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UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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592nd MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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OPEN SESSION

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THURSDAY

MARCH 8, 2012

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ROCKVILLE, MARYLAND

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

COMMITTEE MEMBERS:

J. SAM ARMIJO, Chairman

JOHN W. STETKAR, Vice Chairman

HAROLD B. RAY, Member-at-Large

SAID ABDEL-KHALIK, Member

SANJOY BANERJEE, Member

DENNIS C. BLEY, Member

CHARLES H. BROWN, JR. Member

DANA A. POWERS, Member

JOY REMPE, Member

MICHAEL T. RYAN, Member

STEPHEN P. SCHULTZ, Member

WILLIAM J. SHACK, Member

JOHN D. SIEBER, Member

GORDON R. SKILLMAN, Member

NRC STAFF PRESENT:

KATHY WEAVER, Designated Federal Official

GETACHEW TESFAYE

SHANLAI LU

SHIE-JENG PENG

JEAN-CLAUDE DEHMEL

DAVID RUDLAND

AL CSONTOS

TIM LUPOLD

CHRIS BROWN

ROB TREGONING

ALLEN HOWE

JASON PAIGE

TONY ULSES

PAUL CLIFFORD

BENJAMIN PARKS

ALSO PRESENT:

DARRELL GARDNER
TIM STACK
BRIAN McINTYRE
SAM HOBBS
STEVE KLINE
STEVE MIRSKY
MIKE KILEY
STEVE HALE
MARK AVERETT
CARL O'FARRILL
RUDY GIL

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

P R O C E E D I N G S

8:29 a.m.

CHAIR ARMIJO: Good morning.

The meeting will now come to order. This is the first day of the 592nd Meeting of the Advisory Committee on Reactor Safeguards.

During today's meeting the Committee will consider the following:

Selected Chapters of the Safety Evaluation Reports with Open Items Associated with the US Evolutionary Power Reactor (EPR) Design Certification Application;

2: Source Terms for Small Modular Reactors;

3: Extremely Low Probability of Rupture;

4: Turkey Point Units 3 and 4 Extended Power Uprate Application.

The meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act.

Ms. Kathy Weaver is the Designated Federal Official for the initial portion of the meeting.

There will be a phone bridge line. To preclude interruption of the meeting, the phone will be placed in a listen-in mode during the presentations and Committee discussion.

A transcript of portions of the meeting is being kept and it is requested that the speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

I will begin with an item of current interest. Mr. Charles Brown, Mr. Harold Ray and Dr. Michael Ryan have all been reappointed for another term as Members of the Advisory Committee on Reactor Safeguards. We are fortunate to have them reappointed and to have them accept. Congratulations.

(Applause).

CHAIR ARMIJO: With that, I will now turn the meeting over to Dr. Dana Powers who will lead us through the discussions related to the selected chapters of the SER or the U.S. EPR Design Certification application.

MEMBER POWERS: Thank you, Mr. Chairman.

As most of you know, we have been going the EPR Design Certification, the SER with open items. The general strategy is that we've reviewed the SER with open items and passed that on to a second phase provided we find no barriers to resolution of those open items. This gives us an opportunity to look and identify an open item.

What you're going to see is a fairly eclectic set of chapters presented here today. This strategy that the staff had proposed for the review I was deeply suspicious of because of its rather piecemeal approach, and quite frankly I didn't think it was going to work. And, in fact, it has worked marvelously and it has worked marvelously largely because the staff has enforced a discipline they don't bring us chapters until they feel that there a path to resolution, and the Applicant has been extremely accommodating and has done technically excellent work in what they presented to the Committee. So, it has in fact worked.

But because it's piecemeal, I've asked first of all, that the Applicant in his presentation provide us an overview of the EPR design for reminders of you who are not on the Subcommittee what the general design objectives of this rather interesting reactor are. And it is particularly important that we understand this because these reactors are being constructed worldwide and we have plans to construct at least one or two of them here in the United States, maybe more.

I've also asked that the staff review

their general strategy, and I've asked in their presentation to emphasize the many occasions where they have done completely independent analyses of particular technical aspects. I personally find this when they do independent analyses particularly comforting in their examination of these issues because there's just two eyeballs looking at the same issue, both of them coming up with positive results. You can derive some satisfaction from that.

The upshot if you're looking of raging controversies in these presentation, you're not going to find them. On the other hand, this is an interim part of the overall review. We will get to review this material again when the open items have been closed.

With that introduction, I'm going to ask the Getachew to testify, give us some opening comments.

MR. TESFAYE: Thank you, Dr. Powers.

Good morning everyone. My name is Getachew Tesfaye. I'm the NRC Project Manager for AREVA's U.S. EPR Design Certification Project.

The staff has completed a presentation of the review of the Design Certification, that is the Safety Evaluation Report with open items. So the SER can be a chapter-by-chapter presentation of the Safety Evaluation Report with open items began on November 3, 2009 and concluded on February 23, 2012.

On April 8, 2008 we briefed the ACRS Full Committee on seven chapters that were completed through March 2010. These are Chapters 2, 4, 5, 8, 12 and 17.

On April 21, 2010 we received a letter from the ACRS Full Committee Chairman on these seven chapters. The letter states that ACRS has not identified any issues that merits further discussion.

On May 27, 2010 the staff submitted its reply to ACRS.

Today we will brief the Full Committee on Chapters 6, 7, 11, 13, 15, 16 and 18. As Dr. Powers indicated, this is going to be a very high level briefing with highlights of confirmatory items, confirmatory analysis that we have independently performed to certify our safety findings.

We plan to conclude our Full Committee briefing on the Safety Evaluation Report with open items in May of this year with a briefing on the remaining four chapters, Chapters 3, 9, 14 and 19.

Thank you, Mr. Chairman.

MEMBER POWERS: Thank you.

Just for Members information, the chapters we're looking at are Emergency Safety Features, Digital I&C, Radwaste Management, Conduct of Ops, Transient and Accident Analysis, Tech Specs, Human Factors.

At this point, I will ask our vendor if he wanted to begin the presentation from AREVA.

MR. GARDNER: Certainly. Thank you, Dr. Powers.

Good morning, Mr. Chairman and Members of the Committee.

AREVA's glad to be back here today to--

MEMBER POWERS: Oh, lie to us again. This is a heck of a way to start a presentation, lying to the Committee.

MR. GARDNER: We are continuing to make good progress moving through the Design Certification review with the staff and with the Subcommittee. For this session of the Committee, AREVA will present a summary level overview of the U.S. EPR design as well as selected FSAR chapters as has been described by Dr. Powers.

Our presentation for this Committee is non-proprietary, however should Member questions involve material that AREVA considers proprietary, we would request that the session be closed or deferred until such time as we could have a closed session --

MEMBER POWERS: Or we just stop the questioning.

MR. GARDNER: The AREVA team presenting today will be Tim Stack and Brian McIntyre. We look forward to your interactions today.

Tim?

MR. STACK: Thank you, Darrell.

And as Darrell said, my name is Tim Stack from AREVA and I'll be doing the presentation today.

When we look at this presentation, it largely follows the same style we had in April of 2010 where we gave an overview, which is about a third of the slides. Then we'll do a high level of the chapters. We tried to go through some of the chapters that are easier pretty quickly. Obviously, you will be more than welcome to stop and ask questions. Mr. McIntyre will not be happy if I make that request, but that's okay.

And with that, we'll move on.

As Darrell indicated and as Dr. Powers indicated, first we'll give a quick overview of the design. As a part of that we'll cover the main objectives of what we tried to accomplish with the

EPR, the major design features, the main safety systems, how we protect it from external hazards and what we've included for severe accident mitigation. Then we'll do a chapter overview chapter-by-chapter where we'll basically just cover what are the topics in the chapter and what are the main highlights.

Keep in mind while we go through this, typically most of these ACRS presentations to the Subcommittee are anywhere from 30 to as much as 150 slides. So we're trying to get those down into two or three slides per chapter so we could just cover the main highlights.

With that in mind, as far as the EPR overview and the development of objectives, it is an evolutionary active plant. It's based on existing PWR technology based on what we've learned from our operating plants, our construction experience as well as our R&D.

The main goals were improved economics for our customers and improved safety for our customers and the public as well.

On the safety side we have:

Improvements in design margins compared to the operating fleet;

We have an increased redundancy as well as separation of safety trains;

We've received a lower core damage frequency versus the operating fleet;

We've included severe accident features from ground up as well as design for external hazards from the ground up, and;

Then we've also worked to keep exposure down to the workers and the public.

When you look at the major design features, the easiest way to look at these is pretty much comparing and contrasting them with the operating fleet.

For the nuclear island what we see is that we have a proven four-loop design, PWR design very much like many operating plants in the United States and abroad.

We have a four-train safety system architecture versus a typical two-train architecture in the U.S.

We have a double containment versus typically having a single containment.

We have an in-containment refueling water storage tank versus having an RWST that's located outside containment. We'll cover some of the advantages of that in more detail a little bit later.

We've included severe accident features from the initial design of the plant versus trying to backfit them or backfitting Severe Accident Management Guides like we've done in the operating fleet.

We have separate safety buildings versus having all the safety trains in one nuclear aux building.

We have an advanced digital design for the control room versus having an analog design.

And those are the main features of the nuclear island.

Then we move over to the electrical design. The design shed power to house load versus most operating plants can only tolerate partial load rejections and continue to operate.

We've included four emergency diesel generators versus most operating plants only having two.

We have two smaller diverse station blackout diesel generators and most operating plants only had one if they credit an alternate ac source.

And from the site characteristics standpoint we've designed for airplane crash for both military and commercial on an air track, and we've designed for explosion pressure waves as an external hazard as well.

In general, we feel the EPR reflects the full benefits of the operating experience and all the 21st century requirements.

When you look at the general layout of the NSSS, what you see is that the very conventional 4-loop design related to a reactor vessel in the middle, four steam generators, a pressurizer, four reactor coolant pumps. That portion of it as far as the general layout is very standard.

One of the items that was changed in this compared to the operating fleet was increasing volumes of the primary and secondary to slow down the primary system response and the secondary system respond. We'll cover some of the action times later in the presentation that we've included for operator action times when we cover Chapter 15. But in general, the NSSS is built on our operating system that's coming primarily from the French N4 plants and the German Konvoi units.

The four train architecture, I'll show the picture and we'll explain our N+2 concept as well as some of the physical separation. On one of our later slides I'll do the overall basic layout of the site.

What you see in this is we have our

reactor building here in the center. I go in here to the bottom. This is the fuel building that is here, which is divided into two division. Then there are four safeguards buildings: Safeguards 2, Safeguards 2, Safeguards 3 and Safeguards 4. The main control room is housed in Safeguards 2 and 3. We'll cover this in a later slide well.

What you see is this blue structure is a shielding structure that's covering the blue building, the reactor building and Safeguards 2 and 3. We'll explain the physical protection a little bit more in the next slide.

When we look at our N+2 concept, basically where we are designing is we will postulate a single failure Division 1. We will postulate preventive maintenance in Division 2. We will postulate the initiating event defeats one of these safeguards streams in Division 3, for example, like broken ECCS injection line such that that line is not providing an accident-mitigating feature function. And then finally, we'll go to the fourth division and it will be provided the accident-mitigating function.

In general, we need to energize two out of four divisions for success. And again, that's in not knowing which one has the break in it such that we would have to energize two divisions.

When you look at the main safety systems of the plant --

MEMBER BLEY: But you only need one actually working for success, is that right?

MR. STACK: In most events we only need one. In selected accidents, we need two.

MEMBER BLEY: Okay.

MR. STACK: So, for example, if we look at a steam line break, we will initially align one and we will realign a second operating train to an intact generator.

MEMBER BLEY: Yes.

MR. STACK: But for most accidents, we only need one.

MEMBER BLEY: Okay.

MR. STACK: When we look at the main safety systems in the plant, we have basically for safety injection we have passive accumulators, we have the low-head safety injection system with a combined RHR system in meeting that safety injection. All those are pretty typical for the operating fleet. And we have our in-containment refueling water storage tank.

So, when you look at the alignment of

these you would start with the IRWST. From those sections take it for the medium-head safety injection; that's going to tie in and pump into a cold leg. There's also an accumulator on that discharging into the cold leg.

The low-head safety injection we're also taking a section off the IRWST. It's going through an RHR heat exchanger to remove heat from the fluid. And it's being directed initially back to the cold leg. But one hour into the event we will also realign the discharge of the LHSI pump to provide hot leg injection. That's aimed at suppressing core boiling. So we actually have both alignments that are made up from an ECCS perspective.

What we do not show in here, we also have an extra borating system. The extra borating system is providing highly concentrated boric acid. Basically, it's relied on for tube rupture mitigation and achieving core shutdown requirements.

In addition, in one train we have a non-safety related containment spray or containment heat removal system which is really providing severe accident mitigation such that we are taking a section off the IRWST, we're pumping it through a heat exchanger and back into the containment. And for severe accident mitigation that's providing the long term heat removal.

MEMBER SIEBER: How long were your coolant pump seals be tight without cooling

MR. STACK: For reactor coolant pump seals we have a Stansfield seal system. We have three seals like you typically would on a reactor coolant pump, and then there's a Stansfield seal which provides positive closure. Right now the Stansfield seals are qualified for the SBO durations that we have.

MEMBER SIEBER: Which are? What are those durations?

MR. STACK: Right now our required duration is eight hours. We've done other testing to demonstrate they'll go well beyond eight hours.

MEMBER SIEBER: Okay. Thank you.

MR. STACK: You're welcome.

So, basically that covered the primary side. We'll move over to the secondary side now.

On the secondary side we have four trains which are identical. And what you see in these is we will have emergency feedback delivery from an EFW tank or coolant. We'll take suction from that to an emergency feedwater pump. We will discharge to a steam generator.

Coming on the steam side, we'll have a main steam isolation valve. We're not showing all the details, obviously there's a bypass around it as well.

On the upstream side we will have a main stream relief frame which provides 50 percent of the heat removal, and then we will have two 25 percent spring-loaded safeties.

And when we look at the main steam relief train, it's somewhat unique for the United States. It's used in the Konvoi units. It's providing safety grade depressurization. It's relied as a part of the LOCA accident mitigation as well as the tube rupture accident mitigation. And it's also allowing you to do safety grade cold safe shutdown.

So that's basically the steam side. In addition on this what we see is our four emergency feedwater pumps are all motored waters, they're all backed by the off-site grid, as well as the EDGs. Two of four of those are also backed by Station Blackout diesel generators. And then we show the inner connecting -- there's also inner connecting piping on the suction side and discharge side to allow us to interconnect the water sources as well as the discharge.

MEMBER SIEBER: So you used the main

feedwater pumps also as auxiliary feedwater pumps

MR. STACK: No.

MEMBER SIEBER: They're separate?

MR. STACK: When you look at the pumps in general on the plant, we have four motor-drive main feedwater pumps, which are providing normal main feedwater. For startup and shutdown we have a separate starter feedwater pump such that we're not relying on emergency feedwater for startup and shutdown. And then we have the four emergency feedback water pumps.

MEMBER SIEBER: Okay.

MEMBER SKILLMAN: Tim, why are the emergency feedwater pumps all motor-driver versus some turbine-driven?

MR. STACK: The main reason is really twofold. In many of the operating plants, as you well know, we've had problems with the turbine-drive pumps. They've tripped on moisture induction.

MEMBER SKILLMAN: Yes.

MR. STACK: We've also had problems with high-energy line breaks in the buildings that they're resident in. So, as soon as we out a turbine-drive pump in one of our safeguards buildings, we immediately have to deal with all the high-energy line breaks with them.

We've also done reliability analysis and risk study to look at what would happen if we changed some of the pumps, let's say two out of four, from motor-driven to turbine-driven. What we determined is a result of that was we really didn't get any significant safety benefit.

We also knew that we were going to have more reliability problems with the turbine-driven pumps than the motor-driven pumps. So that was our main rationale for let's stick with the motor-driven pumps and let's get away from the turbine-driven pumps, which again have been problematic.

MEMBER SKILLMAN: How are you assuring that common cause isn't an issue here, because that's the reason people have a diverse kind of pump?

MR. STACK: In general on these, we would look at if you were looking at common cause there with the emergency feedwater pumps, this whole method of secondary heat removal is backed by a primary feed and bleed as diverse means.

MEMBER BLEY: Okay.

MEMBER SKILLMAN: Thank you, Tim.

MR. STACK: And then let's go and look at external hazard and protection and shielding. And what we have in the design, shown in brown, is an inner post-tension concrete containment building with a steel liner. We also have separately an outer wall that's reinforced concrete. The outer wall is providing airplane crush protection as well as protection from an explosion pressure wave.

Looking at the annulus between these two, we have a filter vented annulus for dose control, and that's the general method that we're using for dose management and as well as for protection.

MEMBER BLEY: How big is that annular area?

MR. STACK: The annulus --

MEMBER BLEY: The width of it?

MR. STACK: I'm going back and trying to see if I have the exact dimension.

MEMBER BLEY: It wasn't Sandra Sloan --

MR. STACK: It's about a Sandra Sloan.

MEMBER BLEY: It's pretty good size.

MR. STACK: It's pretty good size.

I was looking if I have my dimension on the --

CHAIR ARMIJO: You probably could play a handball game in there.

MEMBER BLEY: Six or seven feet, give or take.

MR. STACK: So now we're going to talk a little bit more about our -- so that was showing you here that we have the blue shield building is providing protection. And I showed in my early slide where we provide that protection. First that we do, quicker review of the site layout, and this is really covering the nuclear island the turbine island.

And let's get oriented first in the center. We have the reactor building, below it we have the fuel building. Again, we have Safeguards 1, 2, 3 and 4. Again, in Safeguards 2 and 3 that's where the main control room is.

Immediately adjacent to it we have an access building.

Down here we have an nuclear auxiliary building and a radwaste building.

Additionally, we have two diesel buildings which house, this one on the side diesels 1 and 2 and on this side 3 and 4. And then down at the bottom we have ultimate heat sink towers here, 1 and 2 and then the top 3 and 4.

Then above it we're showing the turbine building and adjacent to the turbine building we have the switchgear building.

MEMBER BANERJEE: What's these alternative sink towers?

MR. STACK: Mechanical draft cooling towers.

MEMBER BLEY: I'm getting out of the safety buildings, but what's in the auxiliary building?

MR. STACK: Basically anything else that's not relied on for basic -- you know, and a lot of these we will go and have other auxiliary for the plant going back -- I'm drawing a blank right off the top of my head. In general, in a lot of these you'll find some equipment related to the CVCS that's not safety-related and it's not required to be housed in one the safety-related buildings. I can look that up real quick, though.

MEMBER BANERJEE: How much heat do each of these cooling towers remove?

MR. STACK: When you're sitting and looking at their -- their normal duty is based on the limits of either the design basis duty or the normal heat rejection for the train. And they have to accommodate both.

MEMBER BANERJEE: How much is that?

MR. STACK: I would have to go and look it up. My recollection is about --

MEMBER BANERJEE: Roughly.

MR. STACK: -- roughly 300 million BTUs is what my memory tells me. But -- but what I will say is we will be happy to look that number up.

MEMBER BANERJEE: Could you use per second or something or hour, whatever?

MR. STACK: BTUs per hour. Thank you.

MEMBER BANERJEE: Okay. But the standard is certain number of megawtts.

MR. STACK: Yes, it does. I haven't committed that one to memory.

MEMBER BANERJEE: Okay. Okay.

MR. STACK: Okay.

MEMBER BANERJEE: And how much power do they draw?

MR. STACK: As far as the -- the power consumption is included in the EDG sizing because they're carried by the EDGs.

MEMBER BANERJEE: But they wouldn't fall short without the EDGs?

MR. STACK: That's correct.

MEMBER SIEBER: Are your diesels radiator cooled or service water cooled?

MR. STACK: The EDGs are water cooled.

MEMBER SIEBER: Service water cooled?

Well, yes, service cooled to them.

MEMBER SIEBER: Okay.

MR. STACK: And the SBO diesels are air cooled.

MEMBER SIEBER: Radiators?

MR. STACK: Yes.

MEMBER SIEBER: Okay.

MR. STACK: So we looked at this, so I'll come back to our external hazard protection. What you see is the shield building shown on the previous slide is really covering the reactor building, the fuel building and Safeguards 2 and 3, and again that's where the main control room is. So we're physically shielding those buildings and we provide physical separation for the buildings that are shown in grey such that at most we will damage two trains if you were to have an airplane crash that was to hit these diesels buildings or hit these cooling towers. The other ones on the opposite side will be physically protected by separation.

So that's the overall strategy for external hazard protection.

Severe accident mitigation, we're just about to the end of our background. Our design features considered the phenomena that were considered in SECY-90-016 and SECY-93-087. And when we look at the main features in this, there's a high pressure core melt depressurization system which is shown up in here where there are two trains with two valves in each train where we can depressurize the primary system.

We also have an ex-vessel stabilization conditioning and cooling system -- lots of words. The quantity that's held up in the discharge channel from underneath the reactor vessel first and then it's discharged into its cooling channel. In that cooling channel initially for the first 12 hours it gets cooling from the IRWST and then by that point in time we would initiate the active containment spray system that it takes up from there and provide cooling back to that location as well.

And then on the hydrogen side we've included passive autocatalytic recombiners for hydrogen management versus igniters.

And those are the main features for severe accident mitigation.

MEMBER BANERJEE: Were are the recombiner space? They are right at the top or all over?

MR. STACK: They are located throughout the containment. If memory serves me, I believe there are 47 of them, give or take.

MEMBER BANERJEE: Okay.

VICE CHAIR STETKAR: Tim, I was looking at something else. The second bullet on the passive ex-vessel melt stabilization, the last I recall is the IRWST lines to the core -- whatever you call it, core spreading compartments are normally isolated by normally closed motor operated de-energized valves that the operators have to energize and open to then expose the rupture discs. Is that still part of the design or have you --

MR. STACK: It is.

VICE CHAIR STETKAR: It is? Okay. So it's not truly passive in the sense that somebody has to line it up first, actively open the isolation valves such that then it becomes purely passive?

MR. STACK: And when you look at the decision making and you have a conflict between core cooling with the IRWST as well as severe accident, and it's a balance between the two.

VICE CHAIR STETKAR: Yes, yes. The word "passive" has a lot of little hooks in it --

MR. STACK: I understand. Semi-passive is probably great.

Okay. So that basically concludes the

background section. And we'll now move into the overview of the chapters themselves.

And just by way of background, again we mention that it's an evolutionary active plant. We've used the proven analytical methodologies. We followed the SRP as well as Reg. Guide 1.206. We've minimized the exemptions and exceptions, and we haven't applied RTNSS as a part of this design or as part of the licensing process for the U.S. EPR..

VICE CHAIR STETKAR: But you do have a Design Reliability Assurance Program?

MR. STACK: Yes.

VICE CHAIR STETKAR: Okay.

MR. STACK: So now we'll start moving into the guts of the chapters, and first Chapter 6: Engineered Safety Features. And again, try to cover the highlights in these. You're all familiar with what the topics are. The only thing that I would mention here that's extra is an extra borating system is included in here, which you typically wouldn't find.

As far as the focus on the engineered safeguards, really I'm just going to predominately look at the containment systems and then the ECCS systems.

And for the containment I mentioned

previously, again, it's a post-tension concrete containment with a steel liner. We mentioned having our reinforced concrete shield wall.

