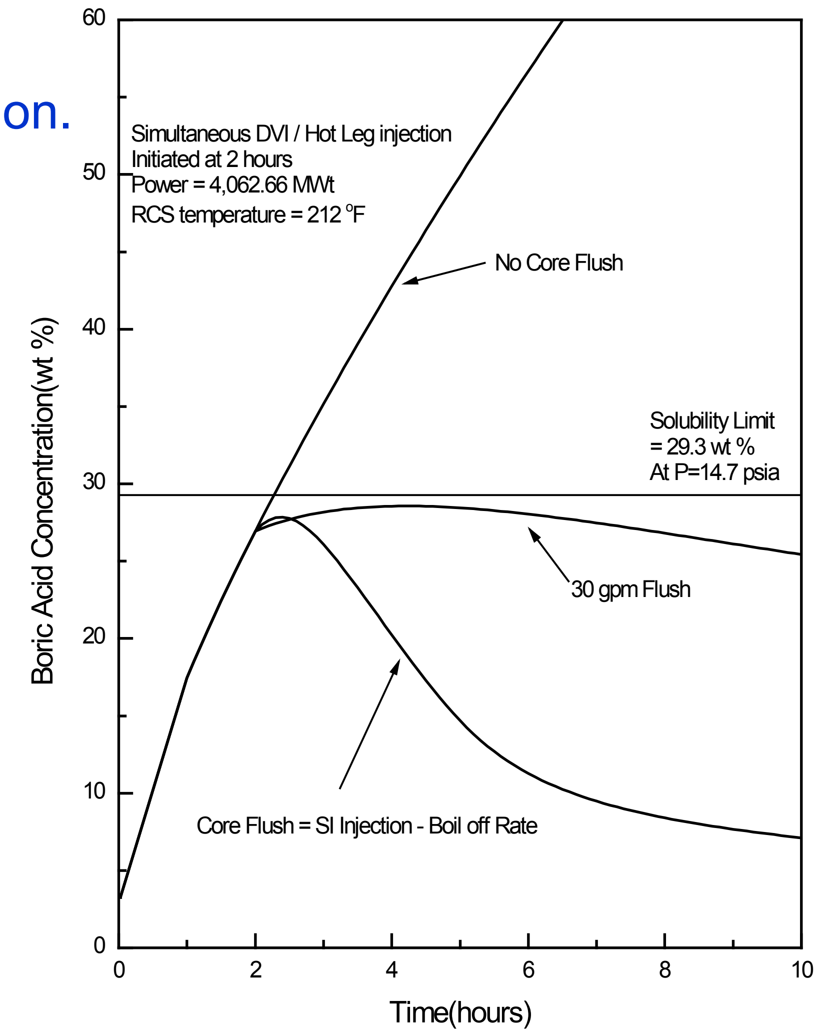
Bottom of Hot-leg

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.3 Core and System Performance

❑ Post LOCA LTC Results

- Three results about boron precipitation.
- No core flush
  - ✓ With no core flushing flow, boric acid does begin to precipitate after 2.3 hours.
- Core flush
  - ✓ When the operator initiates simultaneous injection by 2 hours, there is no boric acid precipitation
- 30 gpm flush
  - ✓ The margin provided for the prevention of boric acid precipitation by the core flushing flow of 30 gpm

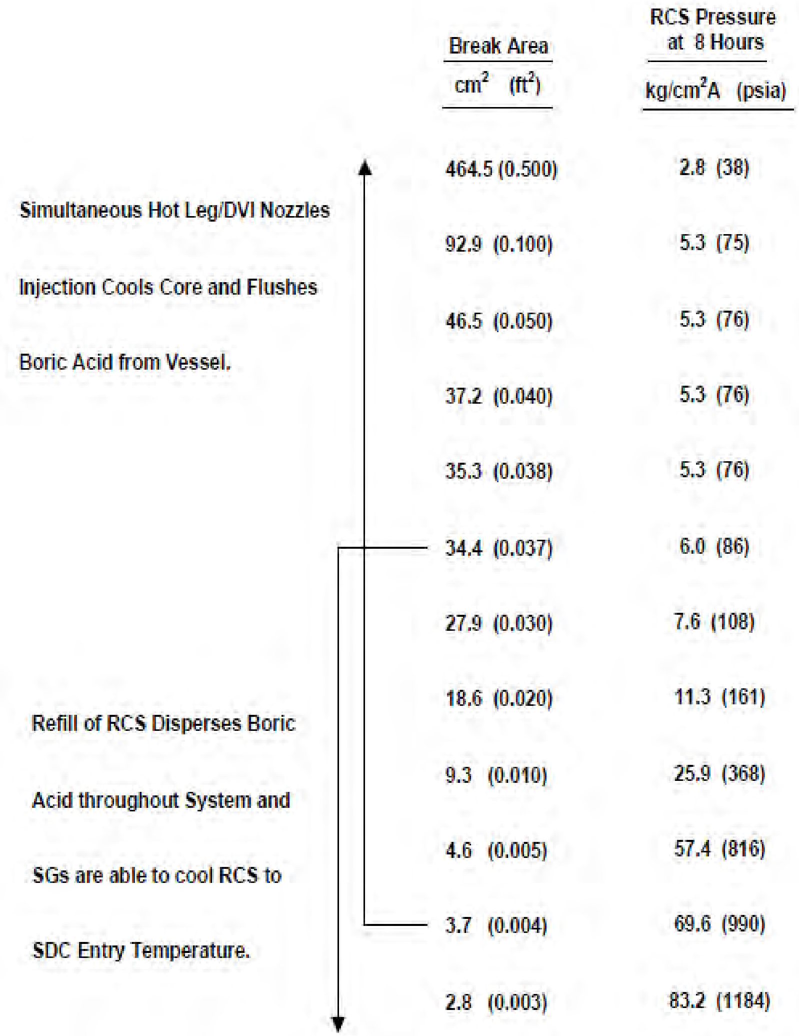


15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

15.6.5.3 Core and System Performance

❑ Post LOCA LTC Results

- The overlap in break areas for which either the large break or small break procedures can be used is illustrated in Figure
- Therefore, the plant can be secured for all break size



NON-PROPRIETARY

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

Post LOCA LTC: Evaluation of In-vessel Downstream Effects

❑ Debris Generation

- According to the guidance of NEI 04-07, RCS hot-leg line (diameter of 42 in) break is selected, and this break location bounds variations in debris generation by size, quantity, and type of debris from other break locations
- Generated debris : RMI, coatings (epoxy, IOZ), latent debris (fiber, particle), concrete, aluminum
- For conservatism, APR1400 assumes that all generated coatings and all latent debris are transported to the sump in the IRWST

❑ Strainer Bypass Testing (Scale-down Test)

- Total fibrous debris at the strainer established 6.8 kg(15 lbm) of latent fiber
- Testing is performed with only fibrous debris since adding particulates may reduce the amount of bypass debris due to clogging at the strainer
- Filter bag is used to collect debris bypassed through the strainer
- Bypassed fibrous debris mass: 1.67 kg (3.68 lbm) (through 4 sump strainers)
- Fibrous debris mass per fuel assembly is calculated to 6.93 g

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

Post LOCA LTC: Evaluation of In-vessel Downstream Effects

- Three LOCA scenarios were chosen
  - Core flow rate and its direction affect the behavior of debris in the core
  - Break location affects the driving force of injected ECCS water

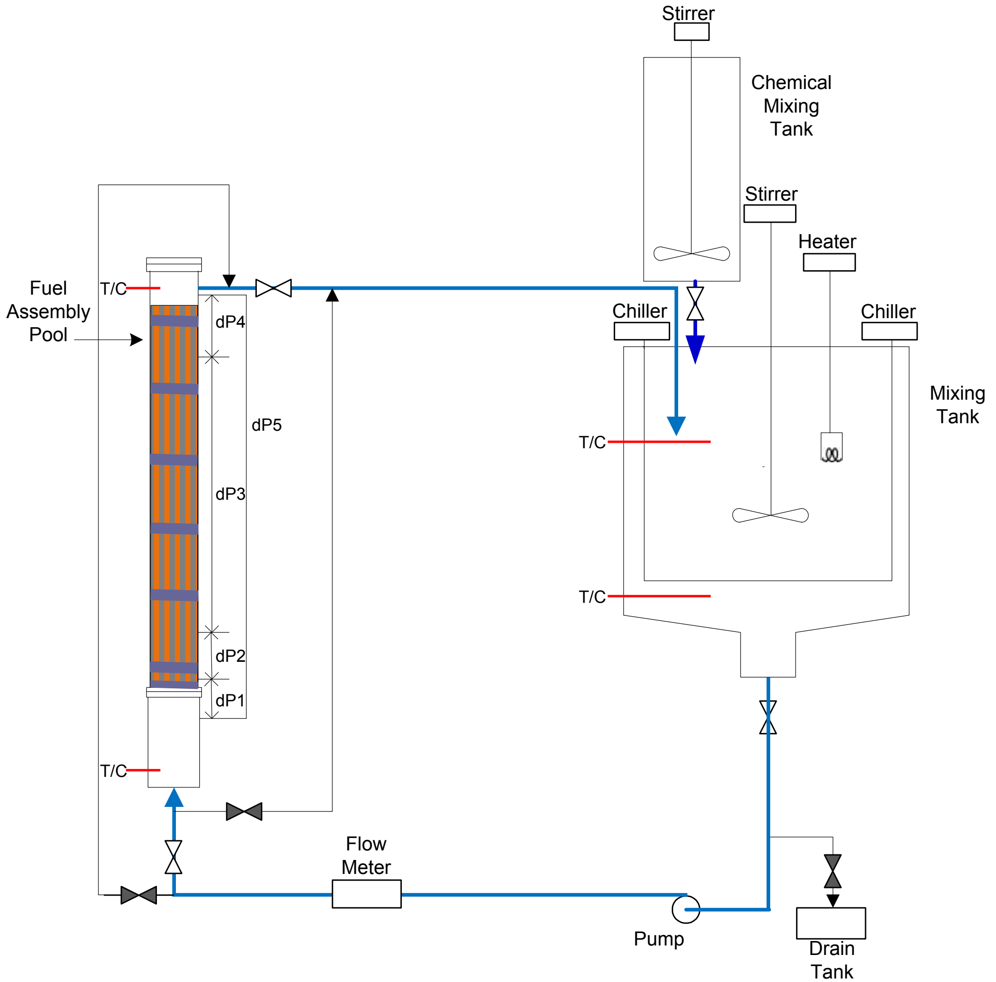
LOCA scenario	Core flow direction	APR1400 flow rate	Flow rate/ FA*	Remark
HL Break	Upward	4,940 gpm	20.5 gpm	Max. safeguard flow rate of four SIs
CL Break	Upward	880.2 gpm	3.65 gpm	Boil-off flow rate at 700 sec
CL Break after HLSO	Downward	2,470 gpm	10.25 gpm	Max. safeguard flow rate of two SIs

* 1/241 of the maximum flow rate for the scaled tests

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

Post LOCA LTC: Evaluation of In-vessel Downstream Effects

- Test loop



15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

NON-PROPRIETARY

Post LOCA LTC: Evaluation of In-vessel Downstream Effects

- **Test column**
  - Mock-up FA of PLUS7 : ½ full length
  - Same components : top/bottom nozzle, p-grid, top/bottom grid
    - ✓4 mid grids (Full length of PLUS7 has 9 grids)
- **Debris mixing tank**
  - Transparent cylindrical shape : 1,880 L (500 gal, 45.6 % of the minimum IRWST water/FA)
  - Heater/chiller are installed to control water temperature
  - A stirrer is installed to prevent debris settling
- **Recirculation System**
  - One recirculation pump, one flow meter
  - Flow rate can be adjustable
- **Control and Monitoring System**
  - Control : water flow rate, water temperature
  - Record : flow rate(1), temperature(4), differential pressure(5)

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

NON-PROPRIETARY

Post LOCA LTC: Evaluation of In-vessel Downstream Effects

- **Input for In-vessel Fuel Assembly Test**

Debris type	Specific type	Debris generated in containment	Assumed bypass debris	Per FA* (g)
Fibrous	Latent fiber	15 lbs (6.8 kg)	3.68 lbs** (1.67 kg)	**6.93*****
Particulate	Coating debris	3.1 ft^3 (280.5 kg)	3.1 ft^3 (280.5 kg)	1,164
	Latent particle	185 lbs (83.9 kg)	185 lbs (83.9 kg)	348
Chemical compounds		408.0 lbs (185.1 kg)	408.0 lbs (185.1 kg)	768 (70 liters)

* 1/241 of the assumed bypass debris amount

** Result from the APR1400 strainer bypass testing

*** 15 g is applied for actual test

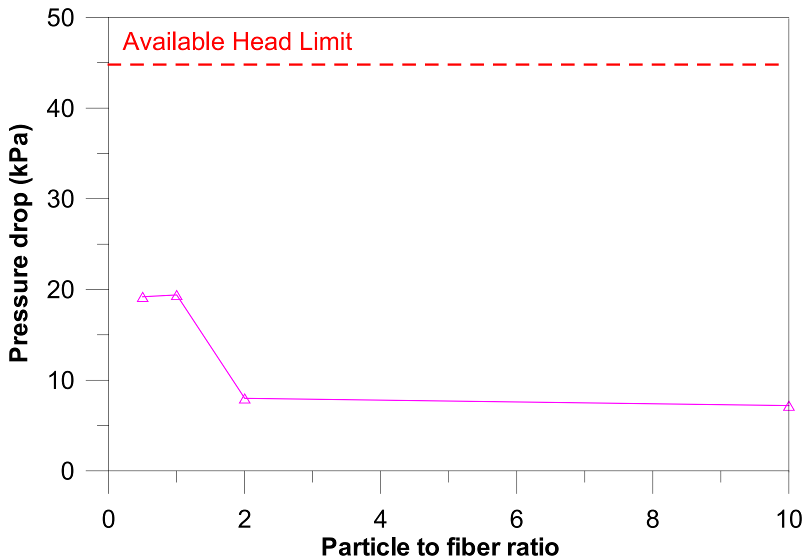
NON-PROPRIETARY

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

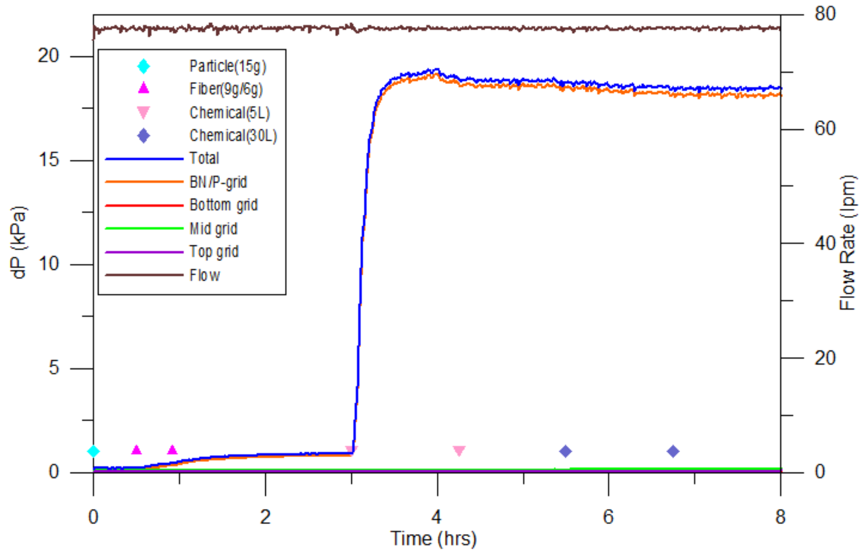
Post LOCA LTC: Evaluation of In-vessel Downstream Effects

❑ Hot-leg break

- 5 Tests had been run to evaluate hot-leg break conditions
- p:f ratio ranged from 0.5 to 10
- Limiting result occurred at p:f ratio = 1
- Meet the acceptance criteria(42.7 kPa) with sufficient margin(54.6%)