It's a large dry containment when you look at it, 2.8 million cubic feet, but there are some that are bigger, like Bellefonte that are large dry containment. Some that are somewhat smaller, Calloway. But it's still a large dry containment. Much, much bigger than most of the operating fleet.

Perspective to that, the design pressure is 62 pounds gauge.

In the bottom of this we have a 500,000 gallon RWST, give you an idea of the magnitude of it.

When you look at the design and we have a two-zone containment, the area in brown is an equipment area that's separated out and basically effectively compartmentalized and isolated with unit net power. And then there's a service area outside of that. Those two zones of the containment are innerconnected by what we refer as a CONVECT system. And what you have is basically rupture flows at the top of, like the steam generator compartments here and then there'll be dampers in the lower sections of these that get actuated such that on a hydrogen line break we convert the two-zone containment into one-zone containment.

We mentioned previously the passive hydrogen reduction system with the recombiners and then the filtered, vented annulus for the dose control.

Of note in this is that the U.S. EPR does not rely on safety-related containment spray or containment fan coolers for heat removal.

MEMBER SIEBER: Does the containment have an equipment hatch?

MR. STACK: Yes.

MEMBER SIEBER: Is it big enough to take a steam generator tube assembly?

MR. STACK: It's big enough to take a steam generator.

MEMBER SIEBER: Oh, it's that big? Okay. So you don't anticipate that some time in the future extra openings would be --

MR. STACK: For the containment? No.

MEMBER SIEBER: Okay. That's turned out to be a problem in post-containment.

MR. STACK: Yes. The containment hatch itself is -- I'm not sure the exact dimension, but it's over 20 feet. It is a large penetration.

MEMBER SIEBER: So it will even take the

steam separator equipment?

MR. STACK: Yes.

MEMBER SIEBER: Okay. Thank you.

MR. STACK: On the ECCS side, so that covered the containment side and we move over to --

MEMBER SKILLMAN: Tim, before you jump to slide 15, on 14 no fan coolers or containment sprays. What's the short answer? Do you simply absorb enough heat with the masses so as to keep the post-LOCA pressure below 62?

MR. STACK: The short answer is basically what you're doing is the steaming from the core is going to be collecting on the surfaces whether they be concrete or steel inside the containment. And initially you're removing heat from the sump, but when we convert from cold leg injection to hot leg injection, we suppress core boiling. And it's really the combination of the ECCS providing hot leg and cold leg injection and stopping core boiling and then removing the balance of the heat from the sump is a success of how you can eliminate the containment spray as a safety-related system.

MEMBER SKILLMAN: I'm just intrigued by the idea that you do not have boxcar fans or sprays and you've intentionally increased the size of the components to get fluid mass greater, so you've got greater mass with that energy and you're still maintaining the containment pressure.

MR. STACK: Right. Just to clarify that, we do have fan coolers and we have non-safety-related fan coolers and we have the non-safety-related severe accident spray. Okay. The equipment exists, it's not safety-related.

MEMBER SKILLMAN: What you're saying is it's not credited for the severe accident?

MR. STACK: Yes, sir.

MEMBER SKILLMAN: Understand. Thank you.

MR. STACK: On the ECCS side, previously we mentioned generally out of the safety injections, additionally from the ECCS perspective we are crediting, and I mentioned the MSRTs, Main Steam Relief Train previously, we are crediting a partial cool down of the steam generators to lower secondary pressure and lower primary pressure as a part of the safety injection scheme. And that's lowering primary pressure below the shutoff head of the medium head safety injection pumps. And that's relied on as part of the LOCA mitigation.

And then relative to the IRWST, again it's providing a single source of water. It eliminates the need to switch over from recirculation mode from outside containment to inside containment like you would normally have. It also supplies sufficient static head that you don't have to piggyback either the medium head safety injection plans for this plant or high head safety plans like you normally do on the operating fleet.

It's providing normal functionality for filling the cavity, what have you, and it's also providing flooding for the severe accident mitigation.

And again, the list point on here is going back to the containment heat removal relative to manual reassignment of the LHSI at 60 minutes to suppress core boiling.

MEMBER SCHULTZ: Tim, in this regard with regard to ECCS capabilities, is that a two-train requirement or a one-train requirement to respond to the sizes of LOCA that you might have?

MR. STACK: As far as these safety injection systems are all working.

MEMBER SCHULTZ: How many do you need?

MR. STACK: Oh, how many do you need on these?

MEMBER SCHULTZ: That's right.

MR. STACK: Okay. From the standpoint of

ECCS you need one for success to provide core cooling. However, I'll clarify with this: I mentioned before that we're energizing two. So for example, back here when we look at a case like this, we can have -- let's say I was to break this cold leg right here, and this was an active train and this low head safety injection pump was pumping into the break such that it's not providing any core cooling. It may not be providing core cooling, but it is providing heat removal from the sump because it's energized and it's active. And we are crediting that function because it's not impaired by the accident. Okay?

MEMBER SCHULTZ: Understand. Thank you.

MR. STACK: Okay. So basically, that covers the ECCS side. Now we're going to move into everybody's favorite topic, I&C.

MEMBER POWERS: No, no, no. It is not everybody's favorite topic. It is only Charlie's favorite topic. Even he is beginning to waver a little bit on that.

MEMBER BROWN: My favorite topic?

MEMBER POWERS: Yes.

MEMBER SIEBER: It's all analog, so don't worry about it.

MR. STACK: So, in Chapter 7 I won't spend time covering, the topics are for the Reg Guide as far as the content.

In trying to synthesize Chapter 7 down into a reasonable number of slides from an overview is always a challenge. What I tried to do here is avoiding the desire to show a picture, I decided to use a table as a more simplified way to convey what the design includes. And what we have, an I'll just cover the main highlights of the I&C design.

And the way this table is laid out, it's broken out into you can see a system name, what it's basic function, the safety class, the number of channels and the technology.

We'll first cover what's going on with this as far as the safety system, then we'll cover what's backing up the safety systems as a diverse actuation system. Then we'll move on to the non-safety side of this.

So when you look at the safety side, when you start with the human-machine interface that starts with our safety information and control system, it's effectively providing backup HMI. The main HMI that's going to be done in this plant is with the normal control interface. But basically that our SICS system is providing the safety-grade means for control.

Again, it's safety-related. And all the safety systems -- as a matter of fact in all the systems that are shown here other than the last one are a process automation of all 4-channel.

One of the other things you'll in all the safety systems, they all use our TELEPERM TXS platform as the family of equipment, which is the family of equipment installed at Oconee that was used in the RPS and ESFAS replacement.

When we look at the safety systems, though, we start with the human-machine interface with our SICS system.

We have a protection system which is providing safety grade reactor tip and ESFAS actuation.

We have a safety grade safety automation system which providing safety grade control.

Then we move to the input side on the Signal Conditioning System. Signal Conditional Distribution System.

And then we move to the output side and we have our Priority Actuation Control System.

So basically, that's the general structure of the safety-related I&C. Again, each of these is built on the TXS platform.

A few things to note in this as well. You'll see that most of this is microprocessor-based, certain parts are not. The PACS module in particular is going to be a PLD that's going to be a 100 percent combinatorially tested. And we'll cover a few more comments on that in the next slide.

So that's basically the safety system. You have a human-machine interface, safety grade actuation, safety grade control, an input side and output side.

To backup that safety system, we have a Diverse Actuation System, our DAS system, which is going to be supplemental grade and quality. It is not going to be implemented in a microprocessor-based system as far as the technology is concerned.

Then we move to our non-safety side where we have our Process Information Control System. Again, this is the primary interface that the operators would routinely use to control the plant. And that would be with an industrial platform.

Then we have a Reactor Control Surveillance and Limitation System which is controlling reactivity. And then finally our Main Process Automation System which is really covering the other process system in the plant.

And the other comment there on that is, again, that's four channel on the nuclear island, two channel on the turbine island.

So, that's the general architecture of the overall I&C.

MEMBER BLEY: Tim, as you going to tell us more or will you tell us more about what you mean by 100 percent combinatorial tested?

MR. STACK: I will cover an item on this slide. Just briefly we have made a commitment to do full 100 percent testing of that PLD. That's really aimed at ensuring that we do not have common cause failures of that model.

MEMBER BLEY: Are you testing signals beyond the range you're expecting them to be when you talk about 100 percent combinatorial testing?

MR. STACK: The best way to answer your question in this is probably to direct you to our technical report that lays out our test plan that we've submitted on how we're going to do our testing.

MEMBER BLEY: Okay.

MEMBER BROWN: Can I provide an observation.

MEMBER BLEY: Yes, Charlie. I'd appreciate it.

MEMBER BROWN: I wish you had included their magic figure of their system just for the eye candy and the eye strain and the brain strain that you undergo. But we did go through it very extensively.

And you can correct me if I'm wrong on this, but PLDs are roughly combination combinatorial type discrete logic functions. So you have a very -- a well known input and process state. It's not like a microprocessor where data is being thrown all over the place and executing routines and maybe it'll get them and maybe it won't and all that stuff. It's like we built computers in the '50s with discrete components. Fundamentally, that's fairly simplified.

So, you can pretty well define what the input states are and what they will do in terms of the response in terms of the testing. So that was virtually impossible doing on a software-based system. When they talk about microprocessor PLD and then in nonmicroprocessor, the old nonmicroprocessor system is really blacksmith technology. It's analog type setups for that particular setup.

And if you look at the PLD agent for this before, I mean AP -- was it 85, there's one of the other projects like FPGA, they're kind of like field-programmable data, but they're all different but they're all in this logic device family.

So, I don't think that's a big problem. I came through that. This is a very integrated system of all the ones we've looked at, more integrated in my opinion than the other plants. That is why I wished we had the big -- the one picture that we had a discussion of in the informal meeting, and then they presented in the Subcommittee meeting. Just to get an idea.

VICE CHAIR STETKAR: My simple response to your question is no.

MEMBER BLEY: That's what I figured. Okay.

VICE CHAIR STETKAR: But that's just me.

MEMBER BROWN: It probably doesn't I mean, fundamentally the combination of logic --

MEMBER BLEY: The phrase, it caught my ear. I'm going to go look at that test report.

MEMBER BROWN: You can test all the different logic functions --

MEMBER BLEY: Yes.

MEMBER BROWN: But you cannot test across every exceeding -- exceeding every range of every function that you may put in. That's the way I would define it.

MEMBER SIEBER: And there is adequate separation of channels?

MEMBER BROWN: Yes, but because of the nature of it you're fundamentally really not going to confuse another channel with something going on at one of the other ones. So, anyway --

MEMBER SIEBER: That sounds pretty good.

MEMBER BROWN: -- that's a complicated observation.

Do you happen to have backup slide with that really eye strain --

MR. STACK: You didn't like that. You didn't like that.

MEMBER BROWN: No. I just wanted the Committee to be able to understand --

MR. STACK: You told us no.

MEMBER BROWN: I didn't tell you no. Oh, I'm sorry.

MEMBER REMPE: And he complained about a slide and he likes it.

MEMBER BROWN: No, I didn't say I liked it. Okay. It was inadequate to describe the functionality and how the system operates.

MR. STACK: Okay.

MEMBER BROWN: But it describe the

complexity of the system that they're proposing.

MR. STACK: Okay. So the truth is, is this mechanical engineer is not going to be explaining that.

MEMBER BROWN: I didn't think so. That's why I made an observation.

MR. STACK: Okay. So, main features of the Distributed Control System are the menu that's online: Self-testing, a very robust architecture which is aimed at providing defense-in-depth, diversity, redundancy, independence and priority setting.

There is a high degree of automation which is aimed at reducing operator burden.

We just had an interesting discussion on the PACS modules and the combinatorial testing, and we will get you that test report number that's laying out how the testing will be done.

We also mentioned that we have in our Diverse Actuation System, that will be a nonmicroprocessor-based system which is aimed at avoiding common cause failures in the protection system software.

Then from the communication side from our protection system and our SAS system they're unidirectional to the non-safety systems. One way directed communication that's isolated. From an interdivisional communication standpoint within the PS divisions we only rely on that for voting logic. And those are isolated with optical communication.

And then on the SAS side we rely on interdivisional communication when necessary to perform a safety function. One of the examples I've listed here is looking at CCW interlocks for our RCP thermal barrier coolers. And basically what you're seeing is you have four reactor cooling pumps each of them needs cooling to the thermal barriers, and that's the safety-related means to the thermal barriers are cooled. Normally, you only have one train of component cooling water providing cooling to that. And if you lost the one train, you're going to swap to an alternate which requires you to know the status of when one's lost and you can pick up the other. So, you need information on both trains in order to provide the safety function.

And then finally, we provide a service unit for maintenance on the protection system in SAS. That is not continuously connected and we provide switched that will preclude you from connecting that to more than one division at a time.

MEMBER BROWN: Before you leave --

MR. STACK: Okay.

MEMBER BROWN: One other comment for the Committee. Fundamental, if you look at this system it's similar to the systems in some of the other advanced reactor designs. It is fundamentally -- it uses microprocessors for all its voting logic which puts it in a state of being fundamentally not independent of each other. You can contaminate, you can lock them up and the one major open item is the application of their hardware watchdog timer. How does it execute? How does it operate? Is it truly independent because there is some touch points with some software. So the issues have been brought up and that's part of an open item that will be resolved. This is still an open item within the overall SER evaluation, so just to let you know that the other point that we will probably get into is the cyber security aspects of the way the data is transmitted to the main control room and then off to the corporate function. There's effectively one place that could be broken into and then destroy all the data going to the main control room. That's my opinion, by the way. I'm sure they do not share that opinion, but I will. It's going to be subject to some more discussion at a later date.

And they're not happy with that. We kind of ignored it the last time.

MR. STACK: Okay. The last part about the I&C for the EPR that is actually is more unique as well, and this is more as part of the I&C and the overall core protection. We utilize self-powered neutron detectors within the core in lieu ex-core detectors with selected types of trips. And what you see in here, and we talk about 72 of these and what the benefits are and why did we do this. And what you see is there are 12 strings of detectors, each one of these at this axial plane is showing one of the strings of detectors, and there's six axial locations. So you have six by 12 or 72 of them that are monitoring flux in the core, locally monitoring in the core. We are using that for our low DNBR trips and our high linear power density trips.

In order to make this technology work, and part of the reason behind that is this is a large core. It's 241 fuel assemblies and we get much better data and knowledge about what the core is doing internally than we would by ex-core trips.

So what we have done is for these two types of trips we are relying on all 72 of the SPND inputs to each of the channels to take the -- each of the four protection system channels to take their protective action. The benefits are from IEEE 603 it provides more direct measurement of the neutron flux and it reduces our uncertainties with our ex-core based trips.

A little cartoon generally showing how it works. But in general, what you're looking at making each of these look like a surrogate of the hot channel and then making trips based on that such that we have a very accurate trip based on local conditions in the core.

MEMBER ABDEL-KHALIK: How does the minimum DNBR algorithm account for potential variability in flow due to lower-plenum anomaly?

MR. STACK: Relative to this, I would have to defer to our fuels people. If we would like to have follow-up discussion with them on that, we can. I can't speak to the details of the uncertainties and how they've accommodated the flow uncertainties.

MEMBER ABDEL-KHALIK: So the blue block on the left, does that refer to total core flow?

MR. STACK: Yes.

MEMBER ABDEL-KHALIK: So you have no idea how locally within a specific channel the flow rate might be?

MR. STACK: No.

MEMBER ABDEL-KHALIK: And how you account for that potential variability then?

MR. STACK: When you look at some of the variability even on flowing to the core, there's first of a kind of testing in the core flow and the distribution to the inlet to the core. So we have a lot of first of a kind testing that's showing the -- based on here's the number of flow rates we get, here's the flow distribution we except from first of a kind testing that we do when we design a distribution valve.

MEMBER ABDEL-KHALIK: But if goes into the algorithm, it's the total flow. That doesn't account for that variability.

MR. STACK: And what I'm trying to say in this, I would need to go back and see how our people accommodated the variation flow when they did their calculations. I don't know that off the top of my head.

There is a topical report that was submitted --

MEMBER ABDEL-KHALIK: We would appreciate a follow-up --

MR. STACK: Okay. We'll get it.

MEMBER ABDEL-KHALIK: Okay. Thank you.

MEMBER SKILLMAN: Tim, let me ask, you've identified two trips that are based on the SPND. Are the high flux, the high rate of flux and common trips triggered by your ex-core detectors?

MR. STACK: The other trips don't rely on this. These are the only ones that rely on this.

MEMBER SKILLMAN: I understand that. So what I'm asking is do the other flux-related trips depend on the ex-core detectors?

MR. STACK: They do. They're not relying on this.

MEMBER SKILLMAN: Thanks.

MR. STACK: Okay.

MEMBER BROWN: Flip back just one moment just to clarify something. Try to trigger my memory. The 72 SPND signals are needed in all four, that means all 72 detectors feed all four channels?

MR. STACK: That's correct.

MEMBER BROWN: Okay. However, I'm trying to remember the last discussion. You can have a fair number of failures of individual detectors --

MR. STACK: Yes.

MEMBER BROWN: -- before you -- and I've

forgotten how many that is.

MR. STACK: Six.

MEMBER BROWN: Okay. Throughout the entire 72, is that correct?

MR. STACK: Yes. So what happens on the failure of the detectors, we have failure of the detectors. When a detector failure is sensed, we use more conservative trip setpoints. And we've treated undetected failures as a part of the safety analysis.

MR. GARDNER: That's an open item, the--

MEMBER BROWN: Yes, I thought it was. I just couldn't remember from the discussion.

MR. GARDNER: But the detector is part of the methodology where it results in a more conservative setpoint.

MEMBER BROWN: But then there's the other issue is then what about the undetected ones that you may not know about?

MR. STACK: And that's the one where it's going to be treated as an open item and it will be treated as a part of the safety analysis.

MEMBER BROWN: Okay. And the other point, I guess I want to make sure, at least I understand again, is that out of this 12 strings, six detectors in each, those are not summed, they're individual signals that come out and they're processed individually by the instrumentation, isn't that correct?

MR. STACK: There's an algorithm in the topical --

MEMBER BROWN: Yes, I understand. But each detector is sensed --

MR. STACK: Yes.

MEMBER BROWN: -- it's not like the signals come out of six of them, they're all combined and then they go off to the processing?

MR. STACK: Yes.

MEMBER BROWN: Okay. I just wanted to make sure I understand that.

MEMBER REMPE: So this type of detector system, is that going to be used it the Finland plant?

MR. STACK: It is.

MEMBER REMPE: It is?

MR. STACK: Yes, this is used in all the EPRs.

MEMBER REMPE: Okay. And this will be the first time in the EPR that it will have been used, right?

MR. STACK: In the U.S.

MEMBER BROWN: In the U.S.

MEMBER REMPE: And overseas, have you used it in the --

MR. STACK: The first application of this will be in Finland at OL3.

MEMBER REMPE: Okay.

VICE CHAIR STETKAR: I think the staff wanted to make a comment.

MR. LU: This is Shanlai Lu.

And to answer your question about a CHF correlation and your question regarding the core flow. And AREVA did have a design below the lower plate there is flow distribution flow structure to allow uniform flow going through the core. And they had a scaled test. And so staff reviewed that and that is the reason we can rely on single bounded correlation plus the uniform flow assumptions with the core to perform in the application

All right. That part of it, we already reviewed that part.

MR. STACK: Okay. With that, we will wrap up I&C. Want to continue on I&C?

MEMBER BROWN: I'd just want to --

MEMBER POWERS: I don't.

MR. STACK: Okay. We'll move on into Chapter 11, Radwaste Management.

Briefly through this, the main highlights on our radwaste management are that you see various technologies used with it for both processing with evaporators, centrifuge and demineralizers. That's aimed at European experience with the different technologies and providing flexibility.

We also see solid waste volume reduction as a part of the basic design in the radwaste systems as well as main steam rad monitors that are used in the steam generator tube rupture mitigation.

Overall what you see is the liquid gas and solid waste meet 109 CFR 20 Appendix B requirements Pretty standard as far as this is concerned.

MEMBER RYAN: Can you put some kind of an estimate on volume or the reduction you mentioned a minute ago on this?

MR. STACK: I cannot off the top of my head, but we can get you that answer.

MEMBER RYAN: Okay. That's be great.

MR. STACK: And that's really, again, the radwaste management in this is pretty typical.

Next, another easy chapter, Chapter 13: Conduct of Operations. Again, the topics are standard topics in this. In general when you look at Chapter 13 most of the conduct of operations is for the conduct of the COL applicant.

Another item I will mention in here is that the emergency operating procedures will be symptom-based versus event-based as far as the strategy that's going to be used in those.

Moving on to Chapter 15. Another favorite topic.

MEMBER BLEY: Are you going to have the procedures available for the control room electronically?

MR. STACK: Yes.

MEMBER BLEY: Okay. Any special features that you're planning about those, or just kind of like the paper procedures would be?

MR. STACK: There has been much discussion on the content of how that is going to be.

Darrell, I will ask you --

MR. GARDNER: I was going to say, I don't think it's part of the Design Certification so there's a lot of talk about how it would be implemented in the actual plant. But as part of the Design Certification other than the concept of computer-based procedures, I don't believe we specifically provided any detail. That would be, you know, at the choosing at the COL applicant.

MEMBER BLEY: Okay. So you're not doing that as part --

MR. GARDNER: No. That's part of Design Certification.

MEMBER BLEY: Do you have a current mock-up of your control room that your COLs can take a look at?

MR. STACK: The COLs can take a look at some of that. The other thing that they can do is we are building plants in Finland, in France and in China.

MEMBER BLEY: Right.

MR. STACK: And they have the opportunity to review those designs and their progress in those designs.

MEMBER BLEY: Okay.

MR. STACK: Moving into Chapter 15: The Transient and Accident Analyses, standard set of topics that are to be covered.

Highlights of this. Again, what we've done is are design features relative to U.S. EPR similar to existing PWRS? Some of the features that are highlights, though. First, the front line safety systems. Again, what's generally different is we're taking credit for the fact that we have four trains of safety systems for safety injection, RHR, emergency feedwater and the main steam relief trains. I'll mention again, we do have selected two train systems. And to probably answer your question why do you have some two train systems. Simplest example if you were to think about containment isolation or feedwater isolation, it doesn't make sense to have four isolation valves on every penetration. So it was clear that we were going to get some places where we wanted to use selected two train systems. And you look at the two train systems what you find is that they are not impaired by an accident initiator that they are required to mitigate.

So, for example, the annulus ventilation, steam line breaks, feedline breaks, LOCAs cannot impair them directly. And on the extra borating system, it's not relied on for LOCA mitigation.

Moving to the in core fueling on the storage tank, we showed that one previously. It's part of what the main features are, it's being the source of the ECCS. Again, no switch over of any type required.

Operator action times. The operation action times are for the Chapter 15 accidents are 30 minutes from the main control room, 60 minutes for actions outside the main control room that are required.

VICE CHAIR STETKAR: Tim, if I recall your remote shutdown system is not designed for operator control for design basis accident, is that correct?

MR. STACK: That's correct.

VICE CHAIR STETKAR: Okay.

MR. STACK: It's when you're using --

VICE CHAIR STETKAR: It's the ability to maintain hot shutdown for transients?

MR. STACK: Well, you're using it also for mitigation of fires, taking the plant cold.

VICE CHAIR STETKAR: Do you take it to cold from that?