Pressure drops vs. particle to fiber ratio



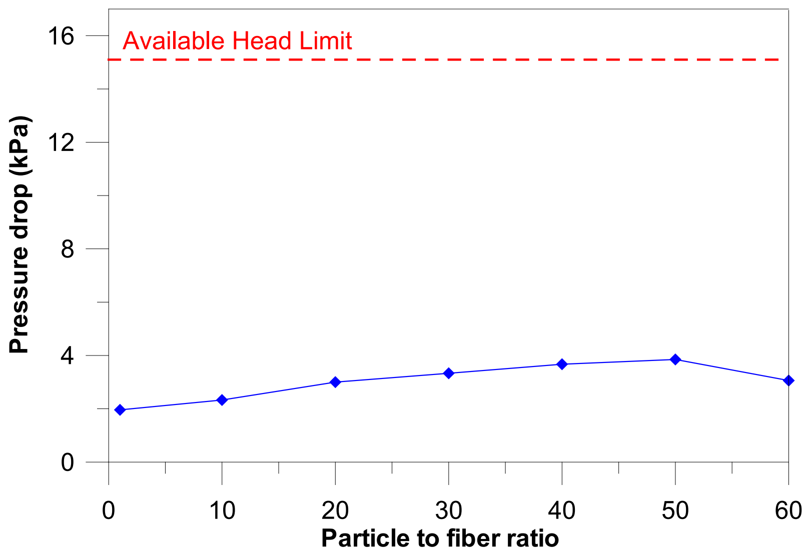
dP result (APR1400-21 ; P:F=1:1, 77.6 lpm)

15.6.5 LOCA Resulting from Spectrum of Postulated Piping Breaks (Cont'd)

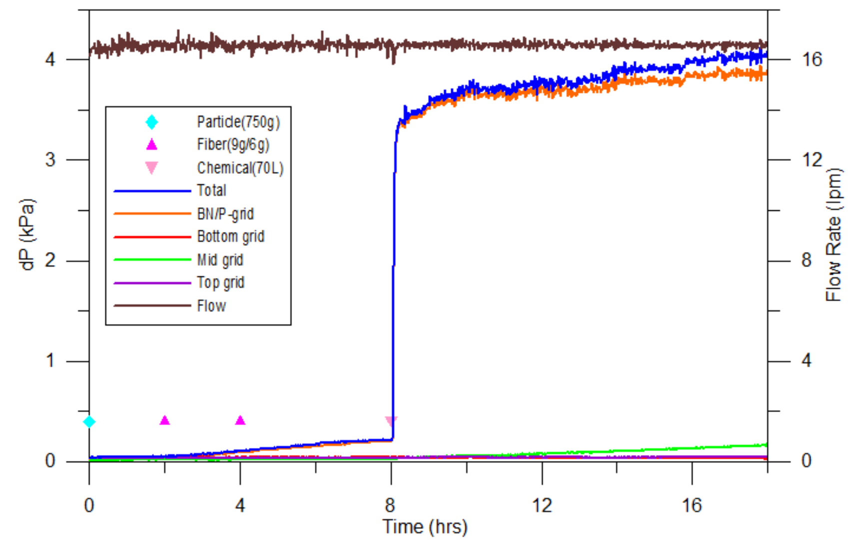
Post LOCA LTC: Evaluation of In-vessel Downstream Effects

❑ Cold-leg break

- 7 Tests had been run to evaluate cold-leg break conditions
- p:f ratio ranged from 1 to 60
- Limiting result occurred at p:f ratio = 50
- Meet the acceptance criteria(13.2 kPa) with sufficient margin(70.8%)



Pressure drops vs. particle to fiber ratio



dP result (APR1400-95 ; P:F=1:50, 16.6 lpm)

15.7 Radioactive Material Release from a Subsystem or Component

NON-PROPRIETARY

❑ Radioactive Gas Waste System Leak or Failure

- The analysis method and radiological consequences of the GWMS leak or failure event are described in Subsection 11.3.3

❑ Radioactive Liquid Waste System Leak or Failure

- US NRC SRP Rev. 3, the section corresponding to a LWMS leak or failure event has been deleted

❑ Postulated Radioactive Releases Due to Liquid-Containing Tank Failures

- According to BTP 11-6, this analysis has been added to Section 11.2.3

❑ Fuel Handling Accident

- FHA in the Containment Building
- FHA Outside Containment

15.8 Anticipated Transient without Scram (ATWS)

❑ The failure of the reactor trip function

- AOO followed by the failure of the reactor trip by RPS
- Required to reduce risk of ATWS for PWR (10 CFR 50.62)

❑ Installed diverse protection system(DPS)

- Equipped with diverse protection system
- Reduction of risk from ATWS events
- DPS includes reactor trip function and auxiliary feedwater actuation signal (AFAS)

15A. Radiological Consequence Analysis

15A.1 Design Targets and Design Features

15A.2 Analysis Methods

15A.3 Design Evaluation for LOCA

15A.4 Design Evaluation for Non-LOCA

15A.5 Radiological Consequences for DBAs

15A.1 Design Targets and Features

❑ Design targets

- To demonstrate that the doses due to DBAs are within the limits
  - ✓ Public at EAB and LPZ : 10 CFR 52.47 (25 rem), SRP 15.0.3 (2.5 – 25 rem)
  - ✓ Worker in MCR : GDC 19 (5 rem)

❑ APR1400 design features to minimize accident releases

- SIS to prevent fuel damage
- AFWS for SG cooling
- CSS with 5,000 gpm capacity for fission product removal
- TSP to prevent iodine re-evolution from IRWST
- CREVAS and FHEVAS for emergency filtration by RMS
- CIAS and CPIAS for early isolation of containment bypass
- Steel-lined containment for limitation of leakage
- Automatic selective dual MCR air intakes for less contaminated air supply
- Positive pressure in MCR for minimization of unfiltered in-leakage

15A.2 Analysis Methods

- LOCA
  - Assumes core meltdown for conservatism
  - Detailed assumptions and parameters for LOCA are presented in 15A.3
- Non-LOCA
  - Uses fuel damage & mass release data based on T/H analysis
  - Detailed assumptions and parameters for Non-LOCA are presented in 15A.4
- Based on AST and TEDE dose criteria
- Uses RADTRAD 3.03 codes (NUREG/CR-6604)
- Conservative X/Q values listed in DCD Ch. 2 were used
- Approaches are consistent with RG 1.183 guidance

15A.3 Design Evaluation for LOCA

- **Source term**
  - Core fission product inventory
    - ✓ Core power : 4,063 MWt (102% of rated power)
    - ✓ Burnup : 56.4 GWD/MTU
  - Release timing & magnitude
    - ✓ Based on RG 1.183
- **Containment building**
  - Sprayed/unsprayed regions : 75%/25%
  - Air mixing
    - ✓ 2 volumes of unsprayed region per hour (SRP 6.5.2)
  - Leak rate
    - ✓ 0.1%/day (< 24 hrs) / 0.05%/day (> 24 hrs)
  - Airborne Fission Product Removal Coefficient
    - ✓ Elemental iodine removal by containment spray : model in SRP 6.5.2
    - ✓ Particulate iodine removal by containment spray : model in SRP 6.5.2
    - ✓ Particulate (aerosol) removal by natural deposition : 10 percentile value of the Powers model (NRC NUREG/CR-6189) built into RADTRAD 3.03

15A.3 Design Evaluation for LOCA

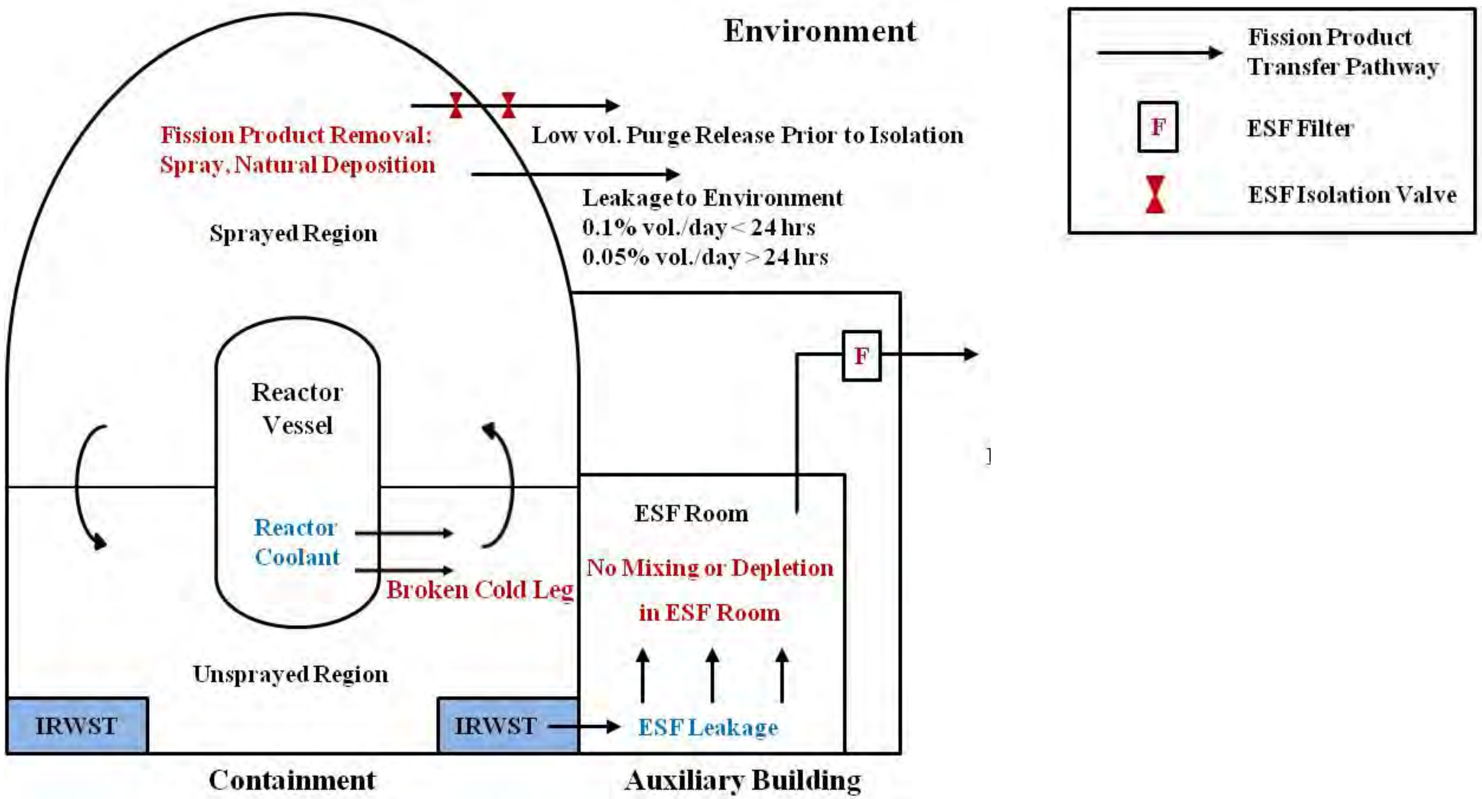
❑ Other parameters

- Containment purge isolation delay time : 5 seconds
- ESF leakage rate : Two times of the design leakage
- ACU filtering for ESF leakage : 95% for aerosols and iodines

❑ Long term pH in IRWST after LOCA

- Calculation methodology : based on NUREG/CR-5950 (1992)
- Calculation code : SOLGASMIX-PV
- Materials considered for pH
  - ✓ Boron Oxides (acidic)
  - ✓ Tri-sodium phosphate (basic)
  - ✓ Hydriodic acid (HI) (acidic)
  - ✓ Nitric acid (acidic)
  - ✓ Hydrochloric acid (acidic)
- Radiation condition
  - ✓ The maximum values of the time dependent total integrate doses (TIDs) in the Containment Building during LOCA condition

15A.3 Design Evaluation for LOCA



Radioactivity Transport Model for Loss of Coolant Accident

15A.4 Design Evaluation for Non-LOCA

Source term

- Primary coolant
  - ✓ Noble gases (DE Xe-133) : 580 μCi/g (TS LCO)
  - ✓ Iodine (DE I-131) : 1.0 μCi/g (TS LCO)
  - ✓ Alkali metal : 1.0% fuel defect (RG 1.183)
- Iodine spike concentration
  - ✓ Pre-Accident Iodine Spike (PIS)
  - ✓ Event-Generated Iodine Spike (GIS)
- Fuel pellet clad gap inventory : RG 1.183
- Radial power peaking factor : 1.8
- Secondary coolant
  - ✓ Iodine (DE I-131) : 0.1 μCi/g (TS LCO)

15A.4 Design Evaluation for Non-LOCA

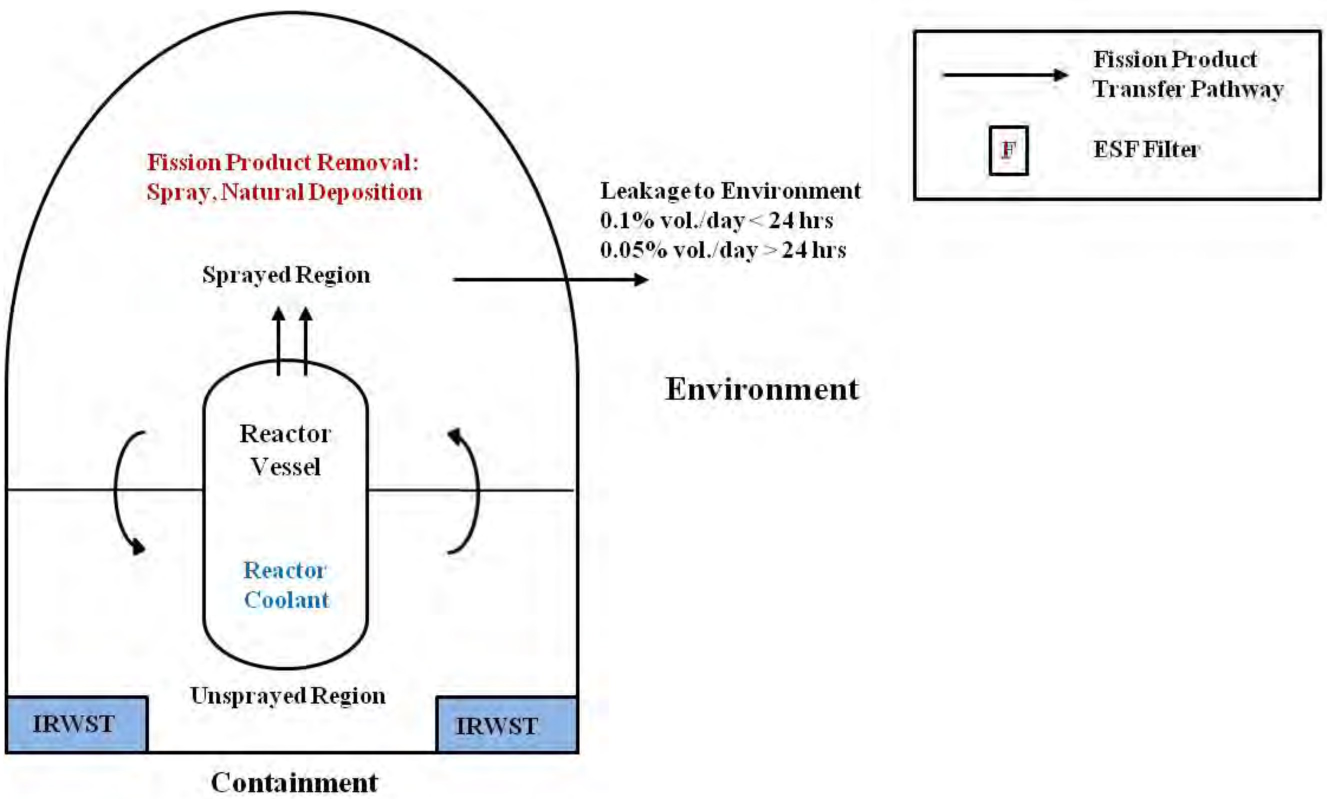
Assumptions and parameters for non-LOCA

- SG leak rate TS LCO for 0.3 gpm (per SG)
- Fuel cladding damage (determined using T/H analyses)
  - ✓ MSLB : 1% of core
  - ✓ CEA ejection : 10% of core
  - ✓ RCP seizure : 7% of core
  - ✓ FWLB, LDLB, SGTR : No failure
  - ✓ FHA : 100% of 1 fuel assembly
- DF of iodine by SG water level
  - ✓ Dryout : 1
  - ✓ Partial uncovery of the SG tube: calculated using flashing fraction based on RG 1.183
  - ✓ Submergence : 100

15A.4 Design Evaluation for Non-LOCA

❑ Release Transport

- Fuel → RCS → Containment → Env

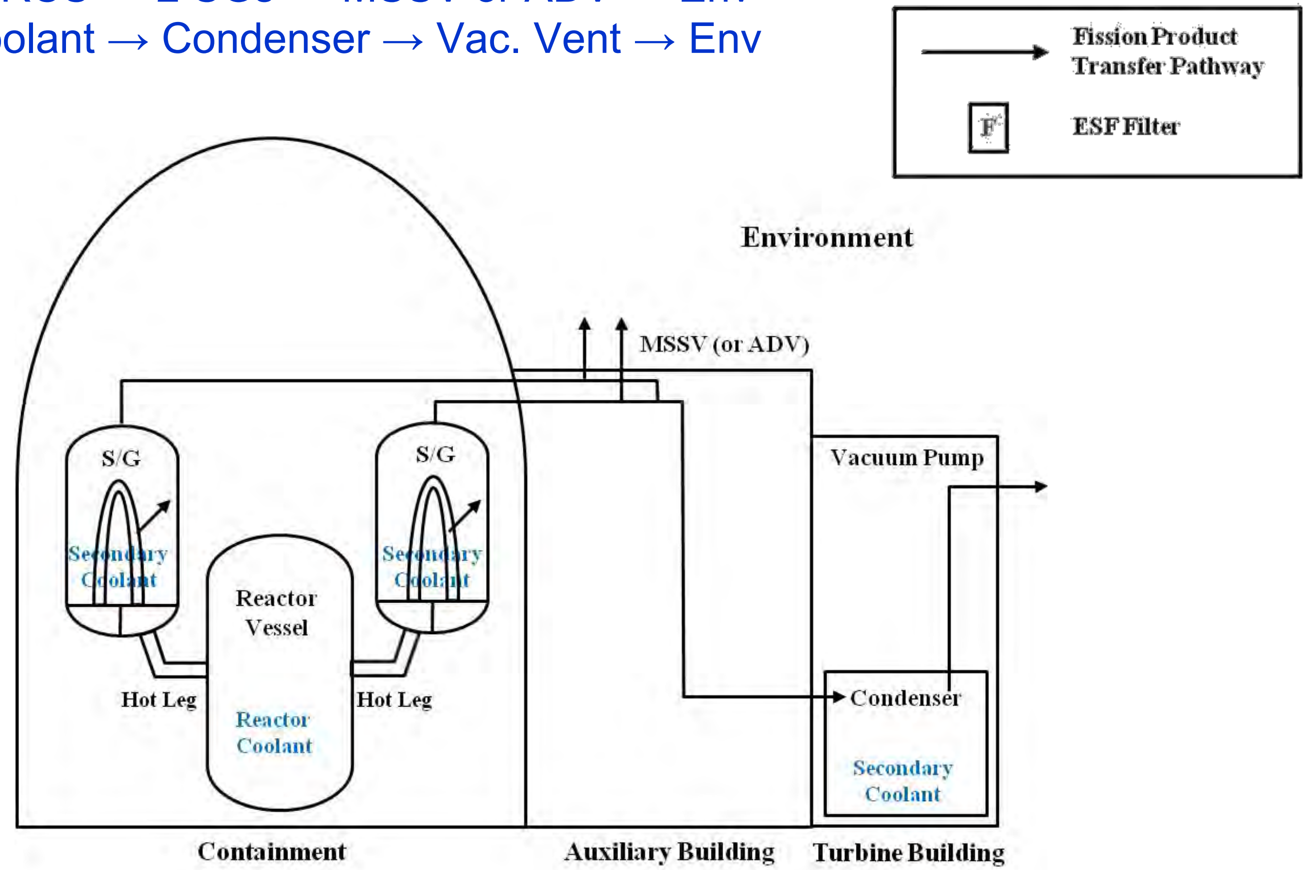


Radioactivity Transport Model for CEA Ejection (Containment Release)

15A.4 Design Evaluation for Non-LOCA

Release Transport

- Fuel → RCS → 2 SGs → MSSV or ADV → Env
- Sec. Coolant → Condenser → Vac. Vent → Env



Radioactivity Transport Model for CEA Ejection (Release through the Secondary System)

15A.5 Radiological Consequences for DBAs

❑ Results

- Doses to the public at EAB/LPZ for all DBAs are well within dose limits of 10 CFR 52.47 (2.5 – 25 rem)
- MCR habitability is ensured for all DBAs by complying the criteria in GDC 19 (5 rem)

15A.5 Radiological Consequences for DBAs

APR1400 Radiological Consequences for DBAs

Design Basis Accident		Results for APR1400 (rem)			Dose Limit (rem)	
		EAB	LPZ	MCR	EAB/LPZ	MCR
Steam system piping failure	1 % Fuel Failure	4.9	3.8	3.6	25.0	5.0
	Pre-accident spike	3.5	1.5	2.1	25.0	
	Event-generated spike	1.0	0.5	2.2	2.5	
Feedwater system pipe break		0.5	0.2	1.98	2.5	
RCP rotor seizure		1.6	0.8	1.5	2.5	
Control element assembly ejection	Containment leakage	5.9	5.6	3.5	6.3	
	Steam system release case	4.0	2.2	2.9	6.3	
Failure of small lines carrying primary coolant outside containment		0.4	0.1	1.95	2.5	
Steam generator tube rupture	Pre-accident spike	0.8	0.2	2.0	25.0	
	Event-generated spike	0.5	0.1	1.96	2.5	
Loss of coolant accident		20	10.6	4.7	25.0	
Fuel handing accident		3.9	0.9	0.9	6.3	

Summary

- APR1400 Transient and Accident Analyses of Chapter 15 demonstrate to comply with requirements of federal regulations and NRC regulatory documents
- There are 12 Open Items in total for Chapter 15 as described in next slides

Open Items

OI	RAI No	Question No	Description	Response Submitted	Status
15.0.3-1	108-7973	15.00.03-1	• The periodic reopening of the control room HVAC outside air intakes in Accident condtion • Related to the RAI 368-8470, Q 14.03.08-14	5/8/2017	Under discuss with NRC staff to resolve
15.4.6-1	17-7917	15.04.06-1	Justification that the complete mixing model yields conservative times to criticality for Modes 4 and 5 without an RCP in service	08/26/2015	Under discuss with NRC staff to lock close the unborated water source isolation valve
15.4.6-2	511-8668	15.04.06-8	Justification that which reactor trip would prevent violating the minimum DNBR under condition of an at power boron dilution	08/19/2016	Response submitted (Under review by staff)
4.2-1	N/A	N/A	Application of suitable penalty to address the impa ct of TCD on postulated accidents	-	DCD 15.4.8 will be revised
4.2-1			The staff expects TCD to result in a higher peak radi al average fuel enthalpy and fuel centerline tempera ture.	-	Under the re-reanalysis applying revised MTC and TCD
15.6.5-1	399-8510	15.06.05-7	The staff evaluation of issues associated with the co des or CAREM methodology will be documented in t he topical report SER.	-	Under the re-reanalysis applying revised MTC and TCD
15.6.5-2	399-8510	15.06.05-7	The error in the moderator reactivity curve input in t he RELAP5 code All of the LBLOCA cases presented in the DCD are i n the process of being re-run.	-	Under the re-reanalysis applying revised MTC and TCD

Open Items

OI	RAI No	Question No	Description	Response Submitted	Status
15.6.5-3	318-8337	15.06.05-2	The entire SBLOCA section of DCD Tier 2 Chapter 15.6.5 and the technical report (APR1400-F-A-NR-14001-P) will be revised.	02/02/2017	DCD and TeR revisions were submitted
15.6.5-4	404-8488	15.06.05-10	The staff is tracking this item until the applicant submits a revision to its response.	-	Under the revision of response
15.6.5-5	398-8457	15.06.05-5	The final analysis and the relevant DCD changes have already been submitted to staff and are currently under review.	01/16/2017	Under NRC review
15.6.5-6	430-8455	15.06.05-22	The revised analysis is not completed.	01/17/2017	RAI response and TeR revision were submitted
15.6.5-8	143-8092	15.06.05-1	The applicant is expected to submit a revised response.	01/17/2017	RAI response and TeR revision were submitted

Acronyms

ACU	air cleaning unit
ADV	atmospheric dump valve
AFW	auxiliary feedwater
AOO	anticipated operational occurrence
APR1400	advanced power reactor 1400
AST	alternative source term
ATWS	anticipated transient without scram
BOC	beginning of cycle
BWR	boiling water reactor
CEA	control element assembly
CEAE	control element assembly ejection
CEDM	control element drive mechanism
CEDMCS	control element drive mechanism control system
CHF	critical heat flux
CIAS	containment isolation actuation signal
COLSS	core operating limit supervisory system

CPCS	core protection calculator system
CPIAS	control element assembly
CREVAS	control room emergency ventilation actuation system
CSS	containment spray system
CVCS	chemical and volume control system
DBA	design basis accident
DBE	design basis event
DE	dose equivalent
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DPS	diverse protection system
DVI	direct vessel injection
EAB	exclusion area boundary
ECCS	emergency core cooling system
EOC	end of cycle
ESF	engineered safety features

Acronyms

ESFAS	engineered safety features actuation system
ESF-CCS	engineered safety features component control system
FHEVAS	fuel handling area emergency ventilation actuation signal
FLB	feedwater line break
FTC	fuel temperature coefficient
FWCS	feedwater control system
GDC	general design criteria
GIS	generated iodine spike
HFP	hot full power
HPPT	high pressurizer pressure trip
HZP	hot zero power
IOSGADV	inadvertent opening of a steam generator atmospheric dump valve
IRWST	in-containment refueling water storage tank
ITC	isothermal temperature coefficient

LCO	limiting conditions for operation
LDLB	letdown line break
LBLOCA	large break loss of coolant accident
SBLOCA	small break loss of coolant accident
LOCV	loss of condenser vacuum
LOOP	loss of offsite power
LPZ	low population zone
MCR	main control room
MSIS	main steam isolation signal
MSIV	main steam isolation valve
MSSV	main steam safety valve
MTC	moderator temperature coefficient
PA	postulated accident
PIS	pre-accident iodine spike
PLCS	pressurizer level control system

Acronyms

POL	power operating limit
POSRV	pilot operated safety relief valve
PPS	plant protection system
PWR	pressurized water reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RMS	radiation monitoring system
RRS	reactor regulating system
RTO	reactor trip override
SAFDL	specified acceptable fuel design limit
SGTR	steam generator tube rupture
SIP	safety injection pump
SIS	safety injection system
SLB	steam line break
TEDE	total effective dose equivalent
TSP	tri-sodium phosphate

VOPT	variable overpower trip

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Presentation to the ACRS Subcommittee

Korea Hydro Nuclear Power Co., Ltd (KHNP)
APR1400 Design Certification Application Review

Safety Evaluation with Open Items: Chapter 15

Transient and Accident Analyses

May 19, 2017

Technical Staff Reviewers

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David Caraher	Carl Beyer	Mahmoud Massoud
Douglas Barber	Glenn Roth	Jim Servatious

Chapter Sections and Presenters

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Section	Title	Presenter
15.0/15.0.2	Review of Transient and Accident Analysis Methods	Shanlai Lu
15.0.3	Design Basis Accident Radiological Consequence Analyses	Michelle Hart
15.1	Increase in Heat Removal by the Secondary System	Tim Drzewiecki
15.2	Decrease in Heat Removal by the Secondary System	Raul Hernandez, Carl Thurston
15.3	Decrease in Reactor Coolant System Flow Rate	Chris Van Wert, Peter Yarsky
15.4	Reactivity and Power Distribution Anomalies	Shanlai Lu, Tim Drzewiecki, Carl Thurston, Andy Bielen
15.5	Increase in Reactor Coolant Inventory	Tim Drzewiecki
15.6	Decrease in Reactor Coolant Inventory	Shanlai Lu, Tim Drzewiecki, James Gilmer, Syed Haider, Dan Prelewicz, David Caraher, Joe Staudenmeier
15.7	Radioactive Material Release from a Subsystem or Component	N/A
15.8	Anticipated Transients Without Scram	James Gilmer

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Staff Perspectives and Review Approaches

APR1400 - Similar to the Previously Approved Reactor Design

- APR1400 is a design evolved from CE System 80+, which was certified in 1994
- CE System 80 plants, e.g., Palo Verde, remain operating safely
- Many system designs are either similar or identical to those of CE System 80+
- DCD application is also similar to that of CE System 80+
- APR1400 has been designed with more safety margins

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Staff Review Perspectives and Approaches

Regulatory Review Approach

- Focus on the changes implemented into APR1400 design
- In-depth review on those safety issues identified after 1995
- Provide overall coverage with the assistance of staff confirmatory analyses on selected areas
- Identify potential issues early on and keep close communication with KHNP on issue resolution path
- Use audits and on-site inspections to clarify the issues

15.0: Transient and Accident Analyses

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- Classification of Events
- Plant Characteristics and Initial Condition Assumed in the Accident Analysis
- Reactor Trip System and Engineered Safety Feature Systems Analytical Limit and Delay Times
- Component Failures, Nonsafety-Related Systems, Operator Actions Considered in the Safety Analysis
- Loss of Offsite Power, Long-Term Cooling, Methodology for Determining Uncertainties and Thermal Conductivity Degradation

Staff Findings

Staff finds that the documented information in Section 15.0 satisfies the relevant regulatory requirements except two Open Items.

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Open Item: Fuel Pellet Thermal Conductivity Degradation

- Staff SER Section 4.2 “Fuel Design” identified this issue as an Open Item for fuel performance evaluation due to FATES3B code deficiency
- As the result, the initial steady state fuel center line temperature and the total core sensible heat/stored energy have been underestimated
- Chapter 15 analyses relying on the initial core conditions are affected

Open Item: Boron Dilution During LOCA Long Term Cooling Phase

- GSI-185 needs to be addressed according to Reg. 1.206
- No analysis was performed specifically for APR1400
- The phenomenon is possible due to the presence of a deep loop seal

15.0.2: Review of Transient and Accident Analysis Methods

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Computer Codes

CESEC-III, TORC, CETOP, COAST, HRISE, STRIKIN-II, HERMITE, ROCS/DIT, CEFLASH-4AS, COMPERC-II, PARCH, RELAP5/MOD3.3K, CONTEMPT4/MOD5, CELDA, BORON, CEPAC, NATFLOW

Methods

Non-LOCA Safety Analysis Methodology
LBLOCA Method (Pending on LBLOCA Topical Report Review)
SBLOCA Evaluation Methodology
Post-LOCA Long Term Cooling Evaluation Methodology

Applicability

System response ranges. Original approval and limitations
Code maintenance program and QA records. Operating system

15.0.2: Review of Transient and Accident Analysis Method

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Audits and RAIs

- Staff conducted two audits to examine the calculation reports and QA records.
- RAIs were issued and resolved about the following:

  HERMITE radial leakage term and limiting pressure

  ROCS code benchmark

  COAST code friction and form loss coefficients

  CESEC-III cold edge enthalpy definition

  STRIKIN-II fuel/cladding temperature during SLB events
- Staff issued RAIs about the application of RELAP5/MOD3.3K to LBLOCA analyses. The conclusion is pending on the review of LBLOCA topical report

Results

Except for one Open Item on the pending LBLOCA topical report review, all transient and accident analysis methods are considered acceptable for their applications to APR1400

15.0.3: Radiological Consequences of Design Basis Accidents

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- DBA source terms, transport, and release
- Core isotopic inventory, coolant activity concentrations
- Post-accident containment water chemistry management
- Evaluation of fission product removal
- Offsite doses, control room and TSC radiological habitability

15.0.3: Evaluation

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- Applicable guidance is given in SRP 15.0.3 and RG 1.183
- Staff performed independent confirmatory analyses for all DBAs and the core isotopic inventory
- Confirmatory items to ensure that changes proposed in RAI responses are incorporated into future revision of DCD
- Applicant's offsite dose results are within regulatory dose acceptance criteria. The DBA dose analyses are consistent with guidance in RG 1.183 and are therefore acceptable with respect to offsite consequences
- Applicant's control room and TSC dose results are less than 5 rem TEDE, however, Open Item 15.0.3-1 remains unresolved. Therefore, the staff is unable to make a finding with respect to control room and TSC radiological habitability

Open Item 15.0.3-1: Modeling of Control Room Emergency Makeup Air Cleaning System Operation

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- Control logic automatically re-opens closed CR air intake isolation dampers at a preset interval for a short time to determine which intake has lower radioactivity concentration
    - COL applicant is to choose both the interval time between damper re-openings and the length of time the intakes are both open
- This mode of operation was not described in DCD Sections 6.4 and 9.4.1, and the DBA dose analyses did not explicitly model the periodic intake of outside air through both air intakes
- Discovered in response to an RAI on ITAAC for the control room intake radiation monitors
- RAI Question 14.03.08-14 sub-question 6.b remains under review. Applicant is to provide final revision of the RAI response. Open item 15.0.3-1 is tracking the issue until resolution

15.1.1 - 15.1.4: AOOs that Increase Heat Removal by Secondary System

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- 15.1.1 Decrease in Feedwater Temperature
- 15.1.2 Increase in Feedwater Flow
- 15.1.3 Increase in Steam Flow
- **15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve (IOSGADV)**

- Evaluation Model
  - CESEC-III for nuclear steam supply system (NSSS) modeling
  - CETOP with KCE-1 CHF correlation for departure from nuclear boiling ratio (DNBR)

- Staff confirmatory calculations
  - Hand Calculations and TRACE
  - Verify input assumptions and identification of bounding event

15.1.4: IOSGADV

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- **Input Parameters and Initial Conditions**
  - Table 15.1.4-1 of SER provides basis for input parameters
  - Applicant assumed a bounding relief capacity
  - Loss of offsite power (LOOP) occurs coincident with turbine trip
  - Analysis performed with and without reactor trip override (cut back feedwater post reactor trip)
  - Operator action taken at 30 minutes to trip the reactor and initiate cooldown

- **Results and Findings**
  - Fuel integrity is maintained (minimum DNBR of 1.336 vs limit of 1.29)
    - GDC 10, 20, and 26
  - Peak SG pressure and RCS pressure remain below 110 percent (Upset Conditions)
    - GDC 15

15.1.5: Steam Line Break

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- **Two sets of cases of Steam Line Break (SLB)**
  - Maximize return-to-rower (RTP)
  - Maximize pre-trip fuel degradation

- **Evaluation Model**
  - CESEC-III for NSSS modeling (cold edge temperature for feedback)
    - Cold edge temperature implementation verified during inspection
  - CETOP with KCE-1 CHF correlation for DNBR

- **Input Parameters and Initial Conditions**
  - Table 15.1.5-2 and Table 15.1.5-3 of SER provides basis for input parameters
  - Double-ended break upstream of main steam isolation valves
  - Events evaluated with and without LOOP

15.1.5: Steam Line Break – Cont.