MR. STACK: Yes.

VICE CHAIR STETKAR: Okay. But not for a LOCA?

MR. STACK: It's not relied on for that.

We mentioned previously the automatic partial cool down of the steam generators, this is with the main steam relief trains on the safety injection signal. Again, that's being used as part of LOCA mitigation.

We have safety-related warm and high activity in the steam lines for tube rupture mitigation.

VICE CHAIR STETKAR: For other Members who may or may not be familiar with European versions of this design, this is a bit different from the European version because this is simply an alarm function and not an automatic actuation function as it is in European versions of this design.

MR. STACK: And part of what's happening with that is the number of tubes you're breaking and how fast things go down for the U.S. versus Europe.

VICE CHAIR STETKAR: The number of tubes you have to assume are breaking in your licensing analysis?

MR. STACK: Yes, sir.

We also have an automatic trip of the reactor coolant pumps for LOCA mitigation. Most of the operating plants, that's a scripted fast action, one or two minute reactor trip where it's an automatic reactor trip.

We mentioned our use of the ESPNDs for the low NDBR and the linear power density trip, and then finally we've used the alternative source term for our base analysis.

CHAIR ARMIJO: What is the current level at which you have a high linear power density trip?

MR. STACK: I'm not sure --

CHAIR ARMIJO: Is it ten percent above normal operating power or is it -- you know, what is that value at which you trip the reactor?

MR. STACK: Well, in the high linear power density it's looking at a local flux at that level, and it's setting it to what it's limits are at. It's not necessarily like an ex-core based trip.

One of the other things that's happening again with the SPND-based trips, I mean you have failures, you continue to select different setpoints that are more conservative. So in these, it doesn't have a fixed setpoint at all times. If you use SPNDs, I'll get you a different -- I'll go to a different setpoint.

CHAIR ARMIJO: Well, let me ask it a different way then. You have one string of SPNDs --

MR. STACK: Yes.

CHAIR ARMIJO: -- and you detect some high linear power discrepancy. What happens then? You reset or --

MR. STACK: And I chose this picture. Basically when you're looking at the DNBR trips, you're looking at an axial profile. In the linear power density, you're cutting an axial plane. So you're looking at these 12 at this level and you're comparing those with the hot spot at that elevation.

CHAIR ARMIJO: And if the power is higher than your expectations --

MR. STACK: Then you're going to trip the reactor.

CHAIR ARMIJO: And what is that trip delta? Is it ten percent above peak normal operating peak power or is --

MR. STACK: And I do not know the limits on that and the basic, but the alternative but we can get that for you.

CHAIR ARMIJO: Yes, would you please?

MR. STACK: Okay.

MR. GARDNER: Shanlai, did you have anything?

MR. LU: Yes. This is Shanlai.

I answered the question to the degree of what I am able to remember at this point, and I think AREVA pointed out about we rely on the localized rate of regeneration at the limit and there is a certain margin for them to trip the reactor. And that margin we reviewed, too.

CHAIR ARMIJO: Yes. I'm just asking what the value --

MR. LU: And I remember -- it was two years ago.

(Many talking at one time.)

MEMBER SCHULTZ: Tim, is that localized and you mentioned it was on plan based question.

MR. STACK: Yes. What's happening is that you're basically making all -- you're figuring out where the hot spot is at that elevation and you're biasing all these others to read like they're the hot spot.

MEMBER REMPE: Could you talk a little bit more about the SPNDs? Are they all the same emitter? How about your lifetime, are you planning ever refueling outage you're going to have to replace them or how long do they last?

MR. STACK: I cannot remember the details of the life is ESPNDs. We can also get that you.

MEMBER REMPE: Okay. And are they all the same emitter and what is the emitter they're using?

MR. STACK: I believe it was -- well, let me not go there. I would rather get you the answer right.

MEMBER REMPE: That's fine. It's not urgent.

MR. GARDNER: If the staff knows?

MR. LU: Yes. This is staff to answer a question.

We do have a online calibration system.

MEMBER REMPE: Okay.

MR. LU: And they do that, at the minimum they do that every 15 days. But they can go up to every ten minutes, 15 minutes to recalculate the SPND in case they see the failure. They are degraded -- I'm sorry.

MEMBER REMPE: Yes.

MR. STACK: So, I'm not sure if that answered your question.

MEMBER REMPE: I got a little more. I'm just curious.

MR. STACK: Okay. It struck me you were interested with the overall life and how long do they last as far how often do they need --

MEMBER REMPE: The expected lifetime, and then also I assume they're all the same emitter and what it is.

MEMBER ABDEL-KHALIK: If I can follow-up on something else that was provided earlier by the staff regarding the DNBR trip and the impact of slow maldistribution you indicated that AREVA had submitted a topical report describing a scale test to measure the flow distribution. I assume that this was a geometrically similar scale test, but is it dynamically similar test?

MR. STACK: We can follow-up with that. We can certainly point you to the topical report.

MEMBER ABDEL-KHALIK: Has the topical report been submitted and reviewed by the Subcommittee.

MR. LU: Yes, staff has reviewed that and we presented report.

MEMBER ABDEL-KHALIK: No, no, no. I'm asking whether it was submitted to the ACRS Subcommittee.

MEMBER POWERS: No. No, it was not.

MEMBER ABDEL-KHALIK: We would like to get a copy of that.

MR. STACK: Very good.

So basically that's going to, I know we're at a very, very high level. That's going to complete our overview of Chapter 15 as far as kind of what's unique and special about it.

Looking into Chapter 16 on Tech Specs. The generic tech spec for the U.S. EPR is the approved standard tech spec format. You can see the format that's there.

As far as a pattern, we chose NUREG-1431 as the primary tech spec that we were going to pattern after and most of the tech specs follow that. That was for a Westinghouse PWR. In addition to that, we've used other standard tech spec NUREGs as well as other precedents as appropriate.

For example, when you look at NUREG-1341 it uses a risk-based tech spec for the accumulators. We're using deterministic, we're not using risk-based tech specs, so we use the B&W tech spec value as our pattern. But in general when you look at these, we've chosen some other precedent that's been used previously or we've chosen that precedent being either one of the tech specs 1431 or one of the others as well as selected other precedents from other licensing applications.

And then we mention the N+2 safety concept here again in words of what that is, and that N+2 concept is embodied in the tech specs.

The main features that are different that we'd like highlight on the tech specs themselves, first in Section 3.3 on the Instrumentation for the digital lines, due to the nature of the I&C itself, we chose to use LCOs and actions that were component based versus function based like you typically would see. We've also included tech specs for the diverse actuation system as a part of that.

In 3.4 we've allowed for limited 3-loop operation with the reactor coolant pumps.

MEMBER BROWN: What do you mean by "component-based?"

MR. STACK: Okay. If we can look at a component as an STND or pressure transmitter, or you can step up to other components as part of the equipment versus a whole channel being done. Microprocessor units versus, say, a function of reactor trip on pressurizer pressure. It's a function-based tech spec as opposed to, for example, if there's hardware processing those trips.

MEMBER BROWN: Okay. I guess I don't totally understand. I know a microprocessor fails, the channel goes out.

MR. STACK: Right.

MEMBER BROWN: So, I mean you do have the A and B subsystems in some parts for each channel, but that's not replicated, and that's in the ALUs if I remember there's something like that. It's not replicated everywhere.

MR. STACK: In the tech specs we'll provide limits on what's required for operability and they will dictate shutting the plant down --

MEMBER BROWN: Yes, I can understand it on the detector or, you know input data processing. It's just when you get farther into the guts of a channel I wasn't quite sure what you all meant by that.

MR. STACK: But again, the limits on those whether it's an APU or an ALU have all been laid out in the tech specs themselves.

MR. GARDNER: I will say that it's still an open item that we're working with the staff to resolve --

MEMBER BROWN: Okay.

MR. GARDNER: -- and what it's going to finally going to look like when it's all said and done.

MEMBER BROWN: Well, that might be subject for some additional discussion then later is what you're talking about; how that gets resolved. All right.

MEMBER ABDEL-KHALIK: How do you specify the LCO for a component if it is providing -- if it is supporting several functions? What determines the time then? Which function is dictating?

MR. GARDNER: In the timing for the LCOs and the actions are really based on the precedent set for the operating fleet that are deterministically-based.

MR. STACK: We're also trying to capture actions that look at what things will be effected by the loss of the component so that you end up with an action. If you're having to look at all the actions, it should be taken due to the loss of that component in prescribing those actions.

MEMBER ABDEL-KHALIK: One of them must be most limiting I would imagine, and that sets the LCO for that specific component regardless of what function is it impacting.

MR. GARDNER: You could have, say, a shutdown action. But there may also be other actions maybe that's not limiting, but independent actions that need to be taken due to that failure. There may be just different actions. I don't know if I would characterize them as limiting or not limiting. They're actions that need to be taken due to the failure.

It's not like you would pick one and not do the other. You need to do those other things.

MEMBER ABDEL-KHALIK: Yes, I understand. I understand. But whether it's a 24 hour LCO or a 72 hour LCO, I'm trying to figure out which action dictates what time would be associated with that LCO.

MR. GARDNER: We're still working it. I will say again, as Mr. Brown alluded to, that within the digital control system, even within a specific division, there's redundancy built into that. So there's multiple logic units, multiple control units within a single division, so you have to have a lot of equipment component failures before you're into the division not being able to perform a function.

MEMBER SKILLMAN: Tim, I'd like you to explain limited 3-loop operation and talk a little bit about connecting that thought to the broadness of the core in the flow distribution that was asked about.

MR. STACK: Okay. And what's happened on this is we've gone and analyzed for limited 3-loop operation. Right now if I look at my notes, the LCO on that is two hours. Basically what the intention is -- let me back up.

We've had other plants where we've sheared shafts we've operated. We've operated on when we had to reactor coolant pumps per loop per steam generator. We would continue operation, automatic reduction in power and continued operation -- safe continued operation for plants where we had multiple reactor coolant pumps per steam generator.

And here we've analyzed for using one reactor coolant pump in a loop. We're setting the LCO at two hours. Basically what we're trying to do is include provisions to will allow you to have some pretty much an immediate problem that has an immediate fix, otherwise we're going cold. So it's not like we're going to operate for an extended period of time in this configuration.

MEMBER SKILLMAN: So the intention is to always operate with four loops operating and only in a casualty situation operate for two hours with less than four?

MR. STACK: You could characterize it that way.

MEMBER SKILLMAN: Okay. Will this flow situation be represented in this topical report that we've requested?

MR. STACK: This is analyzed as a part of the safety analysis.

MEMBER SKILLMAN: Are the downcomers in the reactor vessel stripped? Each of your cold legs enters the reactor vessel, right? Flow goes down, right, into the lower plenum? In theory it's made uniform by this gridding in the lower planum --

MR. GARDNER: Yes.

MEMBER SKILLMAN: -- so that there's uniform flow into the fuel assemblies?

MR. STACK: Right.

MEMBER SKILLMAN: A three pump operation. You do have a pressure disturbance and you have asymmetrical flow then -- non-symmetrical flow.

MR. STACK: I understand.

MEMBER SKILLMAN: So, I'm curious whether or not this is represented in this typical report.

As far as the one that did the initial testing, I'm wondering whether or not --

MEMBER ABDEL-KHALIK: The scale testing.

MR. STACK: The scale testing.

MEMBER SKILLMAN: Yes. I'm wondering whether or not in a 3-loop operation you really have uniform flow across the face of the core from the bottom up.

MR. STACK: I understand. We'll confirm that as a part of the scale testing.

MEMBER SKILLMAN: Thank you. Let me ask one other question. When you're in three loop operation does the idle loop flow backwards?

MR. STACK: Well, I'm trying to think on this. I have not gone back and reviewed the analysis on that. I'm too familiar with the B&W plants where it does.

MEMBER SKILLMAN: Me, too. I'd like to know. Flow reversal is peculiar when you're in that situation, the reactor coolant system is different.

Thank you.

MR. STACK: Okay. In section 3.6 Containment Systems. Again, we did not include a containment spray. We talked previously about why that was not credited for the safety analysis and it does not appear on the tech specs.

And then finally in Section 3.8 Electrical Power Systems. One of the features we mentioned in the April of 2010 discussion of alternate feeds in the electrical power distribution system. And basically what happens in the alternate feeds is you provide ties between divisional pairs. Here divisions 1 and 2 are one divisional pair and divisions 3 and 4 are another divisional pair. The reason why you do that is we have certain safety-related SSCs that if I had a divisional EDG that was out of service, they would lose the level of redundancy that they need. So what we're doing as a part of this is when we make up the alternate feed, we're sizing the diesel to carry its full division plus its alternate-fed loads. And it's capable of carrying that in this design. And this tech spec acknowledges -- this portion of the tech spec acknowledges that as something unique to the design.

Chapter 18: Human Factors. This is just an overview of what the topics are from Human Factors.

As far as highlights, our HFE program is really described in nine implementation plans that have been reviewed by the NRC staff. These programs will be implemented by the COL Applicants. We've provided a Tier 1 DAC, Design Acceptance Criteria, for these. Our HFE program does -- it is consistent with NUREG-9711, Rev 2.

When we look at the scope of the program it's covering the main control room, the main shutdown stations, tech support center and risk significant local control stations as a part of it.

When we look at our past analysis it covers a broad range of actions from looking at operations, maintenance, testing as well as safety critical actions in different operating modes. It also includes risk significant human actions that have been identified in Chapter 15 of the PRA. And then finally, unique tasks that have not been utilized on existing PWRs, for example things related to some of our severe accident heat removal or we don't have something equivalent to that on our operating plants.

VICE CHAIR STETKAR: Tim, we had a briefing on this topic, according to my notes, last August. So, it's been a while. At the time that the third bullet under your task analysis, we had some questions about how that the process that you were using to identify risk significant human actions in particular for shutdown mode, because you were numerically weighting those by the relative risk during shutdown rather than considering their contribution to the shutdown risk. Have you made any changes in that process, do you know?

MR. STACK: We are still working through the details of that, and we will provide an answer back on that question.

VICE CHAIR STETKAR: I was just curious whether any evolution had happened. Okay. Thanks.

MR. GARDNER: And we do acknowledge the caution and understand it and appreciate you clarifying that later.

VICE CHAIR STETKAR: Okay. Thanks.

MR. GARDNER: But we are going to get that answer.

VICE CHAIR STETKAR: I was just -- because, you know it's been a few months. I was just curious.

MR. STACK: And the last two points in this slide are basically the human system interface we're doing virtually and physical mock-ups with some part task situation. And then finally an integrated system validation on a full scope simulator to ensure that our HFE program does all the things it needs to do.

VICE CHAIR STETKAR: If my notes are correct, you said that each of those integrated V&V scenarios, you typically planned to evaluate somewhere in the ballpark of 25 to 50, and that's not an absolute number, but it's not 3 and it's 300 type scenarios. And each of those scenarios challenge several different types of operator interactions. Is that correct?

MR. STACK: Yes.

VICE CHAIR STETKAR: Okay. Okay. Thank you. Just to get a sense of kind of the scope of that last bullet.

MR. STACK: And with that, that concludes our presentation of the EPR review as well as our chapter evaluation. Before we turn it over to Getachew, I'll ask are there any other remaining questions?

Thank you.

MEMBER POWERS: Thank you, Tim.

At that point we'll ask the staff to present their assessment and Safety Evaluation Report.

It's worth reminding the Committee that the staff has provided us with a Safety Evaluation Report in which they highlight the open items they have, but bring it to us only when they feel that there is a pathway to resolution of those open items. And again, that is the feature of this approach to the Design Certification that makes it feasible to do this in somewhat of a piecemeal fashion.

Bonjour.

MR. TESFAYE: Good morning again.

The purpose of this briefing is to go over the highlights of the staff's findings in the areas we made in certifying, and also to give you some information regarding independent confirmatory analysis the staff has done on these chapters.

Before I do that, I will go over the history of this project for those who are not familiar with what we've done.

Again here, I have with me three members of the technical staff:

Dr. She-Peng give us an analysis of what they've done on Chapter 6.

Mr. Jean-Claude Dehmel on Chapter 11 and'

Dr. Shanlai Lu on Chapter 15.

And these are the major milestones for the U.S. EPA Design Certification application review.

The application was submitted in December of 2007.

We completed Phase 1 of the review, which is the preliminary Safety Evaluation Report with request for additional information.

And then we presented, as I mentioned during my opening remark to the Full Committee, the first three chapters that was completed in 2010.

And basically the application was completed this year, February 9th. And we completed the Subcommittee presentation two weeks ago. Today we're here to present you 6, 7, 11, 13, 15, 16 and 18. And we hope to complete our Full Committee briefing next May.

Our review schedule, as I said earlier, Phase 1 is completed. Phase 2 is completed. We hope to complete Phase 3 in July. Completion of Phase 3 is defined as the staff responding to ACRS letter. We hope to get the ACRS letter May, and hopefully we'll provide the response in June and testing will be completed in July.

We don't have any review currently in the schedule for the rest of the phases. AREVA just submitted to us their schedule for closing the open items last month. And so we are currently developing a schedule for phases 4 through rulemaking.

Our review strategy, I have described this before. I'll just go over it.

The pre-application activities started three years prior to the application was submitted. And during this time AREVA engaged the staff with the unique features of the design, the U.S. EPR design. And they also submitted some topical reports that was reviewed and approved prior to the application came in. Topical reports suggesting costing methods and quality issues and the like.

And also during the application phase we hold frequent teleconferences, mainly weekly and several audits and public meetings on key topics.

We also used electronic RAI system, which really facilitated getting the information readily.

And as Dr. Powers mentioned a couple of times, we practiced a phase discipline. In other words before we complete and move on from Phase 2 to

Phase 3 and Phase 4, we have to make sure the open items have a clear path forward to resolution. That has always ben respect to that, and we try not to generate any new RAIs in Phase 4 unless they're prompted by design change. And in the case of Fukushima we expect RAIs in Phase 4.

So, we stuck to the phased discipline and the 500 or so open items that we have identified in Phase 3 review, they all have a clear path to resolution.

Now going through the chapters. Chapter 6 these are the SRP sections where we get second findings. As you can see for two sections, subcompartment analysis and containment heat removal, we have not made a safety finding yet. There was no-- in Phase 2 there was not clear path to resolution, so we have deferred those two reviews to Phase 4.

The containment heat removal is due to GSI-191: Related Activities. So we'll bring those to the Committee during Phase 5 of the review.

In Chapter 6 there are a total of 35 open items. And this is, you know in addition to the two sections that I mentioned earlier.

And with that, I will ask Dr. Peng to describe a couple of the confirmatory analysis the staff has done to support their safety findings.

Dr. Peng?

DR. PENG: Good morning, lady and gentlemen.

I got this assignment to present less three items, but I will pass. They only give me ten seconds to present, okay?

The first slide you will see, I would like to give attention to the pressure peak due to the LOCA break around the rapid cooling cold leg pump suction side.

You probably cannot see. My first peak is about 68 psi, 70 psi and at 28 seconds. But my next slide will give you a clearer picture.

I will bring to your attention, the second peak is around --

MEMBER ABDEL-KHALIK: This is psia.

DR. PENG: psia.

MEMBER ABDEL-KHALIK: psig.

DR. PENG: Yes. Now the second peak is around the 3600 seconds and almost the same time at a magnitude about 68 psia. The design pressure is 62 psig, that's the 76.7 psia. So we have about seven psi margin.

I would like you to understand according to our SRP we need the containment ratio to be less than 50 percent of the peak at the end of 24 hours. So, this will show that they already meet the requirement. It means that heat is more resistance capable to decay of this accident.

Next slide, please.

This is a log-scale, give you a better idea of how the peak going. And a comparison between MELCOR and GOTHIC.

Oh, by the way, the GOTHIC and MELCOR both has been V&Ved completely. And the testing to decide these capabilities show that GOTHIC can have about 3 psi over prediction of the major value. And MELCOR can have 4 psi over the test results.

Are there any questions? I don't have anymore time.

MR. TESFAYE: Take your time. You have two seconds left.

MEMBER POWERS: That was good.

MR. TESFAYE: I think that you've gotten the idea that the kind of confirmatory analysis we've done on Chapter 6. Thank you, Dr. Peng.

MEMBER POWERS: Again, I myself just derive a huge amount of confidence when I see these independent confirmatory analyses.

MR. TESFAYE: Thank you.

The next one is Chapter 7. In Chapter 7 the staff has identified 36 open items, and it has gone through some of them. But an open item here is with Dr. Brown requested this regarding the watchdog timer.

MEMBER BROWN: I noticed the Reactor Trip System has zero down, and I just wanted to remind you that from a Committee standpoint, my standpoint that that issue is open relative to the watchdog timer and its ability to --

MR. TESFAYE: Yes.

MEMBER BROWN: -- satisfy through closeout a reactor trip if all the processors lock up. So, I know you show a zero there, but --

MR. TESFAYE: It's a zero. This is indicating what was described in the Safety Evaluation Report --

MEMBER BROWN: I understand that.

MR. TESFAYE: And, of course, we have generated an RAI to follow your request as an open item.

MEMBER BROWN: Thank you very much.

MR. TESFAYE: That will be tracked and will be discussed in Phase 5 of our --

MEMBER BROWN: Thank you very much.

Chapter 11 we have 24 open items and we have done several confirmatory analysis, and I have here Jean-Claude Dehmel to describe.

MR. DEHMEL: And so for the confirmatory analysis, I have two slides. One liquid effluents, Chapter 121.2 and another one with gaseous effluents, the next slide.

So, basically what we do is we look at the Applicant's information. We try to find the sources and the basis of the information for the input parameters to the computer codes. The GALE code for the liquid and gaseous effluents with reference to the case, as well as the GASPAR code for associated doses.

And then we plug in the data into our own versions of the code and crank out the results and try to see where there are matches or no matches. And typically, this is a majority process.

To start with it, it's rarely that any application, this is not unique, has all the information that we need. So, what we do is glean the information that is available. And, obviously, in this case it's above and beyond Chapter 11. We have to go to Chapter 6 in some cases, Chapter 12 or even in Chapter 15 and maybe Chapter 9 and Chapter 10 depending on the situation.

And then plug in the data. And in some cases since we try to move the process forward, we make assumptions for information that' not there, possibly the Applicant used the raw data, wherever that is. So this could be information from NUREG-0016, for example for the GALE code or the appropriate NUREGs for dose calculations.

And then if we have discrepancies or differences in the result, then we generate RAI. And this takes, you know, three or four rounds before we come to a general agreement on resulting doses.

The obvious requirements here are compliance with Part 20, Appendix B concentration limits as well as doses to members of the public that go beyond EAB in compliance with the Appendix I requirement, the design objectives for liquid effluents.

One thing that was not done in this application is because of the unique aspect -- well, not unique aspect, because initially application, the Applicant did not provide information on a cost-benefit analysis associated with one of the elements of Appendix I because you need site-specific information with respect to population distribution as well as agricultural production in the areas within 50 mile radius, and other site-specific information.

So that part of the analysis is now mandated in the context of the COL application package. So the actual application, the utility that comes forward or the consortium that comes forward has to provide that analysis.

Next slide, please.

So it's a similar pattern with the releases and the source term and Appendix I requirement. And as with Chapter 11.2, there are these COL accident analysis that kind of anticipate operational occurrences, which some of it were initial in Chapter 15 but have been relinquished or moved into Chapter 11 having to do with, in this case, the failure of a gas component and the associated release of noble gases and iodine and the impact of EAB.