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- **Input Parameters and Initial Conditions**
  - Reactor trip on low reactor coolant pump (RCP) speed for cases with LOOP and variable overpower trip (VOPT) for cases without LOOP
  - Single failure of an emergency diesel generator (consequent loss of two safety injection pumps)
  - Operator action taken at 30 minutes to initiate a plant cooldown in accordance with emergency operating procedures (EOPs)

- **Results/Findings**
  - No post-trip RTP (max post-trip reactivity -0.187 % $\Delta\rho$, shutdown by 187 pcm)
  - Pressure-temperature limits are not exceeded during cooldown associated with SLB
  - DNBR analysis shows no fuel failures
  - Satisfies GDC 13, 17, 27, 28, 31, and 35

15.2.1 - 15.2.7: Decrease in Heat Removal by the Secondary Systems

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Events Evaluated

- Loss Of External Load; Turbine Trip; Loss Of Condenser Vacuum; Closure Of Main Steam Isolation Valve (BWR); Steam Pressure Regulator Failure (Closed); Loss of Non-emergency AC to the Station Auxiliaries; Loss of Normal Feedwater Flow

Area of Review

- Sequence of events, identification most limiting event, confirm maximum pressure in primary and secondary sides below 110% design value
- Fuel cladding integrity is maintained

Staff Findings

- Loss of Condenser Vacuum is the most limiting of the events
- Primary and secondary pressures are maintained below design maximum
- Fuel cladding integrity is not compromised

15.2.8: Feedwater System Pipe Break

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Staff Findings

- Staff agrees that CESEC-III when combined with conservative assumptions for break flow, affected steam generator heat transfer, intact steam generator level and reactor trip timing yields conservative analysis results
- Minimum DNBR: Remains above the 1.29 95/95 DNBR limit, preserving the SAFDLs
- Maximum RCS and main steam pressure: remains below 120% of the design pressures including the effects of a LOOP

15.3.1: Loss of Forced Reactor Flow

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Evaluation Model

- CESEC-III for nuclear steam supply system (NSSS) modeling
- HERMITE for neutron diffusion
- CETOP with KCE-1 CHF correlation for departure from nuclear boiling ratio (DNBR)

Staff Confirmatory Calculations

- RES performed TRACE/PARCS confirmatory calculations

15.3.1: Loss of Forced Reactor Flow

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Regulatory Findings: Loss of Forced Reactor Flow Requirements

- The staff finds that the analysis is based on approved codes and methods and is therefore acceptable
- The staff finds that the input assumptions appropriately bound the potential conditions of the plant
- The analysis demonstrates that the SAFDLs are not violated and therefore GDC 10 is met
- The analysis demonstrates that the reactor coolant system and its auxiliaries are not breached during normal operations or AOOs, thereby demonstrating compliance with GDC 15
- The loss of offsite power leads to automatic startup of EDGs, thereby demonstrating compliance with GDC 17
- The reactivity changes are reliably controlled so that the SAFDLs are not exceeded, thereby demonstrating compliance with GDC 26

Non-proprietary

APR1400 Loss of Flow Event Confirmatory Analysis

Dr. Peter Yarsky and Dr. Andrew Bielen

Office of Nuclear Regulatory Research

Sequence of Events

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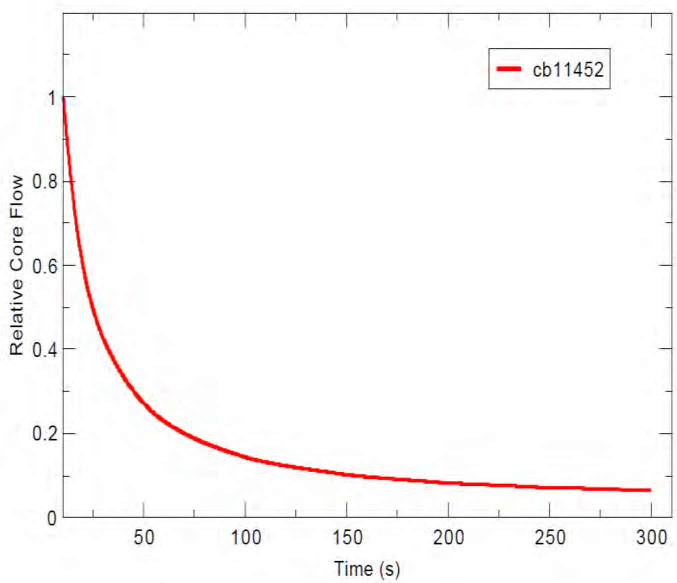
- Event is initiated by a LOOP
- The LOOP causes several trips: reactor coolant pumps (RCPs), turbine, and reactor
- mDNBR decreases as coolant flow decreases during coastdown of the RCPs
- mDNBR margin is restored as reactor power decreases following trip and flow achieves a steady, natural circulation through the reactor coolant system (RCS)
- RCS temperature and pressure initially increase as the loss of flow combined with loss of heat sink result in a reduction of heat transfer from the primary to secondary system
- Auxiliary feedwater combined with relief through the main steam safety valves (MSSVs) remove heat from the RCS and controls steam generator (SG) pressure

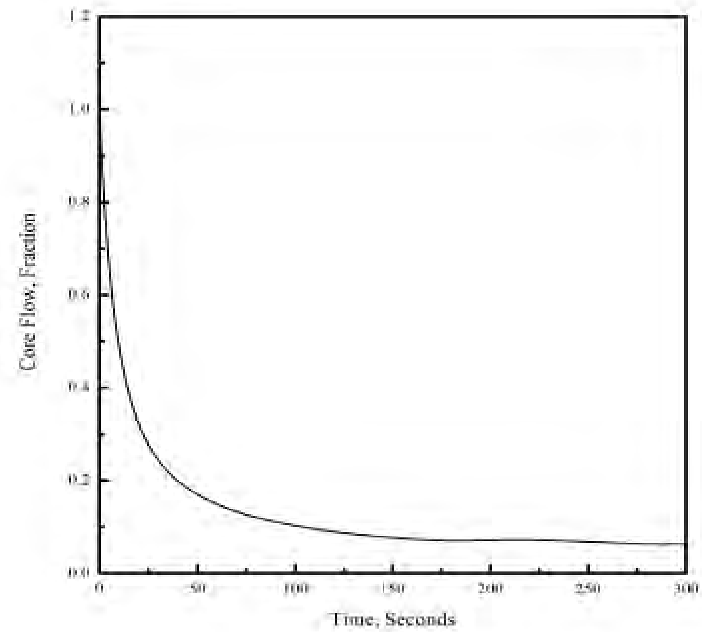
Relative Core Flow

Non-proprietary

TRACE/PARCS



DCD



Relative Core Nuclear Power

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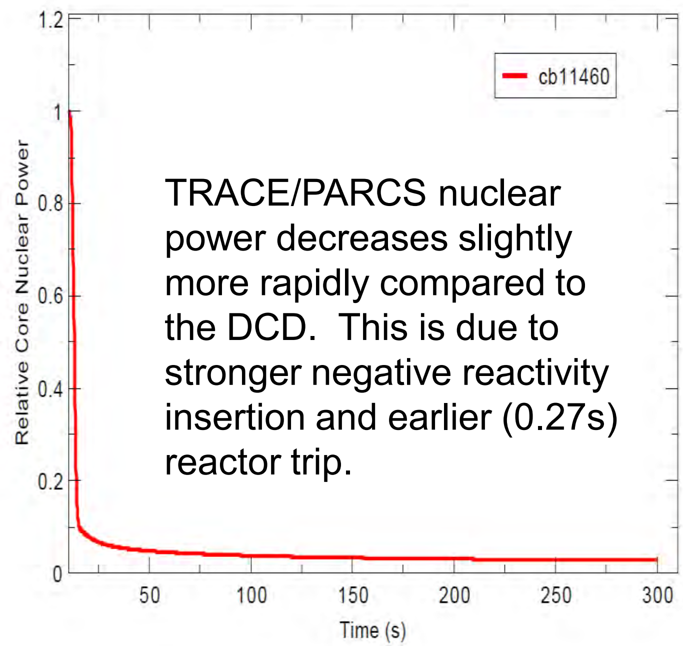
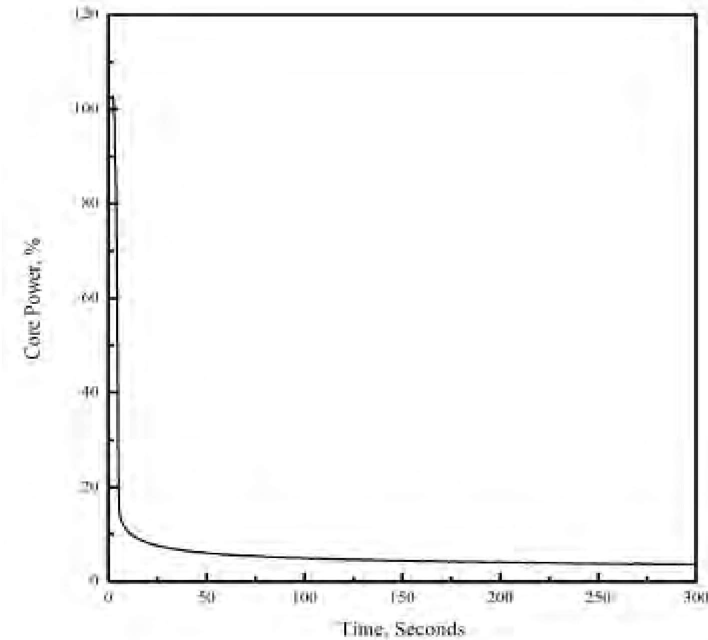
TRACE/PARCS



TRACE/PARCS nuclear power decreases slightly more rapidly compared to the DCD. This is due to stronger negative reactivity insertion and earlier (0.27s) reactor trip.

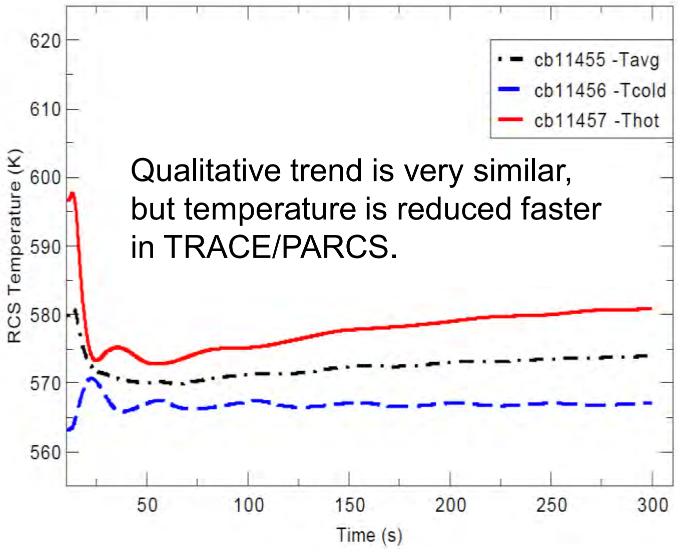
DCD



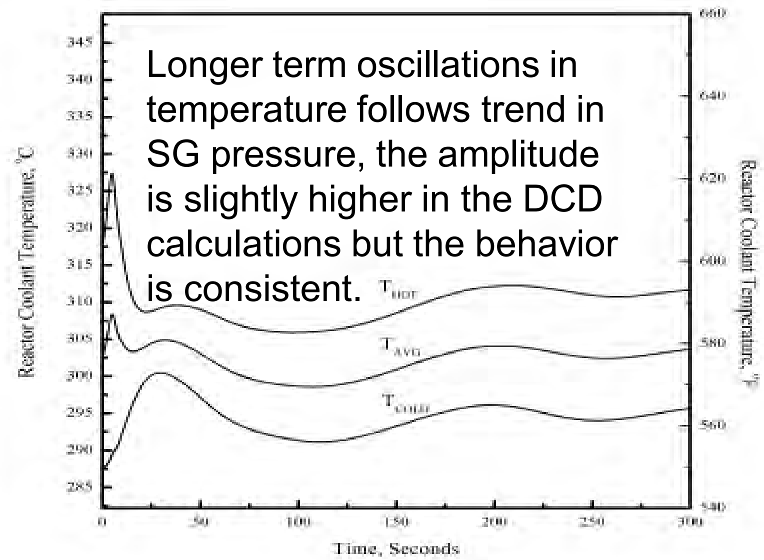
RCS Temperature

Non-proprietary

TRACE/PARCS



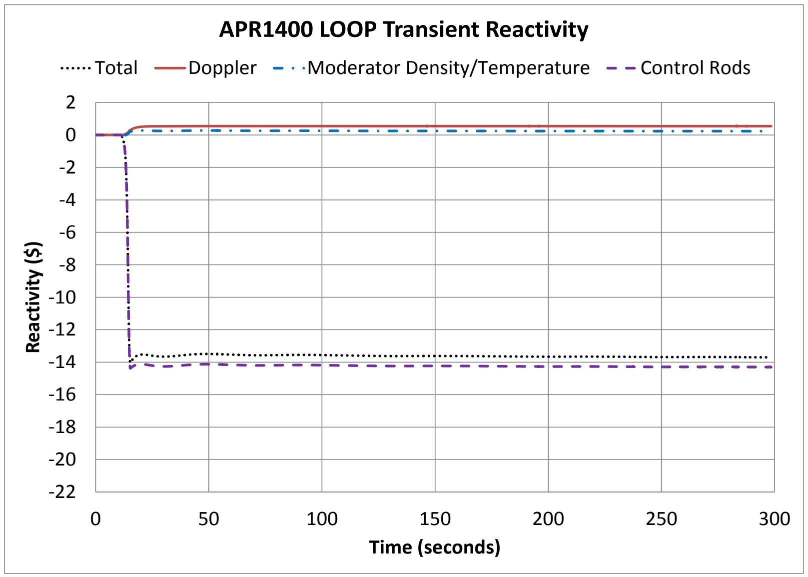
DCD



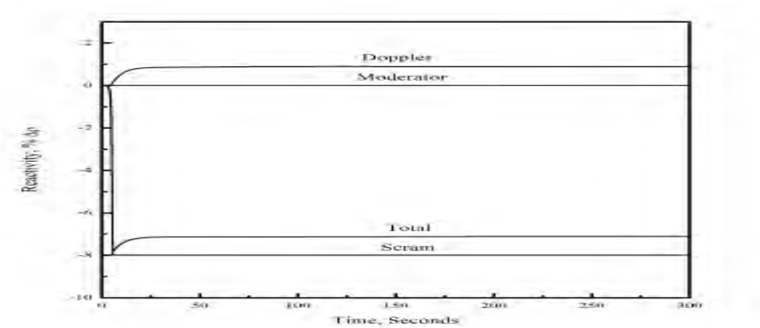
Non-proprietary

Reactivity

TRACE/PARCS



DCD

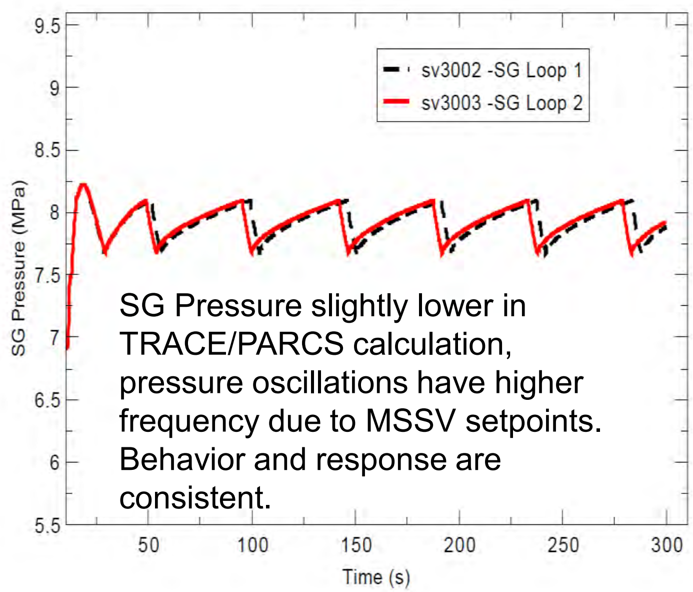


About 9.6%Dk/k rod worth predicted by TRACE/PARCS vs. 8%Dk/k assumed in the DCD calculations. Differences in Doppler and moderator worth are inconsequential and due to differences in input assumptions.

SG Pressure

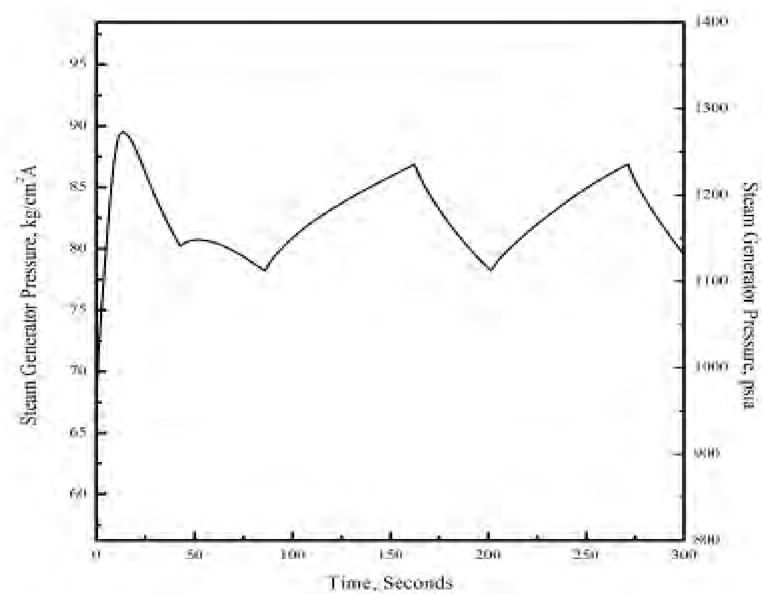
Non-proprietary

TRACE/PARCS



SG Pressure slightly lower in TRACE/PARCS calculation, pressure oscillations have higher frequency due to MSSV setpoints. Behavior and response are consistent.

DCD



mDNBR (note: not on the same scale)

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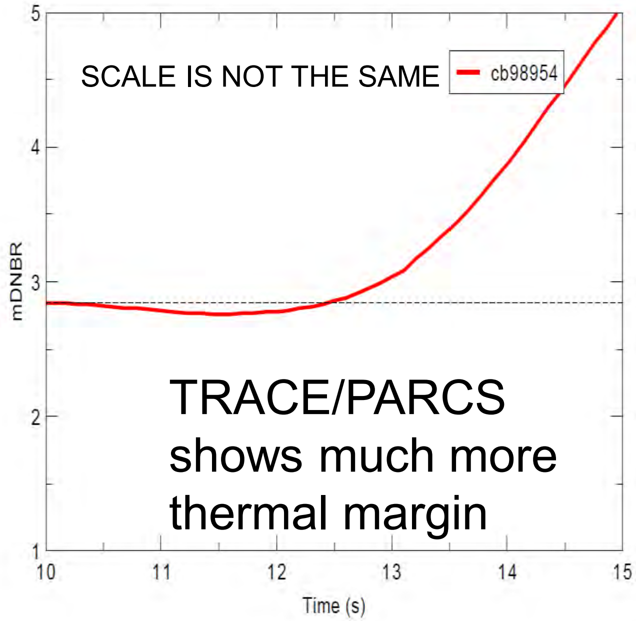
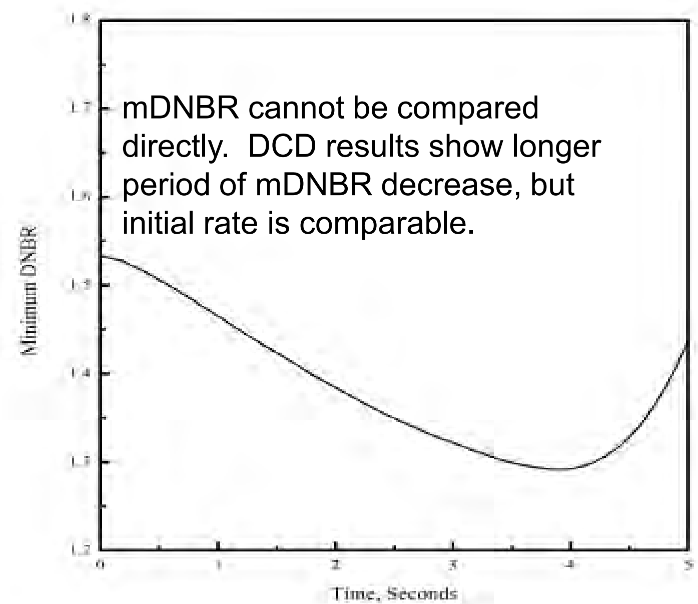
TRACE/PARCS



DCD



Conclusions

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- TRACE/PARCS and DCD analyses compare favorably in terms of major trends and overall system behavior
- TRACE/PARCS calculations indicate that the reactivity assumptions, in particular, the shutdown worth assumed in the DCD calculations, are conservative
- TRACE/PARCS predicts a milder change in mDNBR compared to the DCD, indicating conservatism in the analysis
- DCD analysis results appear to reasonably predict system behavior and to conservatively predict the thermal margin

15.3.3-15.3.4: RCP Malfunctions

Events evaluated

- 15.3.3 Reactor Coolant Pump Rotor Seizure (limiting event)
- 15.3.4 Reactor Coolant Pump Shaft Break

15.3.3-15.3.4: RCP Malfunctions

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Evaluation Model:

- CESEC-III for nuclear steam supply system (NSSS) modeling
- HERMITE for neutron diffusion
- TORC and CETOP with KCE-1 CHF correlation for departure from nuclear boiling ratio (DNBR)

Inputs and Assumptions:

- DCD Table 15.3.3-2 was reviewed against tech spec values and the referenced methodology found in topical report CENPD-183-A.

15.3.3-15.3.4: RCP Malfunctions

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Regulatory Findings – Loss of Forced Reactor Flow requirements

- The analysis is based on approved codes and methods and is therefore acceptable
- The input assumptions appropriately bound the potential conditions of the plant
- The loss of offsite power leads to automatic startup of EDGs, thereby demonstrating compliance with GDC 17
- Compliance with GDC 27 is demonstrated by showing that the operator can achieve cold shutdown during the event
- Compliance with GDC 31 is demonstrated by maintaining the RCS pressure within 110% of design pressure
- The calculated fuel failures bounded by the assumed values in the dose consequence analysis, therefore the requirements of 10 CFR Part 100 are met

15.4.1-15.4.3: Reactivity and Power Distribution Anomalies

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DCD Sub-sections

15.4.1 Uncontrolled CEA withdrawal from subcritical or low power startup condition

15.4.2 Uncontrolled CEA withdrawal at power

15.4.3 Control element assembly misoperation

15.4.1-15.4.3 – Reactivity and Power Distribution Anomalies

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Causes

Applicant considered all causes of event which is consistent with typical large PWRs for all the scenarios described in three sub-sections

Event and Event Analysis Results

For all three sub-sections, with several RAIs issued and closed, the staff finds the following:

- The limiting event progression and sequence were identified and analyzed
- The initial conditions, boundary conditions, core parameters are adequate. Appropriate conservativisms were applied
- The input assumptions under these three sub-sections covers all single failures, loss of power, trip delays, etc.
- Found the consequences of the accident are within SAFDL criteria and system response acceptable

MDNBR and PLHGR meet the SRP acceptance criteria with adequate margin

RCS pressure design limits are met with adequate margin

15.4.1-15.4.3 – Reactivity and Power Distribution Anomalies

Non-proprietary

Results

Analytical Methods

Based on the audits, the staff found that the approved methods were properly used to analyze the limiting events for all three sub-sections except the Open Item on thermal conductivity

Event Analyses Results

For all the events described in Section 15.4:

- SAFDLs are not exceeded using conservative assumptions with considered uncertainties
- General Design Criteria 10, 13, 17, 20, and 25 are met

15.4.4 - Startup of an Inactive Reactor Coolant Pump (SIRCP)

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- **Evaluation model**
  - Bounding hand calculation assumes the RCS average temperature instantaneously changes to the steam generator temperature

- **Input parameters and initial conditions**
  - Bounding isothermal temperature coefficient (most negative for cooldown, most positive for heatup)
  - Bounding values for temperature difference
  - Shutdown margin specified in technical specifications

- **Results**
  - Conservative analysis shows that subcritiality is maintained for SIRCP event (Modes 3 through 6)

15.4.6 – Inadvertent Decrease in Boron Concentration

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Review Results

- Open Item remains questioning that conservative dilution times are predicted using the complete mixing model for Modes 4 and 5 with only one shutdown cooling pump in service
- Open Item remains questioning that the CEA withdrawal event, which credits a VOPT trip, bounds a slow, at power dilution event
- Staff found that other dilution scenarios are conservatively analyzed and satisfy the SRP time criteria

15.4.7: Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

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Review Results

- The applicant has procedural controls including a fuel assembly ID verification process once the core load is complete
- The applicant surveils the planar peaking factor per TS 3.2.2 every 31 EFPD
- The BOC undetectable misloading peaking factor increase is bounded by the peaking factor increase from the CEA drop analysis; hence the minimum 95/95 DNBR limit is not violated

15.4.8: Spectrum of Control Element Assembly Ejection Accidents (CEAE)

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- **Three Analyses**
  - Peak RCS pressure
  - DNBR analysis
  - Fuel enthalpy analysis

- **Evaluation model**
  - CENPD-190-A “CE Method for Control Element Assembly Ejection Analysis”
    - ROCS, CESEC-III, CETOP, and STRIKIN-II
    - ROCS used to obtain ejected rod worths, pre and post axial and radial power distributions which are used in point kinetcs calculations (CESEC-III and STRIKIN-II)
  - Statistical convolution method (fuel failure fraction based on DNBR)

15.4.8: Control Element Assembly Ejection

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- **Input parameters and initial conditions**
  - Tables 15.4.8-1 through 15.4-8-4 of SER provide basis for input values
  - RPS actuated on a variable overpower trip (VOPT). VOPT setpoint includes excore penalty to account to decalibration
  - Analysis considers a LOOP and single failure
  - Operator action is taken at 30 minutes
- **Results/Findings**
  - DCD analyses do not account for thermal-conductivity degradation – Open Item

15.4.8: Spectrum of Rod Ejection Accidents Staff Initial Cycle Confirmatory Analyses

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NRC Methods

- Three-dimensional transient core response using TRACE/PARCS coupled code suite
- PARCS standalone depletion calculations performed for Section 4.3 confirmatory analysis used as initial conditions for BOC and EOC analysis
- TRACE Model
  - Based on model used for LBLOCA analysis
  - Added 3D Cartesian vessel to represent reactor core - each radial node in PARCS one-to-one mapped to radial node in core vessel
  - One TRACE heat structure per assembly
  - DNBR calculator implemented via control systems
- PARCS Model
  - Essentially the same as Section 4.3 analysis

15.4.8: Spectrum of Rod Ejection Accidents Staff Initial Cycle Confirmatory Analyses

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Applicant Assumptions vs. Staff Methods

- Applicant uses a conservative point kinetics method to evaluate reactivity feedback and power response
  - PK gives analyst considerable freedom in biasing input parameters
  - Conservative inputs for ejected rod worth, MTC/DTC, delayed neutron fraction
  - Resulting power transient fed to conservative thermal-hydraulic analysis
- Staff methodology is to physically simulate the removal of a control rod from the initial core at BOC and EOC for various power levels
  - Ejected rod worth is dependent upon initial flux shape and depletion characteristics
  - Delayed neutron fraction dependent upon depletion characteristics
  - Fuel temperature and moderator feedback explicitly calculated
  - Neutronics reacts to changes in T/H and vice versa

15.4.8: Spectrum of Rod Ejection Accidents Staff Initial Cycle Confirmatory Analyses

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Results and Conclusions

- Staff calculations indicate considerable conservatism in applicant's methods and results
  - Extremely high rod worths
  - Very low β
  - Conservatively low Doppler feedback, no credit for negative moderator feedback

- For example, at HZP applicant assumes rod worth of 1.08$
  - TRACE/PARCS EOC HZP rod worth ~0.26$
  - Results in Power increase by factor of 3 rather than 8 order of magnitude
  - Realistic results indicate ample margin for initial core

- Confirmatory analysis supports reasonable assurance finding with respect to APR1400 REA analysis

15.5.1: Inadvertent ECCS Actuation

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- **Qualitative evaluation (No evaluation model)**
- **3 Cases considered**
  - Reactor coolant system (RCS) pressure above safety injection (SI) pump shutoff head
  - RCS pressure below SI pump shutoff head
  - Plant is on shutdown cooling (SDC), low temperature overpressure protection (LTOP) is available
- **Input parameters for evaluation are provided in Table 15.5.1-1 of the SER**

15.5.1: Inadvertent ECCS Actuation

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- **Results**
  - MODE 1 and MODE 2: SI cannot inject, thus no impact to RCS
  - MODE 3 and MODE 4 (not on SDC):
    - Cooldown: RCS pressure limit at LTOP enable temperature is greater than SI pump shutoff head
    - Heatup: RCS pressure limit at LTOP disable temperature is above SI pump shutoff head
  - MODE 4, MODE 5, and MODE 6 (SDC): LTOP relief capacity is much larger than the four pump SI capacity

15.5.2: CVCS Malfunction that Increases Inventory in RCS

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- **Evaluation model**
  - CESEC-III for NSSS modeling
  - CETOP with KCE-1 CHF correlation for DNBR
- **Input parameters and initial conditions evaluated in Table 15.5.2-1 of the SER**
  - RPS actuation on high RCS pressure
  - Operator action to initiate a cooldown at 30 minutes
  - LOOP occurs coincident with reactor trip
  - No pressurizer heaters to maximize RCS inventory

15.5.2: CVCS Malfunction that Increases Inventory in RCS

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- **Results/Findings**
  - Fuel integrity is maintained (minimum DNBR of 1.5177)
    - GDC 10 and GDC 26
  - Peak pressure in RCS (2,649 psia) and steam generators (1,294 psia) maintained below 110 percent of design value
    - GDC 15
  - Overfill analysis not performed
    - Pilot operated safety relief valves (POSRVs) qualified for water and two-phase mixture passage
    - NRC staff reviewed POSRV design specification during quality insurance inspection
    - CVCS malfunction does not lead to an event with more serious consequences

15.6.1: Inadvertent Opening of a Pressurizer Pressure Relief Valve

Non-proprietary

- Evaluated as a postulated accident as part of small break loss of coolant accident (SBLOCA)
- Pilot Operated Safety Relief Valve (POSRV) opening, due to spurious signal or operator error, is prevented by disconnecting electrical power from the upstream motor operated pilot valve
  - LCO 3.4.10 “Pressurizer Pilot Operated Safety Relief Valves”