That's all. Are there any questions?

MR. TESFAYE: Thank you, Jean-Claude.

All right. Going to Chapter 13. As has already been said, most of this chapter is COL information item. There were three open items on the physical protection area, and I believe they are in the process of being closed out. Nothing significant in this chapter.

Chapter 15, we have 16 open items in Chapter 15. And also in the case of Chapter 6, GSI-191, a portion of Chapter 15 is also marked included in this safety evaluation with open items. That is a Phase 4 activity.

In Chapter 15 we have done some confirmatory analysis, Shanlai will describe what we've done.

MR. LU: Okay. Next slide.

I think as part of our review, applicable to Chapter 13 because that's focused for Reactor System Branch of the review, we do a lot of analysis. And it's not just unique to the U.S. EPR, for AP1000, ESWR and we do that too. So that's what we are using to identify the issues and also to resolve the issues. Sometimes we actually end up reducing the number of RAIs because we know the magnitude of the issue and so we ask to validate so we can make sense to ask RAIs.

So, that's the tool. It's not a nonsensing calculation. Okay.

And then since we did a lot -- which was in the Subcommittee based our presentation and the slides, the curves and equations, animations, movies. So let me summarize the type of analysis that we've been doing.

And for LOCA, to verify the large break LOCA and ECCS performance, and then do the LOCA analysis, thermal hydraulic analysis.

And for transient and accident analysis we have performed a couple of neutronics and thermal hydraulics analysis using own codes.

And fuel thermal performance, this is one of the issues that came up from the LOCA topical report review, and so this is part Chapter 15.6.5 review. And we said "Okay, there is one issue related to this initial stored energy." So we performed a detail fuel thermal performance analysis.

For LOCA we run Applicant's code, that's RELAP5, we ran our own code, TRACE and RELAP5; so all three codes, separately and analyzing same type of large break LOCA event and also for small break LOCA. And what we confirmed is the EPR has quite a lot of margin, hundreds of degrees PCT margin.

As a part of our transient and accident analysis and as mentioned by AREVA we did have a unique DNBR and a high degree of --

MEMBER BROWN: What do you mean by "quite a lot of margin"?

MR. LU: Quite a lot of margin is comparing with --

MEMBER BROWN: I know, but PCP is it 50 degrees, is it 200 degrees?

MR. LU: It's actually close to 500.

MEMBER BROWN: Well, okay. Well, that's in the calibration, I was just trying to --

MR. LU: That's a lot of margin. A lot of margin.

MEMBER BROWN: It's in the eye of the beholder.

MR. LU: So I give you a lot, yes. Okay.

CHAIR ARMIJO: So you have included the thermal conductivity degradation issue?

MR. LU: Yes. That has been resolved and simply because of that it is no longer 600 degrees.

MEMBER ABDEL-KHALIK: Yes, that's right.

MR. LU: But even including that one, that's still not much of it.

For transient and accident analysis, we did confirm the online DNBR and medium power density protection system and based on SPND.

And one of the questions that the Committee asked is that for uniform flow, actually we reviewed that. And as part of our confirmatory analysis we did the TRACE analysis. And the TRACE divided the core and the downcomer into six regions.

So that's a part of the uniform flow, we did not believe it was an issue --

MEMBER ABDEL-KHALIK: In your review of this topical report do you know whether the Reynolds number in the scaled test matched the Reynolds number in the actual plate?

MR. LU: Actually, we only scaled the plate itself, but it flows and it's one-to-one flow. So one-to-one we use the same structure proposed, so the number is identical.

We are talking about the shape of the structure?

MEMBER ABDEL-KHALIK: Right. Are they geometrically similar?

MR. LU: They're geometrically Identical in terms of holes, but that is a separate test. So it's not the entire whole of the core.

MEMBER ABDEL-KHALIK: The problem is lower plenum anomaly is a large scale phenomenon.

MR. LU: Okay.

MEMBER ABDEL-KHALIK: It is not a local phenomenon.

MR. LU: Okay.

MEMBER ABDEL-KHALIK: So if you're telling me that you looked at matching essentially the local geometry rather than duplicating the global geometry, I'm not sure that you can detect any flow anomalies that would result in the lower plenum on a larger scale, including --

MR. LU: Right.

MEMBER ABDEL-KHALIK: -- the 3-loop operation question.

MEMBER SKILLMAN: That is the question I'm waiting to ask.

MR. LU: Okay. I think I can give you just one more. I don't know whether we have time to describe in detail.

MR. TESFAYE: Go ahead.

MR. LU: For the 3-loop operation, we did not particularly performance the specific analysis to address the 3-loop operation. But we did have a feedwater heater trip analysis which only one train has feedwater heater trip, not everything, all four trains. With that one, we had nonuniform flow with temperature going down towards the downcomer. And then with that one, even with the TRACE analysis, we did not see significant difference of the DNBR. And then also, we did not see the -- and since we saw so much margin with the SPND and with the algorithm itself, we did not see the need to even go further.

MEMBER SKILLMAN: Did you intend to do a transition from four pump operation to three pump operation analysis --

MR. LU: Well --

MEMBER SKILLMAN: Let me finish.

MR. LU: Okay. Sure.

MEMBER SKILLMAN: Two things happens: The flow goes up.

MR. LU: Yes.

MEMBER SKILLMAN: Because the pumps flow further out on their head capacity curve, and one loop turns backwards. And so you end up with a very different reactor coolant system.

Now, I'd be curious has that been analyzed, particularly against the flow anomaly issue that Dr. Khalik has asked? Because you can get a non-symmetrical flow up through the base of the core. You can starve a quarter of the core.

MR. LU: Yes, I understand your question.

At this point we have not done particular 3-loop analysis by ourself. And if I recall correctly, I thought that that was part of the trip that would have been initiated if you really get into that kind of flow scenario. But I cannot remember correctly enough. It was two years ago. I have to get back to you on that.

MEMBER SKILLMAN: Since that operating stage has been described, I think there ought to be action to ensure there is analysis to show that the transition from 4-loops to 3-loops thermal hydraulics and the nuclear correlation, show that we do not have DNBR concerns.

MR. TESFAYE: Yes, we will follow-up on that.

VICE CHAIR STETKAR: Designed a trip. They run back power automatically --

MEMBER SIEBER: Is there an overpower trip setdown for 3-loop operation?

MR. LU: I cannot remember. No. I cannot now --

VICE CHAIR STETKAR: Yes.

MR. LU: I cannot remember. And I remember it was --

MEMBER SIEBER: You should be -- and trip shutdown went down to 66 percent when you went from 4-loops to 3-loops.

MR. LU: Yes. Then it is a automatic issue, it should be, right? Yes. That's what I remember, but I cannot really -- like I say, I'll get back to you on it.

MEMBER SKILLMAN: I'd like to follow-up on this.

MR. LU: Sure.

MEMBER SKILLMAN: Thank you.

MR. TESFAYE: We'll follow-up this one.

MEMBER SIEBER: Generally you end up with even though there's flow anomalies, you have more flow with a 3-loop operation per thermal megawatt than you do for four of you have that anomaly for this --

MR. LU: That's what -- yes, thank you.

MEMBER BROWN: One other question. There's an open item on the undetected failures of STND. Right now there are analysis on one string of six, I guess.

MR. LU: That's right. That's right.

MEMBER BROWN: Have you all done sensitivity studies to determine just that yet or are you waiting?

MR. LU: Okay. I'll answer the question.

MEMBER BROWN: Okay. I'll let you answer the question if it's obvious.

MR. LU: Actually it's an RAI. It's an open item.

MEMBER BROWN: Okay. So that's one is still -- so you haven't done any independent analyses--

MR. LU: No, we have not. And again, because we asked in that RAI, particularly we're expecting another revision of the topical report coming in.

MEMBER BROWN: Would that effect a local DNBR issue if you've got flow anomalies and then you've got undetected failures of some --

MR. LU: It depends on the assumptions of where.

MEMBER BROWN: I understand that. But that's the finding it --

MR. LU: That's the question. And actually, this was one of the questions the staff between Reactor Systems and Digital I&C Branch we cannot resist questioning them. I think we identified the issue and had similar questions as what's the impact to the detect of the DNBR and the linear generation rate.

MEMBER BROWN: Okay.

MR. LU: And if you one -- and that all depends on where you assume that particular unidentifiable failure.

MEMBER BROWN: Okay. So we'll hear more about that in the future is what you're telling me?

MR. LU: I guess so.

MEMBER BROWN: All right. You guess so?

MR. TESFAYE: Yes. The answer is yes. And that topical report is one of two topical reports will be falling in Phase 4.

MEMBER BROWN: Okay.

MR. TESFAYE: The other one is mechanical fuel design topical report. So we'll hear more about on the topical report on the future.

MR. LU: Yes. This is Phase 2 --

MEMBER BROWN: I understand. It just seemed to be a number of factors that go into that relative to the detectors of full anomalies and backflow from the 3-loop and all the rest of it.

MEMBER SKILLMAN: Getachew, I heard you say yes we'll follow up --

MR. TESFAYE: Yes.

MEMBER SKILLMAN: -- on the transition of 3.

MR. LU: Okay. For rod-ejection accident, we also compared the peak power value and we recalculated with RELAP, whatever RELAP calculated. And we used our own code, TRACE and PARCS so we could connect this code, and then used a SCALE code starting from the exposure -- burnup assumptions based on that by the Applicant and we generated our own cross-sectional library, our own thermal hydraulics model, our own model to compare. So that's going to be the scope of work we did as part of the rod-ejection accident confirmatory analyses.

The thermal performance analysis is an interesting one. And as part of a large LOCA topical report review and also the review of analysis 15.6.5 review, and then we found out they are using RODEX-3A, which was a legacy fuel performance code to calculate the initial stored energy for the LOCA analysis. And we found out that particular code has a deficiency of the thermal conductivity model. And after we identified that, we performed our own analysis and also based -- we used that fuel thermal performance analysis code FRAP-CON and RODEX-4 came from AREVA. And RODEX-4 has the cracked thermal conductivity model. And then we ran those codes. And then we used that calculated resulted and feeding into the TRACE and RELAP5 and trying to figure out what's the impact of the PRT. It turned out to be the significant in a loop, not significant if changed to the limit by the significant interest of the PCT from 1400-1500 to close to 1700. So therefore, and then we found that it's not only this RODEX-3A, was not only used for EPR but for our present fleet, too. And so we issued a Generic Information Notice. And later we found not only AREVA, GE and Westinghouse and they all used legacy code which did not take into account the thermal conductivity degradation which resulted in the -- mostly right now there is a resulting error in cladding temperature. So therefore, following the first IN particularly issued 2011-21 to all licensees in the operating fleet and also the COL applicant regarding this particular issue.

And in that second IN was mainly calculated to the licensees using Westinghouse methodology.

So, therefore I give you a sense of what we have been doing as part of the EPR confirmatory analysis and then resulting the review also. And then also what actions will follow those confirmatory analysis.

MEMBER SCHULTZ: Shanlai, was the RODEX-4 analysis also utilized for the rod-ejection accident? Is there a reason you used the thermal conductivity degradation for rod-ejection also?

MR. LU: Well, I cannot remember. I do not recall. No. RODEX-4 was not -- no, I cannot remember now. No, it was not used as part of rod-ejection.

RODEX-4 was used later for the heat up of fuel by AREVA. And we found out that code does have a good thermal conductivity model built into that code.

MEMBER SCHULTZ: What was being used for rod-ejection? Was FRAP-CON used for rod-ejection comparative analysis?

MR. LU: No, we did not really perform the confirmatory analysis for using FRAP-CON to confirm the rod-ejection. For rod-ejection our focus was on the peak power and then the duration of the peak power.

MEMBER SHACK: I think Steve's question is that thermal conductivity effects more than just the LOCA analysis, which --

MEMBER SCHULTZ: I am looking for the center-line melting issues associated with rod-ejection, for example.

MR. LU: Okay. Okay. Okay. That part we did take a look. And then another one is after we issued 2009-23, e have a sweep however of the question, and both NRR and NRO, we took an action together and take a look at what's the -- you know, the impact of the -- you know of the thermal conductivity to it, not only just LOCA but also for all the transient and including rod-ejection.

CHAIR ARMIJO: Related to transients, do you recall the how close you came to the one percent cladding strain limit?

MR. LU: Can you repeat that question again?

CHAIR ARMIJO: The PCMI effect has a one percent cladding strain limit and normally there's plenty of margin.

MR. LU: Right.

CHAIR ARMIJO: But after you put in the thermal conductivity degradation effect, how close did you get to the one percent strain?

MR. LU: I cannot remember. I can get back to you.

CHAIR ARMIJO: Can you provide that or has the licensee -- even if you have that, I'd like to see those numbers.

MR. LU: Yes, we can get back to you. That was done by another reviewer on this topic.

MEMBER SCHULTZ: A reference on the reevaluation associated with thermal conductivity degradation would be good. I'd appreciate that.

MR. LU: Yes.

MEMBER SCHULTZ: Thank you.

MR. LU: We'll get back to you on this

one.

I think that's the conclusion that I have for Chapter 15 analysis.

MR. TESFAYE: Thank you, Shanlai.

Next up is Chapter 16. We have 30 open items, but the significant open item was what was discussed earlier, the component-based tech spec for the I&C portion, that's an open item. That's, of course, with the current application so the staff is currently review that. So, we get back to you in Phase 5 of this review.

Chapter 18 doesn't show any open items, the safety evaluation we issued didn't have any open items. However, the question Mr. Stetkar asked is still an open item. I was hoping to get that result on the issue in an RAI. But as I see it now, I think we're going to track it as an open item. The staff will issue an RAI and it will discussed in Phase 5.

And that's what I have. And the next time will be the remaining four chapters in May.

MEMBER POWERS: Thank you.

Our intention is to issue a letter that we will indicate that we are prepared to allow these chapters to go into Phase 4 and indicate what we think some additional RAIs are remaining, but many open items have to be reconsidered. And like we have done in the past.

But the Committee felt that our intention in May is to do the same kind of review on another subset of chapters. And that should begin -- we bring to a close our Phase 3 review of this application. But it is not the last time we will see this application.

MR. TESFAYE: Understand.

MEMBER POWERS: But again, I have to again thank both the Applicant and the staff for what Getachew called the phased discipline, which is bringing to us material with open items but with some idea of how those will be resolved and some confidence that they can be resolved. That has been very helpful. And again, I think that for all the work they have done to help to facilitate this operation, it's worked much better than I thought it would work.

With that, I'll ask if the Committee has any additional questions? So I will ask Mr. Skillman, Professor Abdel-Khalik if they would give me a paragraph on their issue that they have identified here.

And with that, I'll turn it over to you, Mr. Chairman.

VICE CHAIR STETKAR: I believe we are --

CHAIR ARMIJO: Do we have anybody on the bridge line that might want to ask a question? If you are, please identify yourself.

Pretty silent. We should open the bridge line just in case.

IF there's anyone on the bridge line who would like to make a comment, please identify yourself or at least make sound.

Okay. Hearing no sound, we're going to take a break and reconvene at 10:45.

(Whereupon, at 10:45 a.m. a recess until 10:46 a.m.)

VICE CHAIR STETKAR: We're back in session and for this session we're going to hear about source terms for Small Modular Reactors and Dr. Bley will lead it.

MEMBER BLEY: Thank you, Chair Stetkar, I'm Chairman of the Future Plant Design subcommittees. And a lot of the issues that are going to arise with small reactors we really need to understand the source term. As yet staff hasn't really gotten far into their development of design specific review plans, certainly not at this level.

But we have an information briefing today from NEI and some representatives here from the major new products that are coming in this way, mPower, NuScale and Westinghouse. And we're going to hear the beginnings of their work on source terms and where that's headed.

And we'd, I think, especially like to hear areas where you see there are issues and uncertainties that need to get resolved. And I look forward to your presentation. Thank you very much for being here. Sam, you're taking, Sam Hobbs.

MR. HOBBS: All right. My name is Sam Hobbs and I work for Enercon Services. And I'm here on behalf of the NEI Small Modular Reactor working group in the modularity area which happens to be the area in that working group that is working to develop a source term positions paper.

And the basic situation is that what we would like to do is we would like to establish a good technical basis for evaluating radionuclide inventories in SMRs and how they get released and how they get transported out. We are doing initial focus on integrated Pressurized Water Reactors. And I'll talk a little more about that later on.

The situation there is that for water-cooled small modular reactors with up to five percent

UO2 fuel a lot of the regulatory issues are fairly well known. And that's not completely true although there are certainly areas where, because of the differences in scale, there will be some differences.

But what we would like to do is identify the source term related design and operations attributes that are different from those of large LWRs and to propose some potential regulatory requirements in light of the existing regulatory requirements, identify some areas for focused research.

Now, the task force itself is not going to be conducting research, we don't have any funding with that. But what we would like to do, and we'll talk a little more about this later, is to identify areas where research can be beneficial for Small Modular Reactors. And the actual research is going to have to be conducted by others.

Essentially the guiding principles that we're going to be trying to use is we're going to be trying to rely on established or prior work to the extent that we can. And, as I said, we're going to focus on iPWRs, and we would try to identify the iPWR attributes and commonalities between the major designs.

One of the things that is going on is that there are certainly other designs, there's the Next Generation Nuclear Plant which is a gas-cooled reactor. There is the possibility of much more exotic technology, such as Hyperion.

And what we are trying to do is that we would like to maintain consideration of the developments in those areas and to the extent that they have done things that are useful, we'd like to be informed about those areas but we also don't want to close any doors for them as a result of positions that we try to take.

And I think that is something that's going to be very crucial. We do have someone from NGNP that has volunteered to assist and advise us from time to time. And we have a representative from Hyperion who has actually come and listened in on our working group meetings.

So we are trying not to operate in a vacuum and yet we are trying to very much emphasize the Integrated Pressurized Water Reactors. Next slide, please.

What I'd like to do is do a real broad assessment of source term identification plant design and operations, accidents beyond Design Basis events. And we think that accidents beyond the standard Design

Basis events are going to be a very important consideration for small modular reactors. If you take the current regulations and you take traditional LOCA accidents for the Integrated Pressurized-water reactors we do not really have a large break LOCA.

We have some smaller break LOCAs but we don't have a large break LOCA that would lead to core damage. And so what we're going to be grappling with, among other things, is trying to define scenarios that make sense and which we can talk about what would really happen and what would be important so that we don't neglect those areas that need to be covered to make sure that we stay safe.

MEMBER POWERS: Do you have accidents that result in the ballooning and rupture of cladding in the fuel?

MR. HOBBS: I have not seen an accident analysis that results in that. I'd certainly be willing to entertain any responses that we might have from any of the vendors that are here. Steve or Ed. Steve?

(No response)

MR. HOBBS: The accident scenarios that I have seen, that have been laid out to me basically, and I hate to use this terminology, but they're fairly boring. If you go in and you take a look at a simulated accident, which in some cases is simulated in this particular scenario by opening a valve and allowing a discharge, and sure enough you see the color of the valve change indicating that it has opened. And about ten minutes later you begin to see some water, you see some alarms, you see some water level changes but it develops very, very slowly.

And I have not yet seen accident scenarios proposed that actually lead to core damage. And yet I don't see how, if we're going to be responsible in terms of trying to assure public health and safety, that we can neglect the possibility that that's going to happen.

And so that's the reason that I've said I think we're going to have to be looking at beyond Design Basis accidents, in some scenarios, to come up with something that will allow us to have designs that will protect the health and safety of the public.

MEMBER POWERS: The only reason I asked is that the first release of radionuclides actually come from the coolant, because you have a certain amount of contamination that's a trivial amount of activity, typically. Where you get significant amounts of activity is when you balloon and rupture the cladding.

And it's not dependant on having a large break LOCA, or any other accident sequence, it's rupturing of the cladding and venting the gap inventory. You know the gap inventory is a peculiar definition because it's not just the material that's in the gap, it's more than that. And whether that's beyond Design Basis or within the Design Basis, not so much important as it is ballooning and rupturing the cladding.

MR. HOBBS: Right. I certainly would agree that that would be my understanding is that that's where you see significant accident source terms is when you get into scenarios that that happens.

MEMBER POWERS: And typically the problem that you run into is that any kind of a transient event that the cladding, it's relatively thin sheet of metal. Relatively easy to over-pressurize and rupture it.

MR. HOBBS: As I have said I think that is an area that we know we're going to have to deal with. One of the things that we actually want to do is identify, evaluate potential regulatory applications, areas where we think regulations may have to be modified or adapted and to identify areas where research would be beneficial for the integrated PWRs.

What we're going to be trying to be doing is to define some specific research topics where those are appropriate.

MEMBER POWERS: A lot of it gives standings to what you define as research to people that call themselves researchers tend to go off and tell you they're looking at fundamental mechanisms and things like that. People that don't call themselves researchers would define research as just looking at the literature and just seeing what's available. How are you defining research here?

MR. HOBBS: I think it's a combination of those. We obviously don't want to neglect the research and the work that has been done to date. And I think it would be very foolish to do that.

On the other hand there are areas where new things need to be done and we have initiated some conversations with EPRI to find out where they are going and what they are doing. How we might have some common interest with some ongoing research.

They are actually considering, and I think that it is an area that would possibly be beneficial to us, some decontamination factors for aerosols and for escape of radionuclides through cracks. So that is a potential area. That is not an area that we have identified and focused on as something that we're absolutely going to do.

And I will talk a little more about some other areas where we think some research, or possibly some development, may be needed as we get a little further along in the discussion.

MEMBER POWERS: Assuredly the transport of aerosol through narrow openings has been an active area of survey for some period of time. And usually the controversies that arise is people tend to look at Norwood's criteria on that and not look at all the criteria for determining whether things deposit or not in cracking and whatnot.

My understanding is that the Europeans have initiated some additional work on that. For looking at just looking at things like just gas flow-through cracking but they also intend to look once again at the issue of aerosol deposition in the cracking.

And it's been done before, the Dutch looked at fairly extensively back in the 80s. And the problems with those results is that reactor accidents aren't as benign as people draw them on paper. There are shocks and vibrations going through systems.

And when the researchers looked at

deposition they found it to be reversed relatively easily by shocking and vibrating things.

MR. HOBBS: All right. And one of the areas that, as I understand it, they're looking at aerosol droplet sizes and at the possibility of trying to establish droplet sizes that are more appropriate for potential accident scenarios. And that does not address what you just said with regard to shocking the system and possibly exacerbating the releases as a result of that.

MEMBER POWERS: Particle size is an area that depends on where you're talking about. But typically in the containment it's a relatively developed technical field. People are pretty good at predicting these things I think.

MALE PARTICIPANT: Again it depends on how accurate you wanted it.

MEMBER POWERS: Yes, there's always this problem of scientific adequacy and regulatory adequacy and how much you know. No issue ever gets resolved in a scientific fashion. Every research report that's ever been written always has this section, further work can be done, you know, babble, babble, babble. Sometimes we say well it's good enough for regulatory or safety analysis, we don't need it after the third significant digit.

MR. HOBBS: I've actually written some of those sections myself.

MEMBER POWERS: Oh yes, we all have.

MEMBER SCHULTZ: Sam, I've taken what you've said so far to mean that you're looking for programs, EPRIs programs for example, where the technology or the evaluations, the research that they're doing, it's not aimed at SMR., but it's research that is being done that may be applicable to SMR?