15.6.2: Failure of Small Lines Carrying Primary Coolant Outside Containment

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Non-proprietary

- **Evaluated as a double-ended letdown line break (LDLB) outside of containment**
  - Larger than any instrument or sample line (DCD Table 6.2.4-1)
- **Evaluation model**
  - CESEC-III for NSSS modeling
  - CETOP with KCE-1 CHF correlation for DNBR
- **Input parameters and initial conditions evaluated in Table 15.6.2-1 of the SER**
  - Single failure of an isolation valve does not impact the event
  - Operator action taken at 30 minutes to isolate the break and trip the reactor
- **Results**
  - LDLB reduces pressurizer level, but pressurizer heater maintains pressure at operating conditions
  - 44,700 lb of reactor coolant leaks out of break

15.6.3: Steam Generator Tube Rupture (SGTR)

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- **Two analyses**
  - Investigate thermal-margin (i.e., DNBR)
  - Evaluate radiological consequences
- **Evaluation model**
  - CESEC-III for NSSS modeling
  - CETOP with KCE-1 CHF correlation for DNBR
- **Input parameters and initial conditions evaluated in Table 15.6.3-1 and Table 15.6.3-2 of the SER**
  - SGTR analysis credits RPS, main steam isolation, and safety injection to mitigate the event
  - LOOP occurs coincident with reactor trip
  - MSIVs close instantaneously on high steam generator level
  - Operator action at 30 minutes

15.6.3 SGTR

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- **Results/Findings**
  - Fuel integrity is maintained (minimum DNBR of 1.3022)
  - Peak pressure in RCS and steam generators (1,195 psia) maintained below 110 percent of design value
  - Steam generator overfill does not occur
  - Mass leak through break is consistent with the value used in the dose consequence analysis for SGTR

Section 15.8 – Anticipated Transient Without Scram (ATWS)

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- APR1400 design includes a Diverse Scram System, alternately referred to as the Diverse Protection System (DPS), which provides a diverse backup to the Plant Protection System (PPS). Chapter 7 review concluded that DPS fulfils 10CFR 50.62 requirements

- APR1400 Reactor Trip System and DPS design are similar to those of the Combustion Engineering System 80 and 80+ designs

- Applicant analyzed the limiting ATWS events considered in the APR1400 PRA and reached a similar conclusion to that of previously-approved CENPD-158

Staff findings:

- APR1400 design meets the ATWS rule, 10 CFR 50.62

Section 15.6.5 – Large Break LOCA

Non-proprietary

Significant Analysis Issues

- Loss of Offsite Power and limiting single failure assumptions
  - Reactor Coolant Pump (RCP) trip
    - Pump forward/reverse flow resistance
  - Control Element Assembly (CEA) Insertion
  - Safety injection Tank (SIT) check valve active/passive failure
- Unrealistic power spike (DCD Figure 15.6.5-13) due to moderator reactivity curve error (affected all cases)
- Staff review of Topical Report APR1400-F-A-TR-12004-P is incomplete
  - Significant progress made on topical report review, but not complete
  - Tracked as Open Item in Chapter 15 FSER
- Full spectrum LBLOCA analyses being re-performed to address TCD and other methodology issues

Findings

Technical issues resolved except for TCD and topical report methodology

15.6.5: Small Break LOCA Staff Review Summary

Non-proprietary

- SBLOCA Methodology and Computer Codes
- SBLOCA Analysis Input Assumptions
- *Initial Loop Seal Formation and Clearing*
- *Break Spectrum Analysis*

15.6.5: Small Break LOCA APR1400 Conceptual Design

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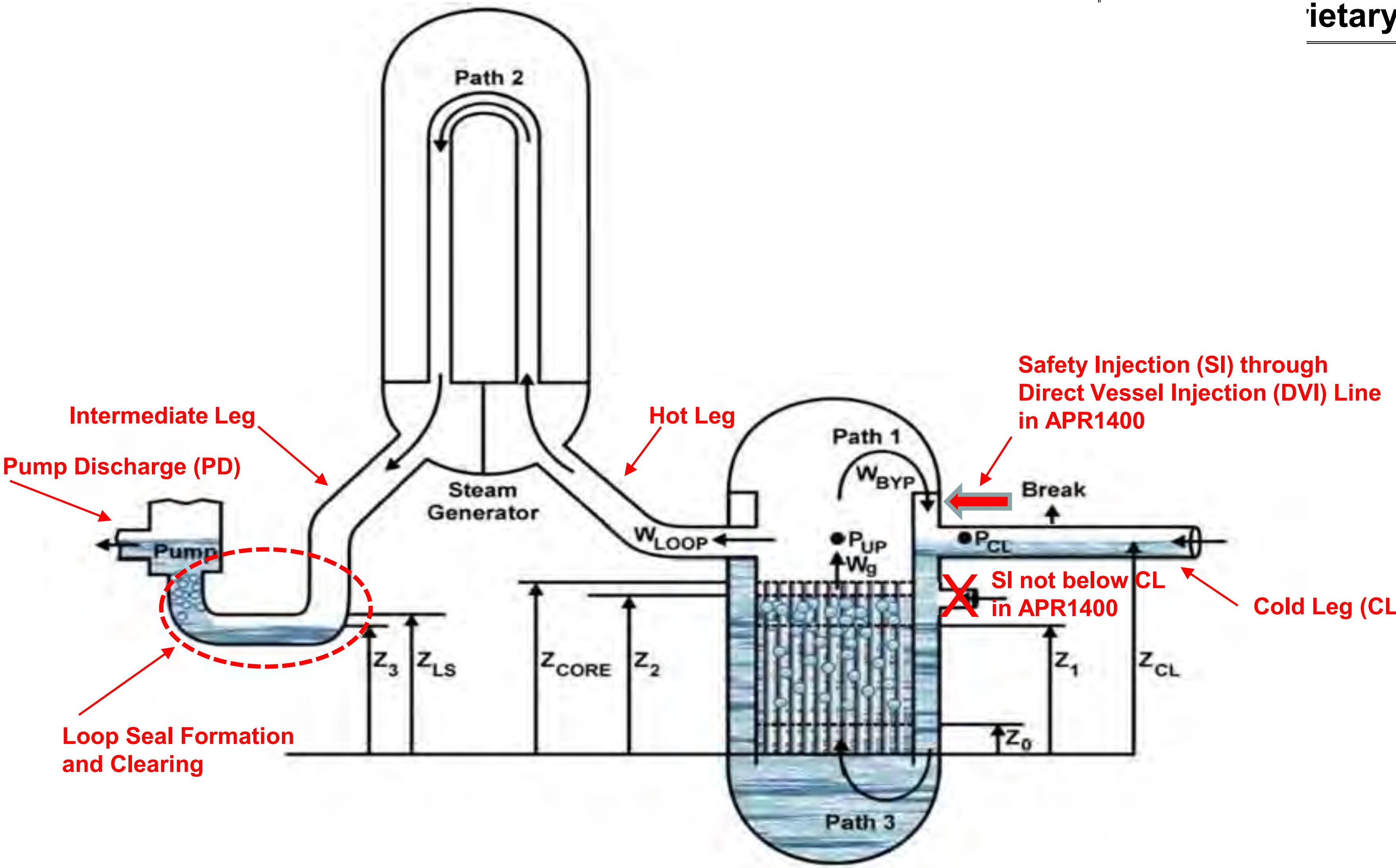

15.6.5: Small Break LOCA Initial Loop Seal Clearing

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- **Loop Seal Modeling Conservatisms in APR1400 SBLOCA EM**
  - Loop seal clearing is delayed until the SG side level reaches the bottom of the horizontal segment of the cross-over piping
  - Appendix K based methodology
  - Lumping two loop seals for intact cold legs into a single loop seal, for the limiting SBLOCA
  - CEFLASH-4AS licensing basis calculations are more conservative than the applicant's RELAP5 and staff's TRACE confirmatory calculations
  - CEFLASH-4AS validated for the loop seal clearing phenomena in Semiscale Test SUT-8

- **Findings**
  - SBLOCA methodology and computer codes conservatively characterize the safety-significant phenomena of (1) loop seal formation and clearing, and (2) PCT during the limiting SBLOCA for the initial phase of blowdown and reflood
  - Overall conservatism in the S1M SBLOCA methodology

15.6.5: Small Break LOCA
Break Spectrum Analysis Summary

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Non-proprietary

- **Summary of Review**
  - Two types of break spectrum analyzed (PD, DVI line)
    - 4 DVI line breaks: 18.6, 46.5, 93, 372 cm^2
    - 4 CL breaks at PD: 46.5, 93, 325, 465 cm^2
  - Break sizes analyzed too coarse to identify the limiting SBLOCA break
  - Finer break spectrum analyses
    - 15 DVI line breaks: 1.5-8.5 inch ~ 11.4-372 cm^2 ~ 0.0123~0.4006 ft^2
    - 17 CL breaks at PD: 2-10 inch ~ 20.2-507 cm^2 ~ 0.0218~0.5454 ft^2
    - PCT, details of loop seal clearing for each SBLOCA break size and any core uncovery, two-phase mixture levels, void fractions, vapor mass flow rate, injection flow rate, core pressure, and break flowrate
- **Findings**
  - 5 inch dia. (127 cm^2~0.1364 ft^2) DVI line break identified as the limiting SBLOCA with highest PCT of 1,684°F
    - **516 °F margin** to the 2200 °F limit
    - TRACE Confirmatory Calculations: PCT_{max} = 1265°F~ **935°F margin**
  - **Confirmatory Item:** DCD/TeR spectrum analysis markups submitted

15.6.5: Small Break LOCA Conclusions

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- Sufficient conservatism exists in the APR1400 SBLOCA analysis using the S1M methodology to meet the regulatory requirements for LWRs given in 10 CFR 50.46 and Appendix K to 10 CFR Part 50

- **Confirmatory Item**
  - Detailed SBLOCA break spectrum analysis acceptable
  - Revisions to the DCD and the SBLOCA TeR submitted

- **Open Item**
  - Justify the upper bound on the SBLOCA break size of 464.5 cm^2
  - Docketed response not submitted

Non-proprietary

APR1400 TRACE Large & Small Break LOCA Confirmatory Analysis

William Krotiuk
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Joseph Staudenmeier
RES/CRAB

ACRS Subcommittee Meeting
May 19, 2017

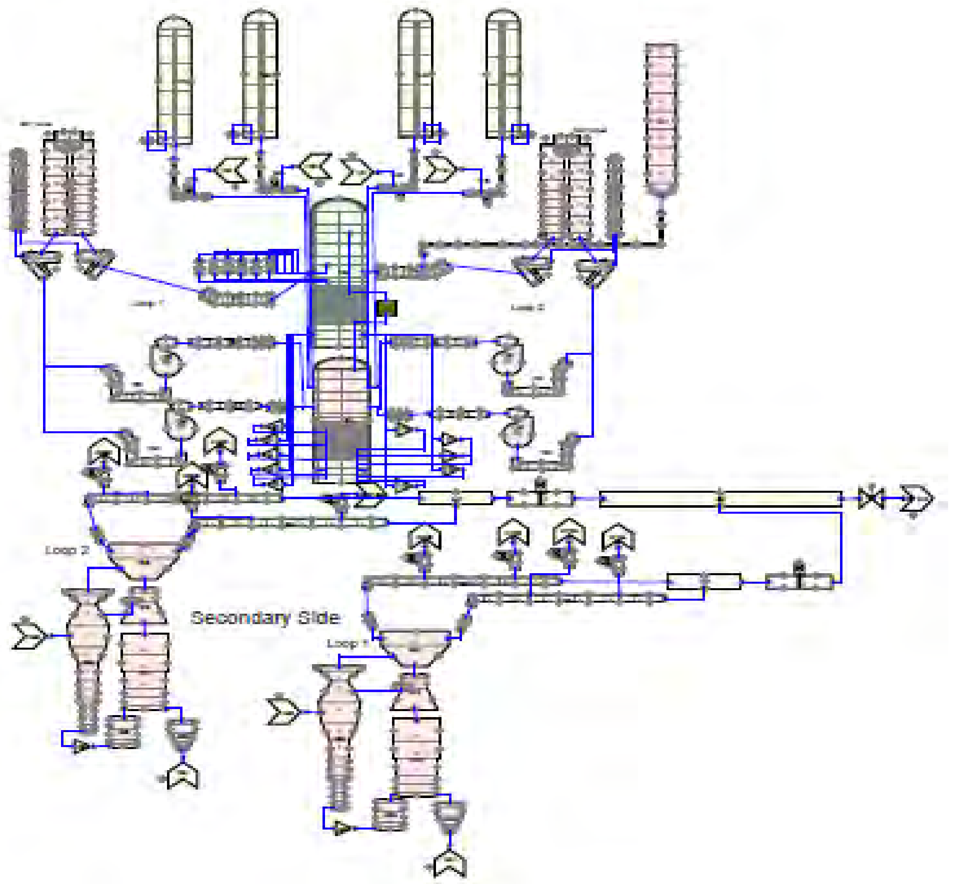
Discussion Topics

Non-proprietary

- **This presentation provides**
  - A description of the independently developed APR1400 TRACE model
  - The TRACE w large break LOCA reported in DCD 15.6.5
  - The TRACE confirmatory analyses results for the small break LOCAs reported in DCD 15.6.5
  - Results for the fuel thermal conductivity degradation sensitivity study following a LBLOCA

Non-proprietary

APR1400 Primary/Secondary System TRACE Model



APR1400 Steady-State Comparisons

Non-proprietary

- A steady-state execution of the TRACE APR1400 was performed to obtain initial conditions for all LBLOCA and SBLOCA transient analyses
- Initial system 102% power, pressurizer pressure and water level, safety injection tank water level and temperature obtained from the APR1400 DCD
- Steady-state system conditions for pressure, temperature, flow, pressure drop and system flow distribution obtained from chapters 4 and 15 of the APR1400 DCD
- Emergency safety feature set points and delays obtained from the APR1400 DCD
- Results for the APR1400 TRACE steady-state conditions closely compare to results from the APR1400 DCD calculated using the KHNP RELAP5 code

APR1400 LBLOCA Comparison

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- Assumptions
  - Double ended guillotine break in the primary system pump discharge leg 1B
  - Failure of two safety injection pumps consistent with the APR1400 DCD
  - All safety injection tank flow available consistent with the APR1400 DCD
- Conclusion
  - TRACE LBLOCA major predictions are similar to or bounded by the results reported in the APR1400 DCD using the KHNP RELAP5 code.

APR1400 System Parameter	Transient Condition	
	APR1400 DCD	TRACE
Blowdown PCT (ºC, K, °F) (trhmax-1999)	892ºC, 1165.2K 1637.7°F	822.1ºC, 1095.2K 1511.7°F
Blowdown PCT location (m)	2.57	2.83
Blowdown PCT time (s)	6.5	9.9
Reflood PCT (ºC, K, °F) (trhmax-1999)	798.9ºC, 1072.1K 1470.1°F	721.4ºC, 994.5K 1330.4°F
Reflood PCT location (m)	2.57	2.60
Reflood PCT time (s)	36.5	50.0
RELAP5 Peak Local Oxidation (%) / TRACE % equivalent clad reacted (ecr)	1.50 --	-- 0.855 max ecr
Fuel Rod Rupture	None	None

APR1400 LBLOCA Comparison

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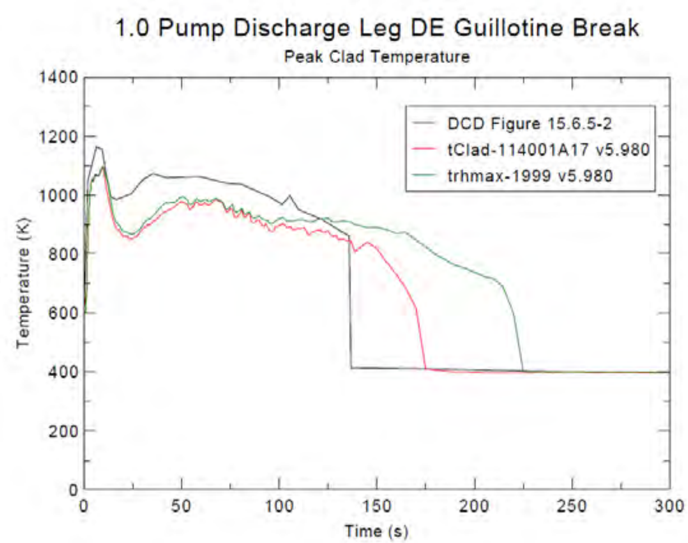
- **Conclusions**
  - The TRACE LBLOCA PCT is below the APR1400 DCD predictions using the KHNP RELAP5 code
  - The TRACE calculated quench time is larger than the value reported in the APR1400 DCD



Red line – Clad temperatures at fuel rod location with PCT, Green line – PCT for all fuel rods

APR1400 SBLOCA Comparisons

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- **Assumptions**
  - SBLOCA analyses performed for a spectrum of break sizes and locations
  - Emergency safety feature failures and delays consistent with the DCD

Break Location	Break Size (cm^2)	SI Pump Operation	SIT Availability
Pump Discharge (PD)	465[1] 325 93 46.5 23.25	Two SI pumps operate, two SI pumps fail due to emergency diesel failure	Four SITs available
Direct Vessel Injection (DVI)	372 93 46.5 18.6	One SI pump operates, two SI pumps fail due to emergency diesel failure, one SI flows out the break	Four SITs available, but one SIT flows out the break
Pressurizer Top - Safety Relief Valve (RV)	27.9	Two SI pumps operate, two SI pumps fail due to emergency diesel failure	Four SITs available
Instrument tube at bottom reactor vessel head	2.8	Two SI pumps operate, two SI pumps fail due to emergency diesel failure	Four SITs available

APR1400 SBLOCA Comparisons

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- APR1400 SBLOCA DCD results calculated using several codes similar to methods used for operating Combustion Engineering PWRs

- APR1400 SBLOCA DCD calculations did not include reactivity feedback.
- TRACE SBLOCA calculations performed with and without reactivity feedback

- Calculated PCTs from the DCD and TRACE analyses did not occur at same break size

Break Location	Break Size (cm^2) / % SE Pipe/Nozzle/Valve Break Area	APR1400 DCD		TRACE Analyses no feedback / feedback	
		PCT (K)	Max. Clad Oxidation (%)	PCT (K)	ECR (%)
Pump Discharge (PD)	465[1] / 9.94% pipe area	771.15	0.0017	678.2/761.3	0.759/0.759
	325 / 6.95% pipe area	765.15	0.0015	836.9/849.9	0.759/0.759
	93 / 1.99% pipe area	838.15	0.0010	**939.5/978.8**	0.759/0.798
	46.5 / 0.994% pipe area	**841.15**	0.0008	801.9/875.3	0.759/0.760
	23.25 / 0.497% pipe area	NA	NA	809.6/778.3	0.759/0.760
	11.625 / 0.249% pipe area	NA	NA	621.2[2]/621.2[2]	0.759/0.760
Direct Vessel Injection (DVI)	372 / 101.6% nozzle area	**897.15**	0.0195	801.4/796.0	0.759/0.759
	93 / 25.4% nozzle area	842.15	0.0069	986.4/832.0	0.759/0.760
	46.5 / 12.7% nozzle area	844.15	0.0018	**1019.0**/816.8	0.796/0.760
	18.6 / 5.08% nozzle area	889.15	0.0029	926.1/**905.0**	0.759/0.760
	9.3 / 2.54% nozzle area	NA	NA	621.3[2]/621.3[2]	0.759/0.760
Pressurizer Top - Safety Valve (PORV)	27.9 / 100% valve area	**841.15**	0.0006	621.4[2]/621.4[2]	0.759/0.760
Instrument tube at bottom reactor vessel head	2.8 / 100% tube area	Not available	Not available	621.3[2]/621.3[2]	0.759/0.760

[2] Initial pre-SBLOCA steady-state temperature

TRACE LOCA Long Term Cooling Assessment

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- **Long term cooling core uncovery due to loop seal re-formation and clearing**
  - Best Estimate LBLOCA and the 46.5 cm^2 DVI SBLOCA are analyzed to determine if a second PCT occurs
- **Assumptions**
  - Sump recirculation time estimated assuming 2 of 4 SI pumps available consistent with the shorter term analyses
  - 625,000 gallons available in the IRWST as specified in DCD section 6.8.4
  - All analyses performed with reactivity feedback.
  - LBLOCA estimated recirculation time is about 4.79 hr. (~17,229 s).
  - SBLOCA estimated recirculation time is about 5.44 hr. (~19,600 s).
- **A second PCT, if present, will be below the initial PCT**
  - Long term vapor generation rate is too low to support extended deep loop seal and core level depression. The associated heatup from a given core depression is less because of lower decay heat
- **TRACE calculations do not show a second core uncovery and heatup**

Non-proprietary

Fuel Thermal Conductivity Degradation Study

- Determine sensitivity of steady-state fuel rod average and average centerline temperatures and LBLOCA PCT to fuel thermal conductivity degradation resulting from burnup.

Fuel Rod Temperatures from TRACE DCD LBLOCA Analysis

TRACE Heat Structure	Primary Fuel Rod	Supplemental Rod 1	Supplemental Rod 2	Supplemental Rod 3
Fuel Rod Description	First Core Avg.	Cycle 3 Hot Rod	First Core Avg. at EOC	BOC
GWD/MTU	17.571	60.0	28.914	2.0
Steady-State Temperatures (°F)				
Max. Centerline at 2.771 m axial height (coarse axial level 49 of 68)	1995.1	3763.4	3301.7	2717.3
Min. Centerline at 0.0 m axial height (coarse axial level 1 of 68)	573.0	599.9	587.1	578.2
Fuel Rod Centerline Average	1468.5	2637.0	2281.0	1865.5
Max. Avg. at 2.771 m axial height (axial level 17of 24)	1302.7	2187.8	1917.8	1619.8
Min. Avg. at 0.0 m axial height (axial level 1 of 24)	565.1	580.8	572.6	569.0
Fuel Rod Average	1037.1	1620.2	1425.9	1220.4
Transient LBLOCA Temperatures (°F)				
PCT	1117.0	1511.7	1424.2	1298.3

APR1400 LBLOCA & SBLOCA Conclusions

Non-proprietary

- **Best Estimate LBLOCA**
  - TRACE predictions are similar to or bounded by the reported APR1400 DCD results. The calculated clad temperatures have a significant margin to 50.46 limits
  - TRACE event sequencing predictions are similar to the DCD reported values
- **SBLOCA**
  - TRACE PCT predictions do not match DCD reported behavior. The differences may be partially attributable to the absence of CCFL modeling in the APR1400 calculational method. The calculated clad temperatures have a significant margin to 50.46 limits
  - The TRACE analyses with and without reactivity feedback illustrate its importance in predicting SBLOCA results. The DCD calculations did not include reactivity feedback
- **TRACE long term cooling calculations show no late heatup**
  - A secondary PCT due to SG primary system condensation or loop seal refill is not expected to occur for LBLOCAs or SBLOCAs unless an external action (e.g. operator action) occurs
  - The magnitude of any secondary PCT, if it occurs, would be less than the initial PCT because of the drop in primary system pressure

15.6.5: Post LOCA Long Term Cooling Boron Precipitation

Non-proprietary

- **Technical Report** - APR1400-F-A-NR-14003, “Post-LOCA Long Term Cooling Evaluation Model” has details of boron precipitation analysis
- **Methodology** – slight modification of CENPD-254-P-A, "Post-LOCA LTC Evaluation Model", June 1980 (Proprietary)
  - Realignment of SI from RWT to sump not needed; IRWST is SI source
  - Revision to mixing volume boundary
- **Long Term Cooling Plan** - operator action needed to assure core cooling, while avoiding boric acid precipitation
- **Timing of Switchover** to combined DVI and hot leg injection by operator - key to avoiding boron precipitation

Computer Codes

- BORON (SI source modified), NATFLOW, CEPAC, CELDA

15.6.5: Post LOCA Long Term Cooling Boron Precipitation

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Non-proprietary

- Majority of issues required clarifications and additional information which was provided and acceptable
- Significant item was assumption regarding size of the mixing volume
- Proposed methodology not consistent with current PWR vendor methodologies
- KHNP decreased mixing volume size (reduced boundary from top to bottom of hot leg elevation in RPV)
- Resulted in decrease of maximum time of switchover to combined injection from 3 to 2 hours. DCD modified

Staff Findings

Boron precipitation long term cooling methodology, analysis and plan are acceptable for their application to APR1400

15.6.5: Post LOCA Long Term Cooling Boron Dilution

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Non-proprietary

Staff Findings

Analytical Methods

In the current version of the SER, RAI 15.06.05-22 which addresses boron dilution following an SBLOCA , is an OPEN ITEM.

- The applicant has since responded providing calculations of boron dilution due to RCP startup and restart of natural circulation (NC) simultaneously in all loops
- PKL tests show NC does not get established simultaneously in all loops, so assumption of simultaneous restart is conservative
- Mixing volumes used in dilution calculations were verified to be conservative.

Applicant asserted that closure of GSI-185 also applied to APR1400 because its geometry with regard to boron dilution is nearly identical to the C-E System 80+

Conclusions

Minimum calculated boron concentration is well above the calculated limit for criticality

15.6.5: Post LOCA Long Term Cooling LTC Loop Seal Clearing

Non-proprietary

Staff Findings

Analytical Methods

Supplemental RAI response provided calculations SBLOCA break spectrum calculations to 2 hours (hot-leg switchover time) to identify limiting loop seal clearing case

- 0.044 ft^2 break limiting with periodic loop seal reformation/clearing in one loop
- Slight periodic uncovery of very top of the core
- Maximum calculated LTC cladding temperature was 627 F

Conclusions

NRC criteria that the PCT remain below 800 F during LTC loop seal reformation has been satisfied, with the exception of one open item

15.6.5: Post LOCA Long Term Cooling In-vessel Downstream Effects

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1. Debris source analysis and fiber only by-pass testing
2. Available driving head across the debris bed
3. Fuel assembly head loss testing
4. LOCA deposition model

Regulatory Requirements And Precedence

15 gram/assembly fiber loading clean plant criteria has been approved and used for US operating PWR plants

(SER on WCAP16793 “Evaluation of Long-Term Core Cooling Considering Particulate Fibrous and Chemical Debris in The Recirculating Fluid” Rev. 2 #ML121020118 December, 12, 2012)

15.6.5: Post LOCA Long Term Cooling In-vessel Downstream Effects

Non-proprietary

1. **Debris source analysis and fiber only by-pass testing**
   - Reflective Metallic Insulation (RMI) is used
   - Removal of all fiber insulation from ZOI
   - Limit latent fiber amount to 15 lbm at the strainer
   - Fiber only by-pass testing was conducted using the in-line filtration system
     - The same prototypical strainer head loss testing facility was used to preserve the 1:1 approaching velocity ratio through the strainer surface
     - Staff audited and observed the actual test
   - Only 6.8 g fiber per assembly is predicted. However, 15 g was used in the fuel assembly head loss testing and the core DP analysis

Staff findings, Debris Source Analysis

- Debris source analysis and testing are conservative

15.6.5: Post LOCA Long Term Cooling In-vessel Downstream Effects

Non-proprietary

2. Available driving head across the debris bed

- All limiting core flow conditions have been identified
- Conservative debris arrival time is assumed
- Limit latent fiber amount to 15 lbm at the strainer
- Core two-phase flow friction loss, acceleration term and liquid density are determined properly in response to staff RAIs and audit questions
- As part of the available driving head, the water column height in the steam generator tube is conservatively treated to add margin

Staff findings, Available Driving Head

- Available driving head across the core has been conservatively determined

15.6.5: Post LOCA Long Term Cooling In-vessel Downstream Effects

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3. Fuel Assembly Head Loss Testing

- A new test facility was deployed by KHNP to measure the head loss across PLUS7 fuel bundle with the presence of debris
- The facility and the testing protocols have been adjusted in response to three staff inspection findings
- 15 g/assembly fiber insulation material was introduced including all the particulate and chemical precipitates
- For all the limiting flow conditions, the measured head loss values were significantly less than the core available driving head

Staff findings, Fuel Assembly Head Loss Testing

- The head loss tests were conducted properly and the measured bundle head loss was less than the available driving head

15.6.5: Post LOCA Long Term Cooling In-vessel Downstream Effects

Non-proprietary

4. LOCA Deposition Model

- PWR Owner's Group LOCA deposition model was used to predict the cladding temperature with the presence of crud build up
- Very conservative fiber and other debris loading was assumed
- The results show that the crud thickness is significantly less than the limit value of 50 mills. The PCT is much less than 800°F limit
- RAI was issued to obtain more detailed information about the inputs

Staff findings, LOCA Deposition Model

- During 30 days of the long term cooling, the crud formation shall be less than the thickness limit and the PCT shall be less than 800°F pending on the final closure of the RAI

ACRONYMS

Non-proprietary

AOO – anticipated operational occurrence
ASME – American Society of Mechanical Engineers
BOC – beginning of cycle
BOL – beginning of life
CE – Combustion Engineering
CEA – control element assembly
COL – combined license
COLSS – core operating limits supervisory system
CPC – core protection calculator
CPCS – core protection calculator system
CR - control room
CRD – control rod drive
CRDS – control rod drive system
CVCS – chemical and volume control system
DBA - design basis accident
DBE – design basis event
DCD – design control document
DNB – departure from nucleate boiling
DNBR – departure from nucleate boiling ratio
ENDF – Evaluated Nuclear Data File
ESFAS – emergency safeguards features actuation system
EOC – end of cycle
EOL – end of life
FHA - fuel handling accident
FTC – fuel temperature coefficient
GDC – general design criterion/criteria
ITAAC – inspections, tests, analyses, and acceptance criteria
IOSGADV - Inadvertent Opening of a Steam Generator Relief or Safety Valve
LPD – local power density
LOCA - loss of coolant accident
MDC – moderator density coefficient
MSLB - main steam line break
OPR1000 – Optimized Power Reactor 1000
PPS – plant protection system
RAI – request for additional information
RCS – reactor coolant system
RG - regulatory guide
RPS – reactor protection system
SAFDL – specified acceptable fuel design limit
SGTR - steam generator tube rupture
SI – safety injection
SIT – safety injection tank
SRP – Standard Review Plan
TCD – thermal conductivity degradation
TS – technical specifications
TEDE - total effective dose equivalent
VOPT- variable overpower trip

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Non-proprietary

Details of TRACE LOCA Model

TRACE Modeling of SITs with Fluid Device

APR1400 TRACE Model BACKUP SLIDE

Non-proprietary

- TRACE APR1400 PWR model was independently developed
  - Developed primarily from KHNP drawings and reports using methods from the USNRC “TRACE PWR Modeling Guidance”
  - Reactivity feedback table inputs obtained from KHNP RELAP5 model input
- Modelled systems include:
  - Primary System
    - Two hot and four cold leg pipes
      - CCFL model included for the hot leg bend at the SG inlet and for the cold legs at the pump inlet.
    - U-tube steam generator (SG) – primary side
      - CCFL model included for the u-tubes at the SG inlet.
    - Pressurizer
    - Reactor pressure vessel with 34 levels, 3 rings and 10 azimuthal segments
      - CCFL model included for the reactor vessel guide tubes and for the upper core plate.
    - Reactor core
      - Detailed fuel rod models specified peaking factors and burnups for three cycles.
      - Point kinetics modeling with reactivity feedback
    - Direct vessel injection (DVI) ECCS system using
      - Four safety injection tanks each with a fluidic device to control flow as a function of level
        - » TRACE model developed for SITs with a fluidic device provided reasonable agreement with KHNP test measurements
      - Four trains of high pressure safety injection pumps

APR1400 TRACE Model (continued)
BACKUP SLIDE

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- Modeled systems include:
  - Secondary System
    - U-tube steam generator – secondary side
    - Steam and feedwater pipes
    - Four main steam safety valves
    - Atmospheric dump valves
    - Feedwater control system to maintain SG downcomer level

Overview of Methods: TRACE/PARCS BACKUP SLIDE

Non-proprietary

- **Reactor systems response computed by TRACE**
  - RCS is modeled with 2 VESSEL components to accurately track temperature gradients in the downcomer
  - Reactor core is modeled with a 3rd VESSEL component that models each fuel assembly individually for detailed representation of the core
  - Reactor trip is caused by LOOP with 550 ms delay
- **Reactivity feedback calculated by PARCS**
  - All control rods that are not assigned to a shutdown bank remain fully withdrawn. This is a conservatism in the RES staff analysis
  - The beginning of cycle exposure point free of xenon and samarium was selected as a limiting state because of low Doppler and moderator feedback coefficients
  - Rod motion is delayed by 500 ms to account for solenoid deenergization. Rods take 4.4 seconds to insert

Overview of Methods: mDNBR Calculation BACKUP SLIDE

Non-proprietary

- mDNBR is calculated using a TRACE control system
- Four assemblies are tracked with signal variables that sense flow and heat flux. These are candidate limiting assemblies
- These signals are processed through control blocks that compute the critical heat flux per the applicant's correlation
- A series of control blocks computes the minimum of the DNBR calculated for these four assemblies
- Since TRACE does not track subchannel flow distribution, the mDNBR should not be compared directly, but the calculation should indicate trends in mDNBR during the transient