MR. HOBBS: At this point that is what they have been doing. We are hoping to have some further discussions with EPRI and to find out if there are some areas that would be more directed that EPRI might have funding available.

If and when we can get the definitions done to the right extent we may want to seek funding from other sources than from the EPRI's Committee that actually funds that directly.

Potentially we could be wanting to talk to Department of Energy, we have not done that yet. I think it would be premature for us to do that at this point because I don't think we have a definition of the kinds of things we would like to look at at this point. But the Advanced Nuclear Power Division in EPRI does have some work ongoing that does appear to have potential of being applicable.

And so we certainly don't want to ignore that. And that's not just a matter of literature research, but it may be that with some very small extensions to that research that it would be even more applicable to us.

MEMBER SCHULTZ: Still a lot of work to be done defining, refining boundary conditions and such.

MR. HOBBS: Yes, absolutely. And to define and refine what we think is needed. And quite honestly we're not there yet.

MEMBER SCHULTZ: Thank you.

MR. HOBBS: We formed this working group and the working group has had one meeting. And so that makes it a little awkward to talk about the details technically and specifically. And yet there was a desire to try to keep ACRS and NRC staff informed and we didn't want to neglect this opportunity to do that.

MEMBER SCHULTZ: That helps to define where you are. Thank you.

MR. HOBBS: Okay.

CHAIR ARMIJO: It's my understanding that these designs are intended to prevent the core from ever being damaged or uncovered as a result of any of your design bases accidents. So how are you going to create a situation where you would actually generate aerosols and all of these things?

Or is that your intent? To just say well forget the fact that we won't uncover the core, we're still going to postulate? It's kind of strange but, you know, I don't understand what you're doing.

MR. HOBBS: Well I cannot speak with certainty to where all of the vendors are going to go because I'm not personally affiliated with any specific vendor. And yet, it is my understanding that we are going to have to do that.

MEMBER POWERS: Part 100 pretty much requires you to do that.

MR. HOBBS: I'm sorry?

MEMBER POWERS: Part 100 pretty much requires you to do that. I mean it's a defense in depth measure --

CHAIR ARMIJO: So it's a non-mechanistic release?

MEMBER POWERS: Absolutely. It simply says that, I mean the thesis behind it is, yes, we're sure that all your accident prevention measures work.

We're sure all your emergency response systems work. But in the off chance they don't, please tell us what mitigation capabilities you have should you get a release.

Now what the alternate source terms says is, and by the way that release into the containment, here's a prescription but it would be better if you had one that was peculiar to your cell. And please develop it. And it actually outlines kind of what you have to do. But yes, it's a defense in depth measure.

It's one of the best regulations in the notebook because it's completely technology independent. So it's particularly suitable for you guys with innovative designs because it's not dependent on what it looks like.

MR. HOBBS: Yet, on the other hand, I do think that there is some desire, once you've decided that you're going to uncover the core, through whatever mechanism you're going to do that, that you would like to do mechanistic source terms to the extent that you can. And not just talk about going back to the days of TID 14844 of saying we'll just take a certain percentage.

And I've certainly seen even some pretty modern publications from IAEA that suggest some percentages. And, in the end, that may not be an unreasonable number to have come up with. And yet you'd like to have some basis for it.

MEMBER POWERS: Yes, I mean the alternate source term definitely lays out a pattern for you to use. Fairly realistic thing, it plays the 14844. The problem with using 14844 is it's not realistic.

MR. HOBBS: I understand.

MEMBER RAY: It is a little hard to calculate risk though, non-mechanistically, if you're using PRA and risk informed regulation and all that.

MEMBER POWERS: You typically don't. Part 100 has --

MEMBER RAY: I understand. I'm just making an observation that if you can't imagine how something can happen it's hard to assign a probability to it.

MR. HOBBS: In terms of taking a look at the integrated pressurized water reactor commonalities the designs are extensions of proven technologies. We're using modified PWR fuel assemblies. Significant increased use of passive features. And, in some cases, there are multi-module considerations to be taken into account.

In terms of operations areas, there are

questions with regard to the control room, the operations, surveillance and maintenance, fuel handling and storage. And there are some innovative approaches in some cases that are going to lead to probably some interesting discussions with the regulators.

MEMBER SKILLMAN: Sam, before you change slides, please. The multi-module consideration, looks like you conceive of a situation where the modules are added incrementally. So at one point in time in one of these plants there is one or maybe two modules with the isotope burden that comes from the burn up and the power history on that one or two modules, but in time perhaps there's 16 or 20 modules and each one has it's own fission product inventory. Its own actinides and its own transuranics.

What consideration is being given to what is really a riddle of burn up, decay, leading to a full plant source term versus just a single module source term?

MR. HOBBS: I would be speculating if I spoke to that in detail. But we have had some discussions and the NuScale, in particular, was thinking about up to 12 modules. The Generation mPower, the material that I've seen typically looks at either two modules or four.

And certainly I think what you have to do is you do have to look at the varying burn ups and with 12 modules, which would be in some cases perhaps the most complex case, which would be one of the designs that we're talking about.

Eventually you would get into a situation that would, in some fashion, be a burn up and source term equilibrium in that you would be having, on a very significant percentage of the time, you would be having one module in refueling and the other 11 generating. And then they would be in varying stages of decay and of having developed a source term inventory.

That's perhaps not quite as much of an equilibrium situation with two modules or with four modules, but you would still need to be looking at that.

The other thing that I think we're particularly concerned with would be are there, and I'll use this term loosely recognizing that there are people in this room that know a great deal more about probabilistic risk than I do, but of common mode failures or of common systems. And an obvious potential example would be radwaste systems that if you have multiple modules there are economies of scale for having a single radwaste system that's serving multiple modules.

In the case of one of the reactors, in the case NuScale, there is one very obvious common feature that is shared between all of the modules, and that is the pool that the modules are submerged in. There are probably some other shared systems and components that need to be thought about and we're not planning to try to address Fukushima issues directly until the regulations I think settle out.

But I think you can take a look at what happened in Fukushima and see that there was a major external event that had all sorts of unforeseen consequences. And what we would like to do is try to avoid situations where we haven't had some amount of foresight in going forward and in understanding the source terms.

MEMBER SKILLMAN: Thank you.

MR. HOBBS: Was that responsive? I'd like to think that it was and yet I feel like I'm talking in generalities.

MEMBER SKILLMAN: The burning thought I have is having multiple modules, with differing power histories and therefore different fission product inventories. And as Dr. Powers said, there will I believe need to be a requirement to determine what is the consequence of release of some fraction of that burden, that radionuclide burden.

And so I've got a good old module A that's been out there for 24 months. I've got module B that's had some problems so it's only got 16 weeks of effective full power. I've got module number three that's a year and a half in. So I have, basically, a riddle of different power histories among these modules. And now if I say, what is the source term for the whole plant, the answer I believe is it varies. It changes from one week to the next, one minute to the next.

MEMBER POWERS: But it's just a linear factor. And the subject on behavior of the radionuclides is computed basically in a logarithmic fashion. And so the fact that you have two or three, factors of two or three uncertainty in the source term pales to the fact that you've got order of magnitude uncertainty and your subsequent calculation will be age here.

I mean we agonize over this all the time. We used to use source terms always with end of life fuel and people said well that's not fair. You should use a more representative one. We go through all the machinations, the various burn ups in the core and things like that trying, but it doesn't make any difference. The uncertainty is associated with the subsequent behavior of the radionuclides.

MEMBER SKILLMAN: I'll take Dr. Powers' word for it, thank you.

MEMBER POWERS: I mean you can do it. It's one of those things that you can do so you just go ahead and do it, because the tools all exist. But it will not, in the end, change significantly what the answer is that you get.

MEMBER SCHULTZ: All right. So, Sam, this is a general slide showing the commonalities among the iPWR grouping? Because they're also commonalities to where we are today with LWR.

MR. HOBBS: Yes.

MEMBER SCHULTZ: All of these topics?

MR. HOBBS: All of the topics. Including even the multi-module topics from multi-unit plants.

MEMBER SCHULTZ: And you didn't discuss storage but that's certainly a feature associated with release and emergency planning on it's own?

MR. HOBBS: Yes, absolutely. And the last topic, the fuel handling and storage is the last button on the slide and that is an area that I think we will be concerned with.

We did, at least for a handful of the parameters, looked at Generation mPower, NuScale Power and Westinghouse in terms of things that are potentially common or that have some differences.

Containment, all of the containments are below grade. Two of them the containments are submerged. With control rod drives we have two that are in vessel, one that is outside vessel.

Reactor coolant pumps. We have pumps inside RPVs with external motors. Seal-less pumps inside RPV with external rotors and one natural circulation driven system that does not have reactor coolant pumps.

All of them have integral steam generators. Two of them are once through, one is Helical Coil.

Refueling frequency is up to four years for Generation mPower. Two years for NuScale and for Westinghouse. So there will be a realm of things that are very similar here and yet things where there'll be some differences. And I wanted to put this out so that there would be some understanding of what we're dealing with with regard to trying to come up with both commonalities and differences in terms of going forward and developing a position paper.

In terms of technical considerations, and we've actually talked about the very first bullet here, the definition of licensing basis events. And the fact that we are likely going to have some difficulty in defining a significant core damage event. And so we're probably going to be looking at things that traditionally would be beyond basis events as a part of a licensing basis event and that's going to be perhaps a little tricky but nevertheless it's something that we're going to have to deal with.

In terms of general areas we're looking at magnitude of releases timing. We'd like credit for passive design and credit for other design features.

Source term treatments. I think the question of what we're going to do is you can have non-mechanistic source terms or you can have mechanistic or you can have a mixed or hybrid approach using some elements of each. And I suspect that we will probably end up depending on the particular scenario and accident analysis that we're looking at with a mixed or hybrid approach. And in some cases we will probably want to do mechanistic source terms. In other cases that may not be something that would be appropriate or necessary.

MEMBER ABDEL-KHALIK: So there are no reactivity or power-maldistribution events that could actually result in local fuel failure in any of these designs?

MR. HOBBS: I will defer to the vendor representative who are here. I don't know of any but that doesn't necessarily mean that that is not the case. Steve and Ed?

MEMBER BLEY: If you'd come to the microphone and say who you are and speak.

MR. KLINE: Steve Kline with Bechtel, representing mPower. To this point we haven't identified a situation that results in that. But obviously we're fairly early in our analysis at this point so it's ongoing. But to my knowledge we haven't identified anything that results in that type of situation yet.

MR. MIRSKY: Steve Mirsky, NuScale Power. Since of the three designs we have an external control rod drive mechanism from the reactor vessel, we have identified the fact that we will be evaluating the rod ejection accident.

MEMBER ABDEL-KHALIK: Thanks.

MEMBER BLEY: Sam, go ahead.

MR. HOBBS: All right. In terms of the detailed technical considerations --

MEMBER BLEY: Can I ask you a sort of an organizational question? I mean your efforts focused on the source terms but you're crossing the boundaries with Design Basis events, licensing basis events, how the vendors will approach licensing issues and risk assessment issues, Probabilistic Risk Assessment. Are you trying to address all of those issues? Is this an integrated program? Or is the focus strictly on source terms but with kind of feelers out to the other areas or is that all coming through the designers?

MR. HOBBS: I think that it would be more accurate to say that the NEI is going to be focused on source terms with feelers into the other areas. I certainly would be willing to be corrected if someone has some other thoughts, but that's been my understanding and the presumption that I've been trying to do the work on here.

MEMBER BLEY: Okay, thanks.

MR. HOBBS: But it does, it crosses boundaries. And that's both unfortunate and interesting.

MEMBER BLEY: Yes.

MR. HOBBS: In terms of the detailed

technical considerations, we've touched on a number of these things in earlier discussion, fission and activation, product inventory. Release reactions, timing of release, radionuclide composition, physical and chemical form of releases, the release thermal energy, fuel damage mechanisms. And those are familiar I think to the NRC staff, certainly familiar even to the non-iPWR community in terms of Small Modular reactors.

In terms of trying to make sure I didn't leave anything off of this slide I particularly looked at NGNP white paper for guidance and, in fact, the very information that they talked about was all relevant to us as well in terms of the very big picture. Certainly differences in terms of specifics.

MEMBER BLEY: So pretty good list. Are there areas in here where you currently expect that you're going to have to do new work? Maybe experiments, maybe R&D efforts, or can you say that yet?

MR. HOBBS: I don't think we can say that yet for sure. I think that the concern that I have heard expressed, probably more than any other, is the timing of releases. And obviously the longer it is before you have a release the more opportunity there is for the very short-lived isotopes to decay. And so that's of significant consideration. Whether that will result in specific new or additional research I think it would be premature to say.

MEMBER BLEY: Okay.

MR. HOBBS: But that, to me, seems to be one of the keys that I have heard come up time and again in the discussions that we've had.

In terms of non-safety considerations, I think that there is a tendency for passive reactors to say we have an accident so the operator ties his hands behind his back and he doesn't do anything for 72 hours.

Well, I've never known an operator who would have consented to that and that is certainly not what we would want to see happening. We would want to see the operators taking the right actions. And for any active design feature that are available to kick in and to minimize accident consequences.

None of these reactors are going to depend on stand-by diesels that have to start in ten seconds, as is the current operating in large water reactor considerations. And yet I suspect all of them are going to have some sort of standby power onsite. And if you did have a loss of power event we would expect that standby power to be available and to be put into place as quickly as possible.

And so that would be an area of an active design feature that we would take advantage of, but would not rely on, for a regulatory consideration. The same think with operator actions. And I don't have, necessarily, real good examples of what the operator would do and I think it depends somewhat on the specifics of the particular design. In terms of --

MEMBER POWERS: Turn your spray on and it will eliminate all of your source term --

MR. HOBBS: I'm sorry?

MEMBER POWERS: Turn your spray on and it'll eliminate your source term conditions.

MR. HOBBS: Well I'm not sure that we have spray in every case.

MEMBER POWERS: Oh if you had sprays then you wouldn't have to worry about this stuff.

MR. HOBBS: Well, I'm not sure I'll get to make that decision, Dr. Powers.

MEMBER POWERS: Well maybe it just gives you another set of things to worry about.

MR. HOBBS: In terms of the scope of the regulatory evaluation and the areas that we're trying to look at, looking at Part 52 and Part 50 as applicable. And there are certainly scenarios in Part 50 that talk about LOCAs and that imply large-break LOCAs and obviously we're going to have to be grappling with that. We've already talked about that to some extent in terms of accident analysis definitions and approaches.

In terms of regulatory guides, obviously we're going to be taking into account Regulatory Guide 1.206, which in some cases provides some pretty good guidance. Regulatory Guide 1.183, and we've already talked about Alternate Source Terms somewhat. And then in terms of the prescriptive approach.

Standard review plan --

MEMBER BLEY: I see you have that up there. Are you closely following or are you actually in discussions with staff about the design specific modifications. Design specific review plans that are under development in licensing the SMRs?

MR. HOBBS: I am not.

MEMBER BLEY: I hope your project is along the way.

MR. HOBBS: Yes. It is my understanding that all of the vendors are in those discussions with the appropriate staff members.

MEMBER BLEY: Okay. That makes sense, thanks.

MR. HOBBS: But I'm personally not because I'm not actually involved in those specifics.

MEMBER BLEY: Okay.

MR. HOBBS: No one seems to want to jump forward and speak to that issue. With regard to the Standard Review Plan, NUREG-0800, when we started off we listed a series of chapters that we thought were particularly relevant. And then we began adding chapters and adding chapters. And I think eventually I ended up that I didn't have Chapter 1 on. I didn't have Chapter 16 on since that was more or less the recipient of the results.

And I didn't have Chapter 17 on. Virtually every other chapter seemed to have some sort of implications with regard to source terms. There might have been one or two -- I always draw a little bit of a blank on administrative programs as to which chapter that is. And most of the chapters I know pretty well.

But generally what we're going to be doing is we will looking through the accident scenarios that would present themselves. Whether it is site oriented, whether it is related to structures, to the design of the reactor itself or the reactor fuel, of auxiliary systems and engineering safety feature systems. And just taking a look at the entire spectrum.

When you get to Chapter 15, I think there is a tremendous tendency to focus on the traditional bad actor in Chapter 15, double-ended guillotine break of the largest pipe that is going in and out of the reactor. And that turns out not to be nearly as big a deal for a small modular reactor with passive design because it's not a terribly large pipe. The steam generators are all integral to the reactor vessel so that you don't have that sort of a consideration.

And you generally end up with fairly small line breaks, which is not to say those are not significant and don't have to be dealt with. And yet, the fact is, Chapter 15 has got a very large number of postulated accidents.

Fuel handling events, control rod drive ejection, which in some case, at least one of the cases would be relevant. Reactor coolant pump failure, which even with internal reactor coolant pumps certainly you're going to want to know what's happening if you get a sudden failure of a pump.

So there are a vast array of accidents in

Chapter 15 which are traditionally postulated. And there are also accidents that are postulated outside Chapter 15, such as tank ruptures, that need to be looked at. And to some extent those will look somewhat similar to current generation of reactors, it'll be on a different scale. Different design specifics, but we're not trying to neglect that.

And yet, on the other hand, what we do need to look at very seriously is the accidents that will lead to core damage. And so we're struggling somewhat with that.

MEMBER POWERS: If I were thinking about this and what's been done that might be applicable to this I would say that the biggest hole that you will probably identify will be the fuel handling accidents, because you have such a unique system and because those are not typically extraordinarily important for large plants compared to the double guillotine-type pipe break.

It might be a bigger hole for you. And you're not going to find a huge volume of literature, I suspect, out there. I'm just guessing.

MR. HOBBS: Okay. That certainly is a reasonable speculation. Something that I think that we will have to deal with. During one of the SMR.

workshops we did have a staff member that talked about the fact that the moving a containment into an area as a part of the refueling operations. And that's certainly something that is unique to one of the designs. And so I'm sure that that will not be the only unique feature.

And some of the other unique features might not be applicable to all of the reactors but I think we will end up having to deal with those. That's not a fuel handling action per se but it's certainly something that impinges on refueling operations.

In terms of applications of source term modeling, we're looking at the whole realm of considerations. Plant design, being able to handle component design and operational life issues. Being able to do design in such a fashion that we're considering worker safety.

In terms of normal operations none of these topics, operational leakage, fuel handling or refueling, maintenance or in-service inspection or testing, are new per se and yet all of them with Small Modular Reactors are going to be somewhat different in scale. And different in approach. And so those are things that we're going to be having to take a look at consider.

Decommissioning, ultimately, will also be a consideration. So those are some of the areas that we're in particular going to look at.

MEMBER SKILLMAN: Before you change that slide, let me ask this question please, Sam.

MR. HOBBS: Sure.

MEMBER SKILLMAN: It seems you have one of these plants that has a dozen modules, I guess as I envision it you have basically maybe a 300-megawatt core and you've got a dozen of them. And so you've got 100/150-megawatt turbine, you've got a dozen of those. And it seems like there is going to be a requirement for the people that operate this plant to spend a whole lot more time on the secondary side. Care and feeding of the turbine, hotwell, drainage, return of water for heating to the steam generator.

For now we have one high-pressure turbine and maybe two or three low-pressure rotors and full attention riveted on this one big machine plus the reactor. Now there is attention on maybe a dozen machines and a dozen turbines. To the extent that there's any leakage then this leakage is confined to that one turbine that is handling that one reactor.

It seems like there is going to be an

exponential shift in how one thinks about which core, which turbine, how that relates to the other cores and other turbines. What's the conceptual thinking, along that line, that you're giving to that type situation, please?

MR. HOBBS: Let me address a little bit the scale. In that the one design that we're currently looking at in terms of that would have that many modules would be the NuScale design, which, as I recall, is about a 45-megawatt design. So that we're talking a little smaller scale. In the case of generation mPower and the Westinghouse designs, certainly talking a larger scale but also significantly fewer modules that are currently envisioned, I think either two or four instead of twelve.

And certainly the secondary side is going to require a great deal of attention. But I think even the multiple modules are going to require more attention regardless. In terms of what the interface will be between modules with regard to source terms, I think that's an area that I'm not prepared to try to address right now. But it's certainly something that we need to think about.

MEMBER SKILLMAN: Thank you.

MR. HOBBS: I don't know that that was very responsive, but that's the best I can do right now.

MEMBER SKILLMAN: Thank you.

MR. HOBBS: Other areas where we're looking at applications of source term modeling include, obviously, accident analysis. We talked about Design Basis and beyond Design Basis, in particular control room dose is likely to be of a significant consideration for a couple of reasons.

Security, our primary thinking there is that what we're going to be doing with regard to source terms will be useful to security personnel in terms of being able to identify a Design Basis threat source term. I don't envision at this time that we're going to have a lot of addressing of the security issues as a result of the NEI position paper.

And yet, to the extent that we can establish the criteria and approach, they will probably have some implications there.

Emergency preparedness, we actually do have an emergency preparedness representative who is very active across a number of different facilities and who is involved in our working group.

MEMBER SCHULTZ: Sam, can I take from this slide, the way at least it's laid out, that there's some thinking that there might be a Design Basis approach associated with control room dose, onsite dose, calculations? And then, with respect to offsite consequences and emergency planning, there might be a beyond Design Basis approach aimed at, as Dana was saying, a Part 100 expectation requirement?

MR. HOBBS: Dr. Schultz, I think it would probably be premature to reach that conclusion. So I wouldn't want to say that. This may have been an unfortunate way of organizing the slide.

MEMBER SCHULTZ: So it's not there yet?

MR. HOBBS: It's not there yet.

MEMBER SCHULTZ: Thank you.

MR. HOBBS: In terms of potential research areas, and I'd like to emphasize that we are very early in the stages of trying to identify potential research. As I had commented earlier, our working group has had one meeting. We're going to be trying to plan an additional meeting as well as an interaction with EPRI in the relatively near future. And so we'll be further along in another month or two.

But areas that we had originally talked about was beyond Design Basis computer code development and verification. And I think there are a number of computer codes that are being used, MELCOR is one that comes to mind, that I guess got used on the SOARCA Project. And I think there are some subsidiary codes, or developed codes from that. I think MACC is another code that was also used there that as I understand was closely affiliated with MELCOR.

And one of the things that we're concerned about is verification and validation of the code, because that is an area that is not necessarily easy to do. We have had some very preliminary one-liner interactions with the Nuclear Regulatory Commission in SMR workshops with regard to that where one of the primary concerns, I think, is the qualification of the analyst, which is certainly something that I think needs to be given a lot of attention. And yet, I think we're also concerned with the validation of the codes and with whether we'll be able to do that.

We're particularly interested in passive removal mechanisms in small containments and then small reactor coolant system volumes. We haven't gone very far in terms of trying to define that.

MEMBER BANERJEE: What codes are you thinking about for that?

MR. HOBBS: For the passive removal

mechanisms? I don't know, we haven't defined that.

MEMBER BANERJEE: You might consider the role of non-condensables. It's almost the codes that we use don't necessarily take these into account very well.

MR. HOBBS: All right. Thank you. We will certainly be studying the transcripts of this meeting very closely. I've heard a lot of good suggestions today.

We have some concern about atmospheric dispersion at close distances, in particular less than 100 meters. One of the areas that I think is of concern is control room dose. In the case of small modular reactors the control room is relatively close to the containment. It is relatively close to the release point. And the typical mechanism of atmospheric dispersion is probably not appropriate.