Heat Flux
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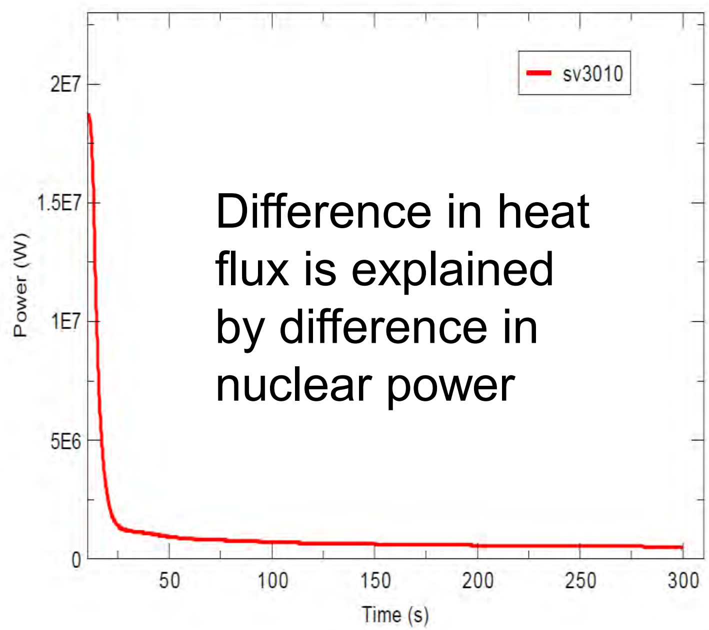
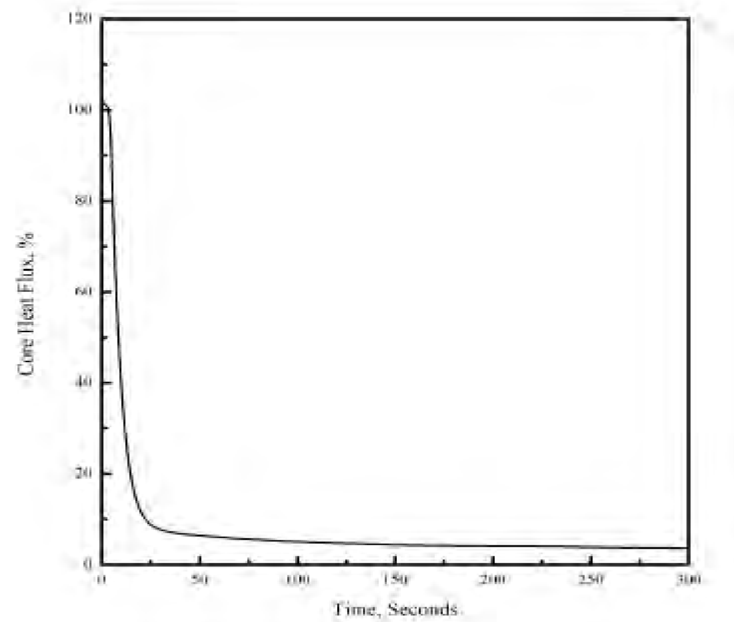
TRACE/PARCS



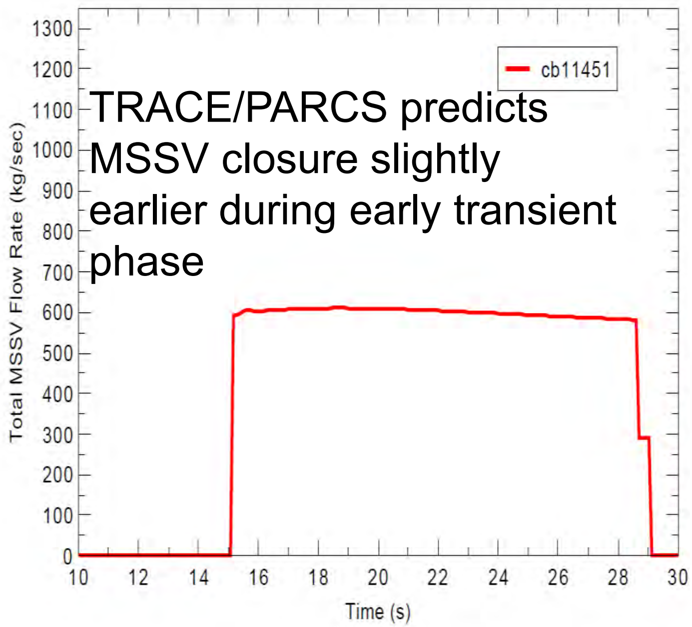
Difference in heat flux is explained by difference in nuclear power

DCD

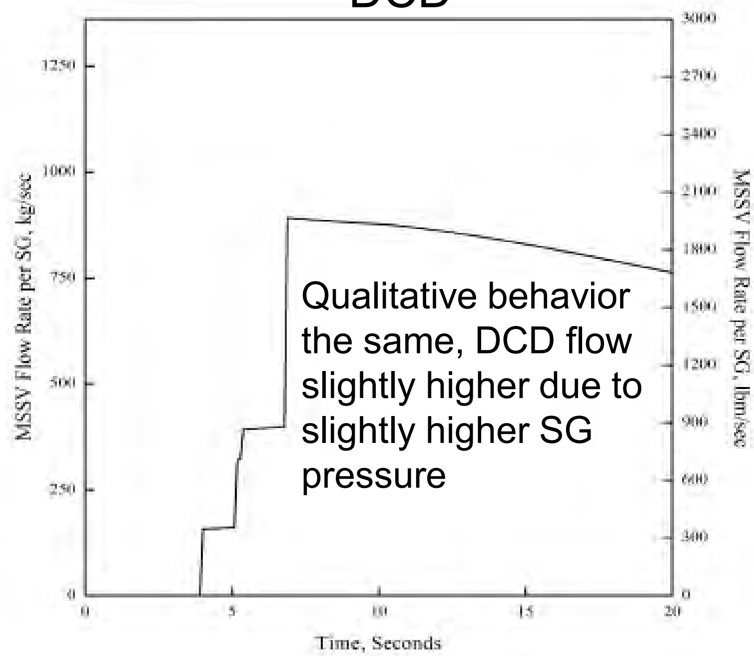


MSSV Flow
BACKUP SLIDE

TRACE/PARCS



DCD



BACKUP SLIDE
15.4.8: Spectrum of Rod Ejection Accidents Staff Initial Cycle Confirmatory Analyses

Non-proprietary

Comparison of "Input" Parameters

DCD Input Assumptions

Case	Power Level (MWt)	β	Ejected Rod Worth ($)	F_q	MTC $(\delta\rho/°C)$	DTC$(\delta\rho/\sqrt{K})$
HZP	1.00	0.00412	1.085	11.49	$9.0x10^{-5}$	-0.00130
20%	796.60		0.901	10.79	$7.2x10^{-5}$	
50%	1991.50		0.626	6.49	$4.5x10^{-5}$	
HFP	4062.66		0.354	4.32	$0.0x10^{-5}$	

TRACE/PARCS Calculated Neutronic Parameters

Core State	Initial Power (%)	Core-Averaged β	Transient Ejected Rod Worth ($)	Transient Maximum F_q
BOC	HZP	0.00687	0.106	2.68
	20%		0.105	2.50
	50%		0.086	2.14
	HFP		0.004	1.83
EOC	HZP	0.00504	0.255	5.70
	20%	0.00500	0.232	3.05
	50%	0.00499	0.196	2.47
	HFP	0.00498	0.036	1.61

BACKUP SLIDE
15.4.8: Spectrum of Rod Ejection Accidents Staff Initial Cycle Confirmatory Analyses

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Comparison of T/H Simulations

DCD Analysis

Case	Maximum Core Power (%)	Maximum Clad Temperature (K)	Maximum Fuel Temperature (K)	Maximum Hot Spot Radial Average Enthalpy (cal/g)	Maximum Fuel Enthalpy Rise (cal/g)
HZP	141.3	620.2	1774.7	75.2	21.7
20%	140.3	859.9	2604.5	113.1	33.1
50%	129.4	842.4	2644.1	120.4	38.4
HFP	156.3	840.4	2763.4	124.7	28.4

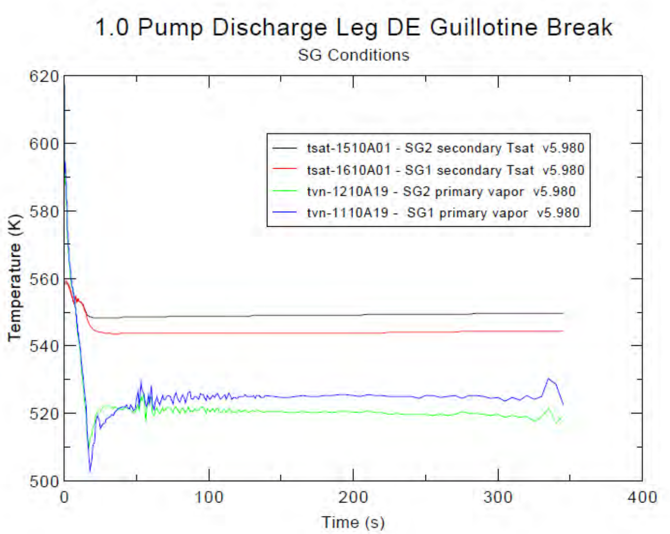
TRACE/PARCS Confirmatory Analysis

Core State	Initial Power (%)	Maximum Core Power (%)	Maximum Clad Temperature (K)	Maximum Fuel Temperature (K)	mDNBR (-)
BOC	HZP	0.03	568.1	568.4	N/A
	20%	25.3	587.1	851.7	8.37
	50%	56.4	605.2	1225.4	3.98
	HFP	102.6	619.2	1895.2	2.34
EOC	HZP	0.076	568.1	569.4	N/A
	20%	27.1	592.7	860.9	7.96
	50%	64.0	614.0	1162.2	3.95
	HFP	106.4	619.4	1686.1	2.60

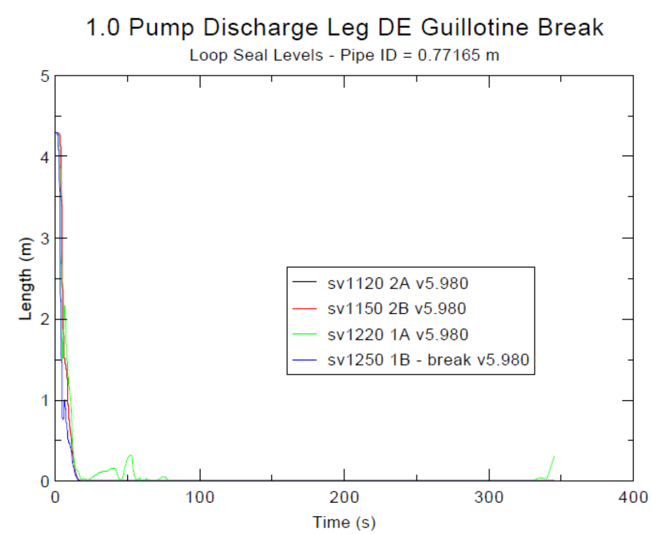
TRACE LBLOCA Loop Seal Reformation and Clearing BACKUP SLIDE

Non-proprietary

- **A second PCT, if present, will be below the initial PCT because**
  - Primary system condensation is not expected because SG primary system vapor temperatures are below SG secondary system saturation temperatures



Steam Generator Temperatures for Best Estimate LBLOCA

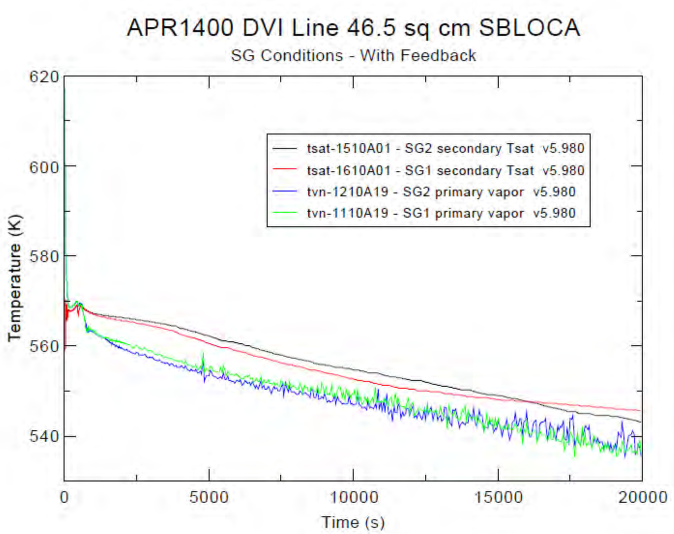


Loop Seal Water Levels for Best Estimate LBLOCA

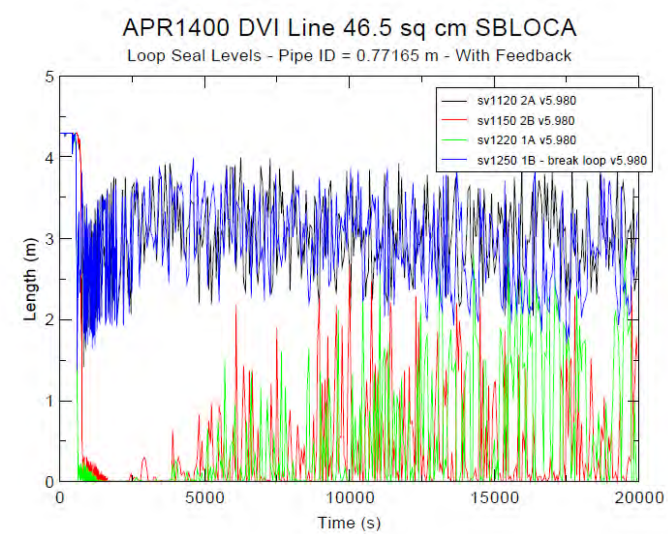
TRACE SBLOCA Loop Seal Reformation and Clearing BACKUP SLIDE

Non-proprietary

- **A secondary PCT, if present, will be below the initial PCT because**
  - Primary system condensation is not expected because SG primary system vapor temperatures at 20,000 s are below SG secondary system saturation temperatures
  - Long term calculations to 20,000 s do not predict a second PCT



Steam Generator Temperatures for 46.5 cm^2 DVI SBLOCA



Loop Seal Water Levels for 46.5 cm^2 DVI SBLOCA

Fuel Thermal Conductivity Degradation Study BACKUP SLIDE

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Non-proprietary

- The fuel thermal conductivity sensitivity analyses demonstrates the importance of burnup in calculating fuel rod temperatures.

Fuel Rod Temperatures Sensitivity Calculation for Additional TRACE LBLOCA Analyses

TRACE Heat Structure	Primary Fuel Rod	Supplemental Rod 1	Supplemental Rod 1	Primary Fuel Rod DCD LBLOCA	Primary Fuel Rod	Supplemental Rod 1	Primary Fuel Rod	Primary Fuel Rod	Supplemental Rod 1
Fuel Rod Description	Core Avg.	Hot Rod	Hot Rod	First Core Avg.	Core Avg.	Hot Rod	Core Avg.	Core Avg.	Hot Rod
GWD/MTU	0.0	0.0[1]	30.0	17.571	20.0	30.0	30.0	40.0	30.0
Steady-State Temperatures (°F)									
Max. Centerline at 2.771 m axial height (coarse axial level 17 of 24)	1692.2	2704.2	3356.0	1995.1	2028.1	3356.0	2162.2	2281.0	3356.0
Min. Centerline at 0.0 m axial height (coarse axial level 1 of 24)	569.7	581.8	591.5	573.0	573.5	591.5	575.7	577.6	591.5
Fuel Rod Centerline Average	1274.9	1910.1	2328.0	1468.5	1492.6	2328.0	1583.9	1664.6	2328.0
Max. Avg. at 2.771 m axial height (axial level 17 of 24)	1155.9	1627.6	1961.9	1302.7	1320.2	1961.9	1386.6	1445.3	1961.9
Min. Avg. at 0.0 m axial height (axial level 1 of 24)	564.2	573.0	576.8	565.1	565.4	576.8	566.4	567.4	576.8
Fuel Rod Average	947.1	1268.5	1461.4	1037.1	1049.0	1461.4	1094.1	1134.0	1461.4
Transient LBLOCA Temperatures (°F)									
PCT	1031.0	1269.1	1396.1	1117.0	1134.6	1444.5	1161.0	1201.8	1464.7

[1] The condition in which the hot rod and average rod are both at 0 GWD/MTU is an unrealistic case which has been run as part of a TCD sensitivity analysis.