I'm not sure where we're going to go with that but it's an area that we've identified that we might need some additional work. There's some potentially applicable existing research --

MEMBER BANERJEE: You may actually need to do some, you know, because of buildings and all sorts of things, you might have to take more of a sort of CFD-type approach of close distances. I know there are differing options about that.

MR. HOBBS: I'm sorry, I'm not familiar.

MEMBER BANERJEE: For computational fluid dynamics dispersion rather than the sort of averaged type approaches that are taken at far distances away from the source.

MEMBER POWERS: CFD works pretty well for the dispersion portion of it. It's horrible for the deposition.

MEMBER BANERJEE: Yes, because of course the turbulence is not properly understood.

MEMBER BLEY: You just wanted to do this so you could ask questions, right?

(Laughter)

MEMBER POWERS: Well I mean the problem that it basically boils down to is CFD is most comfortable treating particles as though they were point particles whereas they're actually finite. And so you have to go in and do something with it to handle the finiteness and size of the particles.

MEMBER BANERJEE: It's a hard problem.

MEMBER POWERS: It's a hard problem to do that.

MEMBER BANERJEE: Which is why we don't give much credit for deposition GSI-191 using --

MEMBER POWERS: I wondered why you didn't do that. You actually had a reason for doing that? It wasn't capricious and arbitrary?

MR. HOBBS: In terms of potentially applicable existing research --

MEMBER POWERS: Oh let me interrupt, and say it is my perception that the Department of Energy, for its nuclear facilities, they're not reactors but other kinds of facilities, has been struggling with this near source dispersion sort of thing. And there may actually be a model up at Los Alamos that has some pedigree to it.

MR. HOBBS: Thank you, very much.

MEMBER POWERS: Yes, in connection with, you know, they're building a plutonium facility and lots of people agonize over plutonium. And they're all near source kinds of dispersion problems. And so it's my perception that they've agonized on it. And that pretty much exhausts my knowledge of it. But they might be a good place to check first before you tried to do something yourself.

MR. HOBBS: Thank you. Thank you very much.

MEMBER BANERJEE: And at one point Lawrence Livermore did some work as well. I've forgotten the name of his code, but he did some work.

MEMBER SCHULTZ: And Argonne Laboratory has done that too.

MEMBER POWERS: It's just a real headache for their radioactive material facilities and so it's very analogous too, so you might want to check with them.

MR. HOBBS: All right, thank you. Those suggestions are the tip of the iceberg on literature research and yet it keeps you from neglecting an area that you hadn't thought about.

MEMBER POWERS: Well the trouble is this stuff is not going to show up in the literature.

MR. HOBBS: So you're going to have to --

MEMBER POWERS: Just call them. Call the DOE counterpart. Like EM and DOE and ask them and they probably get you quickly on the track and probably even tell you what they like and dislike about the models.

MR. HOBBS: All right. Thank you very much. In terms of potentially applicable existing research, we're already talked some about the EPRI studies of removal mechanisms for escape of radionuclides through cracks. I think we may have exhausted that already. It's our understanding that

PNL is engaged in a chi over q study.

And I believe that, actually, NRC research has been assisting in funding some iodine studies that are coming to fruition fairly soon. Right now the name of that program escapes me. But perhaps someone would like to go to Europe to hear the results of that research this summer. I don't think that's something that I'll be doing but that's a possibility.

In terms of our research strategy, is that we're looking for potential sponsoring organizations. DOE is a possibility. We have not yet engaged in conversations with DOE. National Laboratories are a possible organization that might have some interest. EPRI, and we have at least had a discussion with EPRI and plan to have some further discussions with them.

MEMBER BLEY: Sam, following up on what Dana said, my memory is that the Army, the Army's Chemical Weapons Destruction Program, did some work on close in dispersions, non-energetic, work a few years back, Sandia may have helped done that.

MR. HOBBS: All right. Thank you. And obviously in addition to DOE and the National Labs and EPRI there may be some universities that either could be engaged or possibly have some ongoing work already. We're trying to identify possible funding. We have some very significant concerns with meeting the near term technical and regulatory needs. What we would not like to do is to launch research that isn't going to be beneficial in the relatively short-term.

If there is follow-on research to that that would be beneficial in the longer term we'd certainly be delighted to see that launched. But what we are really interested in is something that will enable us to proceed or give us some direction to proceed in the relatively short term.

As commented before that we are, as a working group, not going to be in a position to fund the research, so we will be looking for funding and for people who are interested in these subjects to provide some funding and/or some assistance in those areas.

In terms of our path forward. We are in the very early stages of establishing a dialogue with the NRC. Identifying involved branches and the personnel to talk to. I'd like to put in a plug for the small modular reactor workshops. I think those have been invaluable. They have been going on, as I recall, since October of 2010, and have been going on more or less quarterly. And that dialogue has been very helpful. And we have had people that have shown up that have provided some informal suggestions at some of those workshops that has been very useful.

And of course we're also very interested in seeing what the NRC is doing and thinking about, both in terms of the design specific review standards and I believe the terminology is the IIPR that they're currently working on.

We are going to be trying to develop a position paper. I don't have a date yet for when we're going to be doing that. This year, but I don't want to commit to anything more ambitious than that, although perhaps we'll be able to beat that. I've seen some of these position papers go through 15 or 20 reviews and revision cycles so I'm not going to be too ambitious in terms of a time line.

And, lastly, and I've said this before but I'll say it again, we need to pursue the identification and scoping of the research activities that will have a potential benefit for SMR. deployment. And that concludes what I have to say. If there are any further questions I'd be glad to do what I can to answer.

MEMBER RAY: I assume you're presuming a Part 52, ultimately?

MR. HOBBS: I would say yes and I'll

qualify that slightly.

MEMBER RAY: Good. I think that's wise.

MEMBER SIEBER: Have your own part.

MR. HOBBS: I'm sorry?

MEMBER SIEBER: You'd like to have your own part, right?

MR. HOBBS: Well actually my real reason for qualifying it is that we have at least one potential applicant that is talking about coming in through Part 50 initially.

MEMBER RAY: That's the right way to do it in my opinion, but that's just my opinion.

MR. HOBBS: And so we certainly don't want to say that we only want to do Part 52 and see the door shut on potential utility applicants as well.

MEMBER BLEY: Anything else from the Committee? Any comments from the public on this? Sam, thank you very much for a great presentation and discussion. Mr. Chairman, back to you, early.

CHAIR ARMIJO: Well thank you very much, Dennis. So what we'll do is we'll take a break for lunch and we will reconvene at 1:15.

(Whereupon, the above-mentioned matter went off the record at 11:54 a.m. and resumed at 1:14 p.m.)

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

1:14 p.m.

CHAIR ARMIJO: Okay. Good afternoon. This afternoon I'll take the lead introducing the xLPR Project. That's the Extremely Low Probability of Rupture Project. The Materials, Metallurgy and Reactor Fuels subcommittee was briefed on this project in September of last year, we provided feedback and comments.

The project intends to develop a probabilistic fracture mechanics tool for use in leak before break evaluations of PWR cooling boundary components fabricated from nickel-base alloys and operated in environments capable of causing primary water stress corrosion cracks to initiate and grow.

Since its initiation in 2009 the project has completed a pilot study and its goal is to have a working version, Version 2.0, of the xLPR code ready for use later, late in 2013. The staff is requesting a letter from the Committee, hopefully favorable, providing technical feedback as well as an overall opinion of the work.

With that introduction I'd like to turn the presentation over to David Rudland.

MR. RUDLAND: Thank you. Again my name is

Dave Rudland and I am a senior materials engineer in the Office of Research, Division of Engineering, Component Integrity Branch. My branch chief is Al Csontos sitting back there, and actually for the next three months of my rotation to NRR/DE in the Piping and NDE Branch and Tim Lupold's my branch chief during my rotation.

Again as Tim pointed out, the purpose of the meeting is to give an overview of the Extremely Low Probability of Rupture Program. I'd like to try to achieve a common understanding of where we are in xLPR. What is it? What are the motivations? What's the status? Priority and our path forward. I want to get your review and advice on the project. And as Sam pointed out, I'd like to get a letter that talks about the efficacy of the project.

In talking with the subcommittee we also requested that we have a review and advice meeting with them at least once a year, giving them an update. And I think we'll probably do that more instead of on the project maybe on subtopics that are within xLPR, since xLPR is a very broad topic.

I'm going to start with a little background and the regulatory need for this project, why we're doing this. I'm going to go into a little bit about the plan and what we plan to do and what we have been doing in Version 1.0. I'm going to talk about the pilot study, which was a feasibility study on whether or not it is even feasible to do this type of project and give you the sample of those results.

I'm going to talk about our plans for Version 2.0. Our schedule and the path forward.

In 10CFR50 Appendix A, there's GDC-4 that allows the local dynamic effects associated with large break pipe rupture be excluded from the Design Basis if it can be demonstrated that those probabilities of rupture are extremely low.

And these local dynamic effects are things like pipe whip restraints and jet impingement shields that are there to protect the other equipment in cases possibly the break actually were to occur.

And this portion of the regulations allows the industry to remove these things, which is a great cost savings. And it allows for better inspections and things like that, so it's a great advantage to them to be able to remove that.

The staff developed a conservative, flaw-tolerance deterministic approach back in the 80s that demonstrated that the pipe would leak before a break. Using a set of safety factors and deterministic fracture mechanics. And at the time, when GDC-4 was revised, it was determined that this deterministic, conservative analysis would equate to an extremely low probability of rupture. It was only qualitative --

MEMBER SKILLMAN: Excuse me.

MR. RUDLAND: I'm sorry, yes.

MEMBER SKILLMAN: David, let me ask this question. When this project is complete would it be predictable that a utility that had been struggling with pipe break, perhaps on the secondary side, where the plant is really aged and the utility is reluctant to put in shields and wagon wheels and things like that? Would this final product enable that utility to use this tool to modify or exclude a piece of pipe or a location?

MR. RUDLAND: I think that the LBB applications are mainly for the primary system. But the code itself is going to be structured to be able to handle any type of location. And as long as the basis and technical background for the analysis is justified and approved by the NRC, yes, I don't see why not.

MEMBER SKILLMAN: So the answer is probably yes?

MR. RUDLAND: Yes.

MEMBER SHACK: But you're not planning on putting like flow-induced corrosion type thinning in it? You are?

MR. RUDLAND: Yes, sure. The code's going to be flexible enough to be able to handle that. If there's an application and a need there's no reason it couldn't.

Again, this initial study is focused just on the LBB plants that have been already approved, which I'll talk about in a second, and those mechanisms associated with that. There's no reason why, the way that we structured this code, that it can't handle those types of models.

So in talking about LBB, many of the PWRs have the reactor coolant piping approved for LBB. So they've been able to go in and remove those jet impingement shields. And some of the PWRs have the RCL branch piping also approved.

The first LBB approval occurred in about 1984, and again, they were done for mostly the PWRs. At that time IGSCC was active in the PWR heat effect zones for the stainless steel piping. And --

MEMBER SHACK: BWRs.

MR. RUDLAND: Did I say, I'm sorry, B, BWRs, I misspoke. BWRs. The SRP3.6.3 that was written stipulates no active degradation mechanisms. So those plants, in many cases, didn't even apply for LBB because they had an active degradation mechanism. But at the time the PWRs did not have anything active so they were able to apply and get approved for LBB.

However, we started seeing some cracking in the mid-90s in the PWRs, in Alloy 600 materials. And we had our first crack that was formed in a dissimilar metal weld, butt-weld, of hot leg to a reactor nozzle at V.C. Summer plant that actually was a leaking axial crack.

During that time period the staff had developed an LBB Reg Guide to handle the LBB issues. But it didn't include this active degradation because at the time the ones that had been approved didn't have any active degradation mechanisms.

So the work on the LBB Reg Guide was put on hold for the PWSCC issue. And then in 2009 we kicked off this xLPR Project because the NRC staff at the time decided to take a two-tier approach to handling the PWSCC problem. The industry embarked on a mitigation and inspection schedule that they had defined to help mitigate this problem and to inspect it.

And qualitatively that was acceptable to the NRC but in the long-term we wanted to quantitatively assess these probabilities of rupture. So xLPR was started, again, in 2009, conducted this pilot study, which I'll talk about in a second. And we plan to release the code itself, Version 2.0, applicable to LBB lines in 2013.

MEMBER BLEY: David, just to put this in perspective for me, since your first line up there was oil and gas, and since the PG&E San Bruno break occurred as a result of a pressure excursion, not some external attack on the pipe. How does that fit in with what they did, what you're doing? Have you looked at it enough to understand if that's a challenge or if there's some reason that's not applicable? Or it's about the right frequency?

MR. RUDLAND: Yes, typical LBB, especially the LBB that was done back in the oil and gas time was all deterministically based, right. So they used a rule of thumb safety factors. Consensus safety factors, in their design. And with that it doesn't exclude the probability of some event occurring. And yet, I'm not too familiar exactly with the specifics of what caused that failure, but as you point out most of the failures in oil and gas are third party damage type of --

MEMBER BLEY: Yes, but this wasn't apparently.

MR. RUDLAND: The probability of that stuff is always there, depending on the conditions that they had. And this ensures enough reasonable assurance --

MEMBER BLEY: But it hasn't drawn your attention to something that you really need to understand to see if --

MR. RUDLAND: No because the conditions within a nuclear plant are a lot different than that in the oil and gas.

MEMBER BLEY: Yes, indeed they are, but --

(Off the record comments.)

MR. RUDLAND: What it does is it allows us to calibrate our probabilistic models with actual operating experience, which we don't have a lot of in the nuclear plants because there has been --

MEMBER BLEY: But they've got an awful lot, so maybe it's in line.

CHAIR ARMIJO: Well the only things we've actually ruptured have been flow-accelerated corrosion pipe.

MR. RUDLAND: That's right.

MEMBER BLEY: That's true.

CHAIR ARMIJO: -- anything else has leaked.

MR. RUDLAND: Been leaking. Again, like I pointed out, PWSCC emergent issue really hit the butt welds of the large reactor nozzles in the early 2000s. And through MRP-139, which was an industry document that dictated a inspection and mitigation plan, they set forth to mitigate and decrease the inspection interval on all of the metal welds in PWRs that were associated with the LBB lines.

MEMBER SHACK: Do you what fraction of PWRs actually have nickle welds?

MR. RUDLAND: Of the PWRs?

MEMBER SHACK: Yes.

MR. RUDLAND: Oh my gosh, I think it's a very high percentage. I don't think there's very many that don't. The ones that do use stainless steel.

MEMBER SHACK: Yes, I mean there are some that are all stainless.

MR. RUDLAND: Yes there are some, but not very many.

MEMBER SHACK: But you think that the overwhelming majority use --

MR. RUDLAND: Yes, we have a database actually put together of the LBB that has plant-by-plant and the lines that have been approved in the DM welds. I don't recall off the top of my head, but it's most of them. Yes, it's most of them.

That industry initiative, MRP-139, has been rolled into Code Case 770 through the ASME Code and incorporated into 50.55(a). So it's within that time period, shown here on this flowchart, that an inspection mitigation short-term effort has been set into regulations. Within that also is when xLPR will be developed fully with, again, a hope that we can attack the regulation on a more long-term aspect in around 2015, after we finish Version 2.0 and write the basis for that to change the regulation.

CHAIR ARMIJO: Now, David, the industry people don't have to apply any kind of mitigation unless they've detected something that's indicative of a flaw or --

MR. RUDLAND: That's not true.

CHAIR ARMIJO: They have to apply --

MR. RUDLAND: Yes, they have to follow the inspection, they have to follow this 50.55(a), Cold Case 770 for even the unmitigated cases. If it severely hinders them and lowers their inspection intervals if they don't do something.

CHAIR ARMIJO: Oh, okay. All right.

MR. RUDLAND: So they have to do this. So again, in the long term we want to do this in a quantitative way, by developing a probabilistic assessment tool that we can use to directly assess compliance with the Regulation.

And, to Bill's point, what we want to do is we want to make this code as flexible and adaptable as possible so that the money we spend here is not only applicable to LBB lines, but is also applicable to all range of problems associated with the reactor coolant system.

CHAIR ARMIJO: That would include if you have a different failure mechanism, IGSCC, flow accelerated corrosion?

MR. RUDLAND: Absolutely.

CHAIR ARMIJO: But you'd have to develop a module to deal with that?

MR. RUDLAND: That's right. Yes, the model and the mechanisms has to be well understood and characterized in order to be able to incorporate it into the code. So we tried to develop a framework that will do that. We've also tried to develop a framework and process that is fully vetted within the technical community by the subject matter experts.

We're doing this in a cooperative effort.

We've decided to kick off a cooperative effort EPRI to do this so that we could share in ideas and in resources in that they have some resources that are very valuable to the program and so does the NRC.

So through a memorandum of understanding we have developing, we've become in this effort. And we put the program together such that it is equally staffed by folks from either NRC and their contractors or EPRI and their contractors. So we have equal representation across the board on all of the working groups.

The thought of developing this type of code is very difficult. It's a challenging, it's a daunting task to be able to do this, when we kicked it off in 2009. So we thought the best way to handle that was to go ahead and do a feasibility study, a pilot study, to assess how effective it is to not only develop this kind of code but work in the environment that we want to work in. And make sure that we can do it efficiently.

CHAIR ARMIJO: David, I jumped ahead because you talked about this MOU with EPRI. And you have a slide showing all the organizations that are contributing or working in some way, which includes National Laboratories, Exelon, Areva, Westinghouse.

Are these people really working on this thing --

MR. RUDLAND: Oh yes.

CHAIR ARMIJO: Or is this just labels?

MR. RUDLAND: Yes, like this box here that I have on this thing shows the different groups. And so for instance the models group is broken down into several subgroups. And the subgroups represent some of the major fracture mechanics and technical based applications, crack initiations and crack growth. And then the subject matter experts in there, each of those teams have a couple of different, sometimes up to even a dozen different people, working on that particular problem.

So all those people that are on that list, that we'll get to here in a second but that Sam was talking about, are being funded through this project by one means or another. And are working on the project.

CHAIR ARMIJO: Either by EPRI, their own parent company or by NRC?

MR. RUDLAND: That's correct.

CHAIR ARMIJO: And is NRC the lead organization? That's where I want to get you? Or is it sort of like --

MR. RUDLAND: It's a consensus, so we're

splitting. So the top level development effort is being led by one NRC staff member and one EPRI staff member.

CHAIR ARMIJO: Okay, I understand.

MEMBER REMPE: Will there be additional validation efforts after this is all done?

MR. RUDLAND: Yes, we have a very extensive validation effort that we plan to go through. A verification and validation effort for the code. We're developing this thing under a very strict QA program and part of that is to develop a V&V plan.

MEMBER REMPE: New data or existing data?

MR. RUDLAND: Whatever data we have. Operational data or laboratory data. Depending on the models that we're talking about.

MEMBER REMPE: But no new data obtained, existing data?

MR. RUDLAND: We are developing some new data.

MEMBER REMPE: Oh, okay.

MR. RUDLAND: Yes, so for instance there's some stability issues that we have in complex cracks in dissimilar metal welds and how they fail. So we're actually doing laboratory experiments. And they're developing laboratory experiments on crack growth rate data for some of the Alloy-52 materials, and 152 materials, the higher chrome content that are more resistant to PWSCC. So it's all happening within the auspice of this xLPR effort.

CHAIR ARMIJO: David, when you --

MEMBER SCHULTZ: It's a software QA program as well as a V&V program?

MR. RUDLAND: Yes.

MEMBER SCHULTZ: Thank you.

MR. RUDLAND: And I'll talk a little bit about that towards the end of the presentation.

MEMBER POWERS: When you talk about having data and rigorous QA program, do you subject those data to the kind of review that one does in connection with experimental results used for thermal hydraulics codes, like what do they call it, the I want to say CSARP but it's not CSARP.

(Simultaneous speaking)

MEMBER BLEY: CSAU.

MEMBER POWERS: Yes, CSAU, kind of review where you look at, not only uncertainties in the data, but the applicability of that data to the environment that you're considering?

MR. RUDLAND: Well we tried to consider all that. You know, the problem with the data that we have and the data that we're generating is it's not necessarily always sufficient to be able to fully understand the uncertainties and their impact. And so sometimes judgement calls have to be made. Engineering judgement calls have to be made on the type of uncertainty that is and its impact. And how it can be categorized, but we're attempting to do that, yes.

MEMBER POWERS: Well I mean it seems to me like the CSAU methodology sets up a framework that's very rigorous for that, why don't you just use their framework?

MR. RUDLAND: Well it's something that we'll looking into. What was that again, sorry?

MALE PARTICIPANT: CSAU. Code scaling analysis uncertainty.

MR. RUDLAND: Code scaling analysis of uncertainty.

MEMBER POWERS: And it's applicability.

MEMBER BANERJEE: Applicability.

MEMBER POWERS: It is something that is designed primarily with thermal-hydraulic issues in mind say mutatis mutandis, but still it's a fairly rigorous kind of an approach. And I mean one of the biggest dangers in developing any kind of a phenomenological code is treat data as though they were points of universal validity. And when the code doesn't pass through those points exactly there must be something wrong with the code.

That will frustrate you to death. And the CSAU, I mean, it's, to say the least, the people who developed it were anally retentive. But it is the most rigorous formulation that I can think of for doing these things.

MR. RUDLAND: Yes, I will definitely look into it. Thank you.

MR. CSONTOS: One point onto Sam's question earlier. We are working crossover here cooperatively with EPRI but we are independent and we're separate from when we're developing this program there are two leads, one from EPRI one NRC, because we are independent yet we cooperate in development so it's an MOU.

CHAIR ARMIJO: The reason I was asking the question is if EPRI decides to disengage or drop or reduce their effort and reduce their funding that leaves you guys hanging out to dry?

MR. RUDLAND: Yes, that's right.

CHAIR ARMIJO: But so far you haven't seen any indication of that?

MR. CSONTOS: Yes, and like Dave was saying, there are lots of data and inputs and other models and et cetera that EPRI does have that comes into it that's very valuable.

CHAIR ARMIJO: This is something that they already have?

MR. CSONTOS: They already have it, yes. For example plant specific data, we don't have some of that information.

MR. RUDLAND: And they are rather motivated to stay involved also. To quickly go through the process, this is nothing revolutionary or new here. But for the process we take all of the variables that are uncertain and we try and to characterize that and classify those uncertainties as either irreducible or as uncertainties that can be associated with lack of knowledge.

And we stick those through a stochastic or Monte Carlo type of technique and develop a failure frequency. The failure frequency is a distribution in itself. So at a particular time we can know what the average failure of frequency is and where we sit in terms of the 95th percentile.

CHAIR ARMIJO: This is the approach of a PTF approach?

MR. RUDLAND: Right, this is an approach of probabilistic fracture mechanics. On any of those types of things, this is the same way that it works. And then through improved knowledge we may not be able to change what the average failure frequency is but we can have better confidence and we can have a smaller amount of uncertainty along that particular failure frequency.

And so one of the goals of this program is not only to be able to do this but to be able to point to the places where we can continue research and improve our understanding so that we can reduce those types of uncertainties.

As Sam pointed out that was kind of general to most PFMs. So specific for xLPR we look at a weld-by-weld or joint-by-joint basis. And through inputs that are random in nature, we then have initiation models that will initiate flaws, cracks, within the piping. Those initiations may be single initiations, they may be multiple initiations.

We then allow them to grow by whatever the mechanisms are that are active in the particular analysis. Those cracks may coalesce, may join up, making longer cracks.

MEMBER BANERJEE: Are there good models

for this or is there a lot of model uncertainty?

MR. RUDLAND: It depends. For instance crack initiation there's a lot of model uncertainty, yes. We know a lot more, much more confident on crack growth than we do on crack initiation. And things like that.

MEMBER BANERJEE: But do they branch in things or what happens?

MR. RUDLAND: They do. They do, PWSCC is a very branching type of mechanism.

MEMBER BANERJEE: So do you have good models for this?

MR. RUDLAND: We have adequate models for this. And we have adequate determination of the uncertainties. What it doesn't do, it really doesn't allow us to be able to look at out of plane growth and all that kind of stuff. Now the cracks, the drivers of these types of cracks, the PWSCC cracks at least, are very stress driven. All right, if the environment is appropriate and the materials appropriate then it's going to go where the stresses are.

And if we're able to predict well enough where the stresses are then we can pick the path appropriately or pick the stress appropriately to estimate the crack growth lengths.

CHAIR ARMIJO: David, the initiation step is probably the weakest part of any of these analysis. What kind of data do you have on PWSCC initiation? Laboratory test data or --

MR. RUDLAND: There's a couple of sets of test data, a couple of different types of data that are out there. One are laboratory data, and they're very simple tests done in an environment where a constant load is put on or a constant stress is put on and time is marked is when they start to see flaws. It's pretty simple models.

Then there's operational history. So we have a bunch of operational histories.

CHAIR ARMIJO: The reason I was getting at it is when I worked on BWR ISGCC many years ago and initiations was everything with that mechanism. And what we did to develop improved materials, more resistant materials, we looked at the operational experience and found what had seemed to be a cause and then we reproduced that in the laboratory. And the worst things you could do, of course, create a high residual ID stress on these weldments and the way to do that was very poor welding practice, very high input welds.

But probably the worst thing you could do, we could crack almost anything by doing what we called abusive ID grinding. So post-weld grinding, which a lot of welders like to do because that way they can pass their x-ray requirements. But they leave the material looking beautiful but totally susceptible to stress corrosion cracking.

And what I was trying to get at is will you address those fabrication issues in your initiation model? Because I think that's what makes a huge difference. I don't know if it makes much difference on PWSCC but it's day and night with IGSCC.

MR. RUDLAND: I suspect that it will because it's basically the same mechanism.

CHAIR ARMIJO: It creates a surface layer that's got enormous residual tensile stresses and the cracks initiate there.

MR. RUDLAND: What we did in the pilot study in Version 1.0 was that we took an empirically driven relationship and fit it to the operational history. We didn't really use the laboratory data. There's a slight disconnect that we're seeing between the laboratory data and the operational data. So we took the operational data and we fit that to an empirical relationship for Version 1.0, for the feasibility study.

For Version 2.0 what we've done is we've actually kicked off a PWSCC expert panel to try to figure out what's the best way to handle this. And it's funny that you say that because at the first meeting of our expert panel that's one of the first things the experts said was, you know, because of these surface stresses and the surface treatments, it's difficult to be able to really then characterize what the stresses are at the ID surface.

So it's probably best to take stress out of the initiation and just fit it to the operational data and then you have what you have, because you don't know what these heavy stresses really are in a lot of cases. And then again, that panel met once, they're meeting again several times before they make a recommendation on how we should move forward on Version 2.0. Your points are very well taken and --

CHAIR ARMIJO: Well you know laboratory work can clear up a lot of these questions but those are expensive kind of tests.

MR. RUDLAND: They are.

MEMBER BANERJEE: So the cracks mainly propagate due to stress, the stress field?

MR. RUDLAND: Under this type of cracking, this intergranular cracking yes. It's a combination of environment, which is attacking the material, and loss of chrome. And then the crack is driven by the stress through the wall thickness intergranularly.

MEMBER BANERJEE: Stress and temperature?

MR. RUDLAND: Stress and temperature, correct, yes. They're stress and temperature activated.

MEMBER BANERJEE: So you have a relatively simple model?

MR. RUDLAND: It's relatively simple, yes. Especially the empirical model for the initiation is very simple, very simple model that we're able to calibrate and then develop uncertainties by the operational data. But again, the problem again with that is that the operational data there hasn't been very much. Especially in terms of the lines we're looking at, there hasn't been very much. So we have to use other data making approximations on that also.

The code also then goes through and allows for the application of inspection, through inspection intervals and mitigations. If anything happens to get through and leak, becomes a through-wall crack, the code calculates when that occurs and calculates a leak rate. Keeps track of the leak rate as the crack is larger. If it isn't removed it may fail so there's a stability check. If a particular run, a particular pipe, does fail then code is exited, if not it continues along through the life cycle of that.

So it's a time-based flow problem we allow the program to go until something fails and then we correct for things like leakage detection and inspection and things like that after the run is complete.

MR. CSONTOS: And just to go to the question of models, a lot of these models that have been developed, except for the initiation, have been used a lot in our deterministic evaluations of plant operational issues during the outage season when we see facts. Dave fills out and runs those calcs. So those deterministic pieces have been brought into this so they've been vetted in the regulatory arena prior to being in here.

MR. RUDLAND: Taking that same chart I just showed and putting into more of a flow chart. The purple boxes represent the deterministic fracture mechanics based modules. And the way the code is structured is that those modules are self-contained. They are verified and validated separately and compiled and plugged into the framework at a later time.

So these are actually self-contained and V&V'd modules that are linked together. In the pilot study we chose to do a double-nested loop so that we could separate out the aleatory uncertainty, or the uncertainty that is random. And against that that is the lack of knowledge type of uncertainty. And we do that so that we can understand the effects. Because we know if we can reduce that epistemic uncertainty we want to know how much that actually effects the total uncertainty in the problem.

MEMBER POWERS: Can I understand why the crack growth module feeds into crack initiation module?

MR. RUDLAND: Because in some cases we have pre-existing defects. And so what we do is we allow a crack growth increment to occur from a pre-existing defect, or from a defect that existed in the prior loop of that. We want to make sure we get that crack growth before we check for the next crack.

CHAIR ARMIJO: So crack initiation here means pre-existing mechanical cracks?

MR. RUDLAND: No. In this case the purple module is actually initiating a crack due to a mechanism. Pre-existing defects would occur outside of this and during the sampling phase. A pre-existing defect would come in. This is actually the time loop flowchart. So anything that occurs outside of that is not show on this particular flowchart.

MEMBER POWERS: So what you're saying is in case time-step you grow the cracks, then you check to see if you've initiated new cracks?

MR. RUDLAND: That's correct.

MEMBER POWERS: Okay. And then you check to see if you've coalescence from the existing?

MR. RUDLAND: That's correct.

MEMBER POWERS: A wise code developer of my acquaintance, which seems like a contradiction in terms, once told me that the interphase between two models is itself a model. And it tends to the hardest one to validate. Do you have that problem here?

MR. RUDLAND: Maybe, but the time steps are not very large. The time steps are relatively small over the course of a life of a plant, the time steps are small. So I think that type of uncertainty is not very large.

MEMBER POWERS: I think what he was speaking to is the differences in phenomenology between two models themselves provide boundary conditions for the next model and that itself is a model. And the problem is that frequently it's quite impossible to do a test that actually tests the interface. Or you wouldn't put a model in there.

MR. RUDLAND: That's right. Yes, and for instance correct initiation and growth in a model, I guess, could be the correlation between those two. I mean sometimes that's difficult to understand whether or not they're correlated or not. And that's something we're actually looking into. And hopefully from some experiments we can determine what that correlation is or use it's sensitivity studies to determine its actual impact on the probabilities at the end.

How are we going to use xLPR? Well, again, like I mentioned at the end of the day we plan to get a failure frequency per year for a particular problem. And that failure frequency then will be set on a run for typical parameters, for a problem. Then we will do something to it. In this case we'll put on a optimized weld overlay over the top of the dissimilar metal weld attempting to mitigate the PWSCC.

And that will probably shift the probabilities to the left. It could possibly also reduce the amount of uncertainty on that particular set of probabilities. And then, at the end of the day, we have to develop a criteria to set up whether or not a shift is acceptable, how much of a shift is acceptable or what the level of probabilities are that are acceptable.

So we actually have a group within the program that's trying to develop the acceptability criteria for these types of results.

CHAIR ARMIJO: Within your code could you look separately, after you put that overlay on, could you look separately to see if you've got through-wall compressive stresses on all parts of that structure?

MR. RUDLAND: Well that's a model. That's a model output, so the model will tell us that.

CHAIR ARMIJO: Because if that happens then the game's over.

MR. RUDLAND: Yes, there's always a probability that it's not effective. And so have to turn --

CHAIR ARMIJO: Do you have something in your model that says --

MR. RUDLAND: Yes, something didn't happen right. Yes, that's right.

CHAIR ARMIJO: Okay.

MR. RUDLAND: And that's really what drives those numbers, because those numbers are relatively small.

CHAIR ARMIJO: Yes.

MR. RUDLAND: So this is the graph that Sam was talking about earlier, just showing the team members. These are the team members for the pilot study in Version 1.0 development. We actually have more folks working on Version 2.0. We've actually expanded the models group significantly. And there's more of the EPRI folks involved than were involved in the first version.

CHAIR ARMIJO: A lot.

MR. RUDLAND: Yes, we have a lot of people involved in working on this project.

MEMBER POWERS: There's a wise old saying about too many cooks.

MR. RUDLAND: We only have one or two cooks. And we've got a lot of, you know, vegetables.

MEMBER SCHULTZ: There's also some repetition.

MR. RUDLAND: There are some repetition, you see my name up there a couple of times actually. But we've really developed a very good working relationship. We've developed a consensus-based decision making document so that can orderly discuss technical details and make decisions on a relatively easy scale.

We have a kind of like a differing professional opinion procedure if we actually have somebody that vehemently disagrees with what we're doing, they have a way of documenting that and taking that through the approval process.

So we've spent a lot of time developing the structure behind this program. And it's worked over the last two years, been very efficient and worked very well. And the cooperative effort has been amazing, I think, in how well we've worked together.

So this pilot study. Again, we wanted to do a feasibility study and we actually had three main goals. We wanted to take a look at this structure that I just talked about. As a structure, can it work? I mean when you get 70 people working on a project together can you actually develop a workable tool?

MEMBER POWERS: No.

MR. RUDLAND: That's a good question.

MEMBER POWERS: The answer to that is no.

MR. RUDLAND: I think we were able to do it relatively well. We also needed to determine what the appropriate probabilistic framework was. There was some question about which way we should go so we wanted to look at that. And then we also wanted to know if its even feasible to develop the type of code that we want. A modular based code for probabilistic fracture mechanics.

The pilot study was focused only on dissimilar metal weld in a surge nozzle, pressurizer to surge nozzle dissimilar metal weld. We chose that because when we started this project that had just finished up a pretty extensive effort looking at some flaws and failure analyses for Wolf Creek Power Plant, and the industry had developed an MRP-216 document that had loads and geometry and a lot of things for the pressurized to surge nozzles, so we thought that was a good place to start.

CHAIR ARMIJO: If I remember right the Wolf Creek indications came out, they weren't actual leaks were they?

MR. RUDLAND: No, they were just indications. They were circumferential indications.

CHAIR ARMIJO: So you never could confirm whether those were stress corrosion cracks because they went and overlaid the suckers and moved on.

MR. RUDLAND: That's correct.

CHAIR ARMIJO: So you didn't know it was a preexisting fabrication defect or a stress corrosion crack?

MR. RUDLAND: But what we did was we assumed it was PWSCC and we did a bunch of analyses and developed a bunch of data and pulled a bunch of things together for the nine plants that were being affected by the outcome of those analyses. And then we used that data in xLPR is what we did.

CHAIR ARMIJO: Okay.

MR. RUDLAND: So yes, they never took up boat samples, they never checked to if it was PWSCC.

For the framework, what we did was we had some differing opinions on how we should move forward with the framework. There was some that thought we should go fully open-source, that everything from all the way down to the random number generator should be developed with an open-source, and maintained that way through the life cycle of the software.

And there were others that thought well we could leverage some commercial software that's out there that does that kind of stuff and keeps QA and control and versioning of that separate from xLPR. So that's not a burden on either EPRI or the NRC as we go through the life cycle of the software.

So what we did in the pilot study is we decided to just develop them both in a rougher manner.

In more of a pilot study effort and not a full developmental effort. At the end of the day what we did is we used the same deterministic modules in each one of these codes.

So the purple boxes from a few slides before, we used those same modules in each one of these frameworks. And the only thing the frameworks did was control the flow of the code, the time of the code, the sampling and the data reduction at the end. And the codes, at the end of day, give the same results.

But one of the things we wanted to do was we wanted to investigate this and determine how we were going to chose what the proper framework was. And I'll talk about that choice here in a little bit. But we have two distinct frameworks that we developed in the pilot study.

Uncertainties. We already talked some about uncertainties. And we struggled a lot in the pilot study because we really didn't -- the pilot study, again, was just a feasibility study and we weren't focused on trying to get the right absolute values for the answers.

And so what we did was we made some choices. The models and inputs groups came to a consensus on those uncertainties. It was satisfactory for the pilot study but I think it needs to be further investigated. And we are further investigating it in Version 2.0, on how to choose and classify and propagate those uncertainties.

So these are the choices that we chose for Version 1.0. Some of the things like crack initiation was chosen as an epistemic uncertainty and we'll see in a second that that made a huge impact on the results.

We used things such as importance sampling and Latin Hypercube and Monte Carlo analyses and discrete probability distributions, all in an effort to see which ones seems to work best and which ones gave us the best feasibility of being able to do reasonably timed runs for this particular code.

MEMBER POWERS: I don't understand your choice of Latin Hypercube sampling. Artificially narrows your uncertainty distribution.

MR. RUDLAND: Latin Hypercube?

MEMBER POWERS: Yes.

MR. RUDLAND: Well Latin Hypercube also give you a better sampling across the whole distribution. A more uniform sampling across the whole distribution. I think that was why it was chosen for the epistemic uncertainties.

MEMBER POWERS: Yes, but what you're interested in is what the breadth is. And you've artificially narrowed it. It will not convert to the true distribution.

MR. RUDLAND: Well, what are you worried about? The tails, is that the concern?

MEMBER POWERS: Sure.

MR. RUDLAND: Well that's why we used the importance sampling. So what we did was we ran Latin Hypercube and Aleator and Monte Carlo first to determine where the issues were and then we importance-sampled on those variables that were driving the problem to catch those tails.

MEMBER POWERS: Why didn't you just use Monte Carlo on both?

MR. RUDLAND: That time is very long.

CHAIR ARMIJO: Let him get to the next slide and we'll some interesting --

MEMBER POWERS: You probably don't need any more samples for Monte Carlo than you do for Latin Hypercube. How many are using on Latin Hypercube?

MR. RUDLAND: How many did we use? We had, for most of the runs, we had in the order of about a 1,000 --

MEMBER POWERS: Oh Lord. If you've got that many samples you should not be using Latin Hypercube.

MR. RUDLAND: Okay. I will take that back to the computational group, which is being run by Sandia by the way.

MEMBER POWERS: Well you tell us they're being run by the guy that's apostle of Latin Hypercube sampling. Yes, with 1,000, I mean --

MR. RUDLAND: Well the 1,000 was actually done on Monte Carlo and we did 50 on --

MEMBER POWERS: Yes, 50 is enough. You can use Monte Carlo. The other thing you're gaining is a picture of the optics of where the samples are and you're losing badly on things like simple statistical interpretation.

MR. RUDLAND: I'm going to get you and Helton in the room.

MEMBER POWERS: And we could fight as much as we do at Sandia.

MR. RUDLAND: As you'll notice when showed some of the results we couldn't do it without importance sampling or some kind of adaptive type of sampling anyway, because the tails are driving the problem, of course.

MEMBER POWERS: Yes, and your Latin Hypercube inevitably will narrow those. Now maybe you can live with the narrowing but you're going to have a problem with every time you present by saying you're narrowing your distributions artificially here. And why would you want to do that? If the tails are in fact what are important to you. You've got more than, it has been a very long time since I have had 50 in a sample.

And your sampling uncertainty is not relieved by doing the Latin Hypercube sampling. It is exactly the same. So you haven't gained anything there. And you're losing a lot. The other question is how are you handling the correlation among your parametric uncertain quantities?

MEMBER SCHULTZ: It would be neat if he flipped up the next slide. Flip up the next slide, David. This is really curious, Dana, if you look at these results. Especially if you know which parameters are being run with the Latin Hypercube and which ones are being run with the Monte Carlo and why you get these wildly different shapes of the uncertainty distributions by changing one parameter from epistemic to aleatory. Look at those distributions and now tell me what's going on.

MEMBER POWERS: There is only one that's important.

(Simultaneous speaking)

MR. RUDLAND: But to get back to your question. So your question is how do we handle the correlations. That's a good question. And I don't know if I can answer that or if I'm the right person to answer that, but we can get that answer for you. The input parameters are correlated and we're able to sample them based on the correlation factors. But we can get those answers for you.

In terms of this slide, this is some of the results out of the pilot study. And as you've probably noticed, on the left-hand side the gray lines, going up and down, are the individual epistemic realizations. And you can see that there isn't much variation in those at all, they're pretty vertical.

So there isn't much in there that's really effecting, in one particular epistemic realization, there's not much effect of the aleatory uncertainty. The red line is the mean line and the green line that you see there is the 95th percentile.

When you take the crack initiation and drop it over into the aleatory now you see that those gray lines are now folding over, because you've got a lot more effect of that uncertainty. And that the crack initiation is really driving that problem.

Doesn't change the mean, the mean's the same. But 95th percentile is extremely different. So the problem is really being driven by the crack initiation. And not only that but there's something else because there's still a spread across the data, going from left to right, which is telling me that I still have some epistemic uncertainty that's driving the problem. But in terms of the quantiles, especially, crack initiation is driving the problem.

MEMBER SCHULTZ: Dana, do you have any idea why that behavior would change so much if you're doing just a sampling algorithm and have enough samples? Why those shapes would change if you're doing --

MEMBER POWERS: I would send you to Helton to understand.

MEMBER SCHULTZ: I have no idea.

MR. RUDLAND: I think the reason is because on the left-hand side, in one of those epistemic realizations, there is no aleatory uncertainty. So it's either happening or it isn't. You're either getting rupture or you're not. There's no uncertainty about it. That's why the lines are vertical.

But when you go to this plot over here, on the left-hand side, it's not vertical anymore. Now there some effect of that aleatory uncertainty.

MEMBER BLEY: But that's the extremes. The means ride the same in both of them and the bulk of the distribution is the same. You just get these extremes which come from the flip-flop on the yes/no epistemic, that you always see.

MR. RUDLAND: There's an epistemic uncertainty --

MEMBER BLEY: But the bulk of that hasn't changed. I mean it's just --

MR. RUDLAND: The spread going across, the difference between the mean and the 95th percentile, each of those is really representing that epistemic uncertainty. So we still have something very large in there, epistemically, that's driving this distribution.

MEMBER BLEY: And you can see that epistemic glitch, over and over and over and over again, all the way through.

MR. RUDLAND: That's right.

MEMBER SCHULTZ: I mean that's all it was, is just --

MR. RUDLAND: So let's just look at the mean results for a bit here now since we just talked a little about the distribution. This is one of the example results, just showing the effects of some of the inspection and mitigation techniques. This blue line is a case for a Westinghouse-type pressurized surge nozzle, dissimilar weld with a stainless steel safe vent close by. With no inspections, no leak detection and no mitigation.

So we have a cumulative probability of rupture after 60 years of about five times 10^{-3}. We then ran a case with mitigation, mechanical stress improvement type of mitigation, at 20 years. Reduced the cumulative probability of rupture by about two orders of magnitude.

We then went back to the original and a ten-year ISI and a one gallon per minute leak detection limit. And reduced it from, again, five times 10^{-3} down to about five times 10^{-6}. If I combined all three of them I've got this orange line down at the bottom that's pretty close to one times 10^{-9}.

MEMBER SKILLMAN: David, what is the mechanical mitigation that is your red line? Physically what is mechanical mitigation?

MR. RUDLAND: Mechanical stress improvement is a clamping that's done remote from the weld of interest. So about a pipe diameter away or so from the weld of interest, they take a clamp and they squeeze the pipe and plastically deform it. It imparts a bending stress back to the dissimilar weld which puts the ID in compression.

MEMBER SKILLMAN: I see. So it's a hoop stress --

MR. RUDLAND: Yes, actually it's both. It's a hoop stress under there but it actually changes the axial stress at the weld.

MEMBER BROWN: So you increase the stress at one part to increase the clamping stress, or the --

MR. RUDLAND: That's right, basically put a bend in it, it's like you're taking the pipe and you're bending it, right.

CHAIR ARMIJO: You wind up with tensile stresses somewhere else but not in --

MEMBER SHACK: It's puts the tensile stresses away from the weld.

MR. RUDLAND: You put the tensile stresses where it's not susceptible.

CHAIR ARMIJO: Squeezed out, stretched out.

MEMBER SKILLMAN: Like a Weiner-dog balloon.

(Simultaneous speaking)

MR. RUDLAND: Non-susceptible material. It's about a one percent strain.

MEMBER BROWN: Do you also get wall thinning?

MR. RUDLAND: You'll get wall thinning. What you do is you get almost like a denting in the pipe, so the pipe is -- Yes, so it's by one percent.

MEMBER SIEBER: The clamp stays on forever?

MR. RUDLAND: No, it's plastically deformed. So all you got is the plastic deformation that's causing the reduction in the stress. Turns out to be elastic back at the weld that we're talking about, but it's plastic under the clamping device.

MEMBER BROWN: So it stays the way it looks? It stays that way.

MR. RUDLAND: Yes, it stays that way. And if you run your finger across it you can feel it.

MEMBER BROWN: It has to stay that way.

CHAIR ARMIJO: Why go through these things when you can just do this big weld overlay? And that does --

MR. RUDLAND: The industry has choices due to money, accessibility. In order to access this it's tough, because they've got to clear away whatever, sometimes they're working in the sand boxes to get the clamps down in there. And I went and watched a demo and the guy that's putting stuff on has actually got to hang by his feet to get in there to put the shims in so that they get the right amount of clamping going on. So it's not an easy thing to do. And so they have choices.

CHAIR ARMIJO: It's easier than weld overlay.

MR. RUDLAND: It's easier than weld overlay. It's a lot less I think in terms of the time.

MEMBER POWERS: Do they charge money to watch for this? Do you think some of that's spread in the curves you showed, that series of curves is, just due to the geometry of the crack consolidation?

MR. RUDLAND: Could be. What it is, it's more in the stresses that are driving it than anything else. There's a big uncertainty in the stresses. And then since the uncertainty carries through the entire project, the entire run.

MEMBER ABDEL-KHALIK: How did the ten-year in service inspection impact the mean probability of rupture?

MR. RUDLAND: I don't have that separated out in this plot. But for this mechanism the growth rate is very high and so the effects are very small. So for it to get about order and a half magnitude drop in cumulative probability you've got to go down to about a two-year ISI.

CHAIR ARMIJO: I'm trying to understand that. When you do the ISI you can detect these things --

MR. RUDLAND: Yes, the probability of detecting them.

CHAIR ARMIJO: Probability of detecting goes up and so at that point you do something about it, right?

MR. RUDLAND: Do something about it, that's right.

CHAIR ARMIJO: You do the --

MR. RUDLAND: In the pilot study we took it out and said that it's done. It's not going to ever get worse, the crack is gone and the pipe has got non-susceptible material and doesn't crack. That what we did in the pilot study for a crack that was found by inspection.

MEMBER ABDEL-KHALIK: Oh, I see. You catch it before it's detectible then you assume that it's been mitigated --

MR. RUDLAND: Right, in Version 1.0. In Version 2.0 we're doing something a little different, but in Version 1.0 that's what we did.

MEMBER ABDEL-KHALIK: But in reality though it doesn't, right? It doesn't necessarily have an impact.

MR. RUDLAND: Yes.

MEMBER ABDEL-KHALIK: You're not going to be able to mitigate all the cracks.

MR. RUDLAND: They do. I mean, if they find a crack now they mitigate it. Especially in one of these dissimilar metal welds they have to, yes. And most of time what they do is they either shorten the inspection interval and keep and eye on it, but I don't think there's ever been one left in service, Tim.

MR. LUPOLD: We've never left one in this country in service.

MEMBER ABDEL-KHALIK: Your name and speak in the microphone.

MR. LUPOLD: There's never been a PWSCC crack left in service in this country that we know about. Now there has been in Europe.

MR. RUDLAND: So even if an indication is found in these dissimilar welds they don't even wait to make sure it is PWSCC, they --

MEMBER ABDEL-KHALIK: They just measure it.

CHAIR ARMIJO: Just put it to bed.

MEMBER SCHULTZ: You didn't show, for this presentation, the separate slides. And during the subcommittee meeting there was a question about if you look individually at the effects from inspection or leak detection each of those individually seem rather modest. This shows the composite effect, roughly a factor of 1,000 in this example. But there were more dramatic examples. Do you yet have a feel for why that compounds --

MR. RUDLAND: Did you get my write-up I wrote up?

MEMBER SCHULTZ: I did not.

MR. RUDLAND: I did, I sent the write-up, oh yes, it was like a three or page write-up.

(Simultaneous speaking)

MR. BROWN: It was attached to the meeting minutes.

MEMBER SCHULTZ: Okay, I didn't get the

meeting minutes.

MR. RUDLAND: Let me see if I can summarize quickly for you.

MEMBER SCHULTZ: That's okay if we have it I can read it, in the interest of time.

MR. RUDLAND: Yes, basically it's the difference between dependant and independent inspection. So if you find something or you don't find something, you equal probability of finding, or a better probability of finding it the next time. And those things begin to add up and that's why you'll end up with an effect that's a lot bigger than the individual effect.

MEMBER SCHULTZ: I'll get it. I'm sorry, I'll get it.

MR. RUDLAND: All right, and if there's any further questions.

MEMBER SCHULTZ: I either got it and ignored it or -- Thanks.

MR. RUDLAND: I didn't hear anything so I thought everybody was satisfied with my explanation.

MR. RUDLAND: How many other welds are there in the coolant system --

(Simultaneous speaking)

CHAIR ARMIJO: Let's try and have one

meeting, please.

MR. RUDLAND: How many are there?

MEMBER SIEBER: Yes. I would imagine there's a lot.

MR. RUDLAND: There's not as many as you think. You know, anytime that the reactor is weld to a stainless steel pipe it happens. So the hot leg/cold leg areas or to the --

MEMBER SIEBER: That's carbon steel to stainless?

MR. RUDLAND: That's right. So they use this incanel weld to combine those.

MEMBER SIEBER: Incanel is the bad actor in every one?

MR. RUDLAND: Right. So like the pressurizer also. There are -- That's right.

MEMBER SIEBER: That's the alloy? Okay.

MR. RUDLAND: Now at the reactor coolant pump they also have it there where it goes, sometimes, from the carbon steel to the cast housing of the reactor coolant pump also. Yes.

MEMBER ABDEL-KHALIK: I have a similar question about the one gallon per minute leak detection system. Okay, so you have a leak of that magnitude. You fix it. This scenario has more than one leak that gets fixed during this 60-year period or just one?

MR. RUDLAND: No, just one. Any single realization, a single run through the time loop, if a leak is found it's repaired and you're done.

MEMBER ABDEL-KHALIK: No, but you continue --

MR. RUDLAND: Again, it's assumed that it becomes mitigated after the leak is detected.

CHAIR ARMIJO: But, Dave, the shape of your curve, that dotted blue line, the way I read it, maybe I'm misreading it, is that the one gpm leak was found when it finally leveled off and from then on it didn't grow?

MR. RUDLAND: No, and they're found anytime throughout the life. Again, this is a cumulative probability. So you've got things adding up as you're moving along. And the cracks aren't initiating all at one time, they're a random process also. So some cracks are initiating at 20 years in some realizations and some realizations they're not, they're happening at 10 years. And sometimes they're happening at 30 years.

And so whenever those cracks become leakers and the leaks are greater than one gpm they're removed from the analysis and considered completely mitigated.

CHAIR ARMIJO: So unless you do the mechanical stress improvement or weld overlay initiation is still going on even though you --

MR. RUDLAND: Anytime you change the stress it changes the initiation behavior in the future. So if you change the stresses then initiation behavior in the future has been changed.

CHAIR ARMIJO: Now in this model do you have anything equivalent to what the BWR people do with hydrogen, water chemistry, noble metal, chemical additions? Any of the chemistry effects?

MR. RUDLAND: Yes. In this --

CHAIR ARMIJO: I didn't see a model in there, in your various modules, that had a water chemistry.

MR. RUDLAND: It's built into the crack initiation and crack growth modules.

CHAIR ARMIJO: It's a subset?

MR. RUDLAND: So there are corrections and different models depending on the water chemistry in those particular modules. So for instance in the pilot study the only thing we had in there was the addition of hydrogen, which has been shown to slow down PWSCC crack growth.

CHAIR ARMIJO: Oh it has?

MR. RUDLAND: In the laboratories, yes. But the proposals for Version 2.0 are to add zinc to the water, which is supposed to effect the crack initiation parameters. I mean the technical basis haven't been investigated yet but that's some of the things that are being considered.

Again, this hydrogen addition are the things that we're wanting to put into Version 2.0. I think I detailed that a little bit more in the subcommittee meeting, but didn't have time to do that here at this meeting.

MR. CSONTOS: And zinc. The zinc addition too?

MR. RUDLAND: I said that. So at the end of the day the project team developed this code, ran the sensitivity studies and determined that it was feasible to develop this modular based code. We were able to do that. We actually had an independent contractor some in, develop a module and insert it into the code by himself. He was able to do that, so it was module enough that an independent contractor, not involved in this project, was able to do that.

And we did it within this cooperative

program with the people you saw working here. So I think we did a pretty good job of being able to do this within this cooperative environment between EPRI and the NRC.

However, we did have some problems. We did have some management problems. We originally had an overarching program integration board that looked over the entire project made up of, I think, 12 or 13 engineers. And getting 12 or 13 engineers in a room, at the same time, to make a decision was not an easy task. So we raised that a little bit and so we didn't have such a top-heavy management structure in the project.

There was also a bunch of technical issues that learned. In the end of the day we also chose this commercial software. Not because it was any better or worse than the open-source software developed, but I think it was thought that in terms of revisions and maintenance in the long term, it would be cheaper to do this through a commercial entity than tot allow a national lab or something like that to be able to control that portion of the software.

So in the fall we kicked off Version 2.0 and our objective for Version 2.0 is to basically grow from what we've learned in Version 1.0, but to focus on the systems, models, mechanisms that are applicable to the LBB lines.

MEMBER BROWN: Can you hold on a minute. Did you say commercial qualified or whatever --

MR. RUDLAND: Commercial software.

MEMBER BROWN: Commercial software and open-source. I would have imagined that the commercial software that you got had test developed by somebody.

MR. RUDLAND: Yes.

MEMBER BROWN: Therefore it's theirs?

MR. RUDLAND: It is theirs.

MEMBER BROWN: All the code is theirs. Which means you don't know all the details of all their codes?

MR. RUDLAND: That's true.

MEMBER BROWN: Whereas with the open-source software largely you have, yes you don't have somebody in charge of it, it's different maintenance characteristic. But at least you know all the little nuances that are involved. And I only bring that up because just based on some experience.

MEMBER POWERS: Well usually Sanjoy brings it up.

MEMBER BROWN: Well the problem with the

proprietary code is they can make changes in some areas which may have an impact on your results but which you are not aware of when you finally go to use an updated, upgraded version. As we frequently find out when we download upgrades to commercial code on our computers. And then all of sudden it just stops.

MR. RUDLAND: Yes. You know we considered all this and the issue about --

MEMBER BROWN: Well one other thing. It's not so much it stops, obviously you know you've got a problem. It's what you don't know what's going on in the background which may affect and maybe make your results not representative of reality, because you haven't done what I call test or experimental verification of every little thing as you go along. And say does it still meet those.

MEMBER BANERJEE: Well what is the commercial code?

MR. RUDLAND: It's called GoldSim. It was developed as part of the Yucca Mountain effort. DOE used that in their Yucca Mountain submission.

MEMBER BANERJEE: I mean where is your source code?

MR. RUDLAND: I do not have a source code for GoldSim, no.

MEMBER BANERJEE: But you could get it?

MR. RUDLAND: Well, let me talk about that for a second. So what we decided was that when it comes to the revision changes for GoldSim it still has to go through the same V&V effort through the program so when --

MEMBER BROWN: What program? Their program?

MR. RUDLAND: Their program and our program. So if a version changes we still have to run through our V&V effort again to make sure that whatever change they made to their code doesn't effect our V&V software packages. So that's one.

CHAIR ARMIJO: That's if you're informed, right?

MR. RUDLAND: Pardon me?

CHAIR ARMIJO: You would always be informed of changes that they made?

MR. RUDLAND: Of course, sorry. That's right. And they're very good. It's not a very big company to begin with, and they're very good because the NRC and DOE has been a very good customer for a very long time.

MEMBER SCHULTZ: Like large corporations called General Motors and Airlines and so forth. Is there something in place in case --

MR. RUDLAND: Yes. So what we're doing is we're working out an escrow program so if they go belly up, for instance, then we'll get the source code.

MEMBER SCHULTZ: Oh, okay.

MR. RUDLAND: And they're working it out into the escrow agreement with GoldSim.

MR. CSONTOS: As well as this software, commercial software, has been approved for use by NRC for NMSS and FSME for their efforts in some of the radioactive-waste arena. So for our side it's been already approved, it's been V&V'd for that specific version of it. And so any other version probably has to work with NMSS because we then get a V&V for the next version or whatever.

MR. RUDLAND: They're actually part of the project team because we're using them to make modifications to the code to suit our needs for this project. But they have other kinds, you know, they have a whiskey distillery client where they --

MEMBER SCHULTZ: Okay so they are --

MR. RUDLAND: It's not just us.

(Simultaneous speaking)

MEMBER REMPE: If a company, they're not

a participant, but they want to use it for an application, do they pay?

MR. RUDLAND: No. You have to pay a royalty if you're a developer. So to develop the code you have to pay a royalty, but then the code is compiled and rolled out in a player version, kind of like Adobe, so then you don't have to pay the license in order to run the code. You can change the input, you can change whatever the developer put on the GUI screens. You just can't change the code.

CHAIR ARMIJO: Okay, David, we've probably got to move along here.

MR. RUDLAND: Okay. So in Version 2.0 we want to focus on these LBB lines. And we have developed and expanded the team that we had and the QA program that we had in Version 1.0. And the groups are in the process, most of them are finished, developing work plans, manpower resources, to do this Version 2.0 code.

And one of the things that's very important that I've been touting around when I've been making presentations on xPLR is that we've been talking a lot about the models that go in xLPR and the models that will be in xLPR but it doesn't guarantee regulatory approval. So we're still developing the process of how best to handle the regulatory approval of things that xLPR are used for in the future.

We're going to be making some additions. I've labeled some things in red and some things in black. The things in reds are the ones that we are taking on first. So again, PWSCC was the only modeled in the pilot study. We need to include environmental fatigue.

We only looked at circumferential flaws, we need to look at axial flaws. And then depending on whether or not we have the manpower and the resources to do it in our time frame we're going to expand it to IGSCC and other mechanisms.

There's some issues, some mechanical and fracture mechanics issues, with transitions of surface cracks to through-wall cracks. As cracks begin to leak they form kind of unusual shapes which can effect the leak rate. We need to do some research and some development in that area also, which is new compared to what we did in Version 1.0.

CHAIR ARMIJO: David, you're not going to treat, you don't think you need to treat flow accelerated corrosion?

MR. RUDLAND: Not for the LBB lines. In the primary system it's really not an issue.

CHAIR ARMIJO: Yes, but for just basically rupture of any big carbon steel line or --

MR. RUDLAND: Yes, if we go beyond the primary system we need to do that.

CHAIR ARMIJO: But right now you're sticking with the primary?

MR. RUDLAND: We're not going to do that with the LBB lines, no.

CHAIR ARMIJO: Do we have a water line --

(Off microphone comments)

MR. RUDLAND: Apparently, yes.

MALE PARTICIPANT: On the secondary side.

(Off microphone comments)

MR. RUDLAND: Yes, there's issues in the ASME code with it all the time. You know, there's still code cases going on in developing and work being done on flow, it's a big deal in Japan also.

(Off microphone comments)

MR. RUDLAND: Oh, that's right. And then you update your model.

(Off microphone comments)

MR. RUDLAND: Interaction effects, yes.

MEMBER POWERS: It strikes me as a heck of a good idea, because I just don't want to be --

(Off microphone comments)

MR. RUDLAND: To the framework we're going to be doing some modifications from the Version 1 code. We're going to be focusing on uncertainty propagation and advanced methodologies, and maybe we'll be concentrating on getting rid of Latin hypercube, I don't know. But we're going to be looking at those kinds of things.

(Simultaneous speaking)

MEMBER POWERS: And like a dog with a bone you'll never give it up, but that's okay. Every time you show up here I'll ask you.

(Off microphone comments)

MR. RUDLAND: I won't mention it next time.

MEMBER POWERS: I'll remember.

MR. RUDLAND: So we're going to be looking at a bunch of advanced methodologies to help improve sampling, efficiency. We want the code to run relatively quickly, so we're going to be looking at many more different types of adaptive sampling methods and things like that, as well as ways to propagate these uncertainties without necessarily having to break things up into double-nested loops and that kind of thing. So we're looking at those options.

In terms of modeling, again we're going to be really focusing on PWSCC initiations since it really is the driver, one of the main drivers.

I mentioned earlier, I didn't show in this presentation, but I mentioned earlier about there's still a bunch of epistemic uncertainty that's causing this spread. Well, a lot of it comes from residual stress and so we're going to be updating our residual stress models. And we've been doing a lot more in conjunction with that so if you are to understand the uncertainties in residual stresses and how to --

MEMBER BANERJEE: To do that do you have measurements?

MR. RUDLAND: Yes. So we have a program within the Office of Research that's looking at making measurements and analyses, and we've had an international round robin to try to see how well we can predict the stresses and what the uncertainty is compared to the experiments on these types of welds.

MEMBER SHACK: You treat that as an epistemic uncertainty or a --

MR. RUDLAND: Yes, right now it's an epistemic uncertainty. But again, you know, like I said we need to revisit some of that because there's a big aleatory component to it.

MEMBER SHACK: Right.

MR. RUDLAND: Yes. There's no doubt.

MEMBER SHACK: I mean they're both there. I'm not sure which is dominant at the moment.

MR. RUDLAND: Well, yes. I have to say it's, the aleatory's pretty large, I think, but we just don't know.

(Off microphone comments)

MR. RUDLAND: And then we have to update our crack solutions also, our stress intensity solutions, to be consistent with the residual stress models that we develop.

CHAIR ARMIJO: Dave, just going back. Have you ever run this code Version 1 all the way until you actually calculate a rupture? Not a probability of rupture but an actual --

MR. RUDLAND: Every epistemic realization creates a rupture, yes, or it gets to the end of the time that they're allotted for the code and doesn't.

CHAIR ARMIJO: But, you know, as a crack is growing deeper the whole state of stress in the pipe changes, and your code does adjust for that? Because I think that's what favors leak as opposed to rupture unless you're in pure tensile loading.

Does your code treat those kinds of things?

(Simultaneous speaking)

MR. RUDLAND: Force equations and the influence functions that go into determining how the crack is driven takes that into account. They're developed through a series of finite element analyses that allows for the redistribution of stresses and things like that. So it's taken into account and then those models are incorporated here.

So the redistribution doesn't happen in this thing because it's not something that needs to be done here. It's done off line, the model's developed and then the model's put into here. But it's all accounted for.

CHAIR ARMIJO: There's a group of people, and I'm one of them, that believe that all this stress-corrosion cracking, the initiation is governed entirely by residual stresses. And that once these cracks nucleate and grow, if you haven't got enough residual stress at the surface, i.e., surface to nucleate at time after the first one, the likelihood of new nucleations drops a lot, and from then on it's all applied stress from plant operational stresses that drive it through.

And I don't know if your code people think that way or your modelers think that way but -

MR. RUDLAND: Yes, because you know --

CHAIR ARMIJO: -- that's the way some of us --

MR. RUDLAND: -- the zone that's relieved especially on the surface doesn't cover the circumference for a flaw that initiates. So there's probably a certain area that is relieved that's near the crack faces.

CHAIR ARMIJO: Or the residual stress isn't ever going to be high enough to nucleate a crack but for one location.

MR. RUDLAND: That's a good point, because a lot of times it's the really very localized repairs that are causing the issue, or a grinding effect or something like that.

(Simultaneous speaking)

MR. CSONTOS: There is fabrication that is 360 around with a back chip and reweld that I think if you look at some of the earlier photos. So there's substantial residual stresses in some plants, not all of them, but in some plants because of the way they fabricated the typical fabrication methodology.

So yes, in some cases you have surface grinding effects that produce a superficial surface layer of residual stresses. In some cases you have partial small zones of repair --

(Simultaneous speaking)

MR. CSONTOS: And then in some plants you have a whole 360, you know, a residual stress state, tensile residual stress state.

MR. RUDLAND: In some places you have repairs you don't even know you have.

CHAIR ARMIJO: Right.

(Simultaneous speaking)

MEMBER SHACK: But in your model, I mean if you just test surface stresses, presumably the crack basically arrests?

MR. RUDLAND: If you only have surface stresses.

(Simultaneous speaking)

MEMBER SHACK: You have a local stress that leaks and then you only get a rupture if somehow you keep missing the leak.

MR. RUDLAND: Well, it's not that is that if you end up with high stresses all the way around the circumference you can end up with this long, crazy flaw that as soon as it leaks it's not stable. So you don't get a chance for it to get caught by leak detection before the rupture occurs.

CHAIR ARMIJO: Well, I think Bill did that experiment, didn't you?

MEMBER SHACK: I've done that once, yes.

CHAIR ARMIJO: If you have a pipe in pure tension and you nucleate uniformly around the ID, there's nothing to drive one crack ahead of the other until it goes unstable and then you can have a pipe rupture.

MR. RUDLAND: That's the flaws that we're worried about, right. And that's the kind of thing that these 360 repairs that Al's talking about that's the kind of things that can drive those kind of cracks. And in these cases a lot of times when you look at these things, those are the flaws that are showing up here at the ten to the minus nine probability are those kind of weird cases.

Okay, so in the models like I talked about earlier, this MSIP was the only mitigation technique that we had in the Version 1 because it was just a feasibility study. We are going to be putting FSWOL, which is full structural weld overlay, optimized weld overlay, OWOL, inlays, which is a thin removal of material on the ID, replacing that with a less susceptible material, another mitigation technique that's considered.

Surface treatments like peening and those kinds of things are being considered as well as these chemistry things I talked about earlier, the zinc and the hydrogen additions. We're going to be updating inspection modules and looking at a more accurate crack, surface crack stability modules.

We have a little issue with our leak rate module. It's not very efficient. So we're looking at ways of rewriting or reorganizing our leak rate model so that we can make that run more efficiently.

Some of the bigger updates that we're doing now, which we have to include transient loads as part of the inputs because we're going to be considering fatigue. And so we're going out now and trying to determine and pull together what information do we need to pull from the plants in order to get all the transient information.

So for Version 2, we're going to be focusing on a reactor cooling loop from a Westinghouse plant and reactor cooling loop from a B&W plant. And if we have the resources and funds we're going to go do another or maybe even a BWR. We'll have to see as the project goes on if we have the funds to do that.

QA --

MEMBER SHACK: What would be the mechanisms in the B&W plant?

MR. RUDLAND: You mean fatigue mechanisms?

MEMBER SHACK: I mean the degradation mechanisms you're looking at, just fatigue?

MR. RUDLAND: Yes, just fatigue, though I think they have, I'm pretty sure there's a some dissimilar metal welds in the B&Ws also.

MEMBER SHACK: Is there?

MR. RUDLAND: Right, because the hot legs are carbon steel, but surge lines, I think, are stainless.

MEMBER SHACK: Surge lines are stainless.

MR. RUDLAND: So there are some dissimilar --

MEMBER SKILLMAN: All the B&W plants have dissimilar welds, all of them.

MR. RUDLAND: Yes, okay.

MEMBER SHACK: But high nickels.

MEMBER SKILLMAN: Not sure.

MEMBER SHACK: I mean there's certain stainless to ferritic, but the question of whether there's nickel --

MR. RUDLAND: I'm almost certain there's some in the surge.

MEMBER SHACK: Surge line, yes, there could be.

MR. RUDLAND: Version 1 was we had a comprehensive configuration management system that we put together for Version 1, but it wasn't coupled to a complete QA program. But in Version 2 we are going to have a complete QA program.

And at the beginning, last fall, we put together a QA workshop and brought in guys from the NRC, guys from the industries that were QA experts and asked them, for this type of project what type of QA structure do we need? What kind of structure is needed so that this code can be used in a regulatory framework? And from that we developed our QA program.

We've been kind of hinting about it as we've been talking today, but it includes, you know, it's structured off of a NQA-1, but it's project specific and it's a developmental QA program and not a release type of QA program. So it doesn't have the maintenance level defined yet. It has only the developmental stuff defined, but it includes configuration management and V&V and all of the essential parts of a quality assurance program.

And one of those important things again is the audits, and so we will have QA audits of the program that are going to be aligned with some of the key deliverables that will coming up in the next couple of years, and we'll have a third-party independent audit done for that.

MEMBER SIEBER: Is it possible for some to fail-break without leaking at all?

MR. RUDLAND: Yes. And that's kind of what we're --

MEMBER SIEBER: What's the chance of that?

MR. RUDLAND: Very low probability.

MEMBER SIEBER: Okay.

MR. RUDLAND: I mean again it's very unlikely to occur, but it's the one that you're worried about.

MEMBER SIEBER: Right.

MR. RUDLAND: And so in the case especially of PWSCC you can end up with a very long surface crack, and as soon as that thing leaks you don't have enough area left to the pipe.

MEMBER SIEBER: And you're only hope is ISI for that, right?

MR. RUDLAND: The only hope is ISI for that, that ISI can catch that.

MEMBER SIEBER: Okay.

MR. RUDLAND: But again, ISI has a probability of detection so there's always a chance that you can end up missing it.

MEMBER SIEBER: Yes, right.

MR. RUDLAND: Especially if it's a ten-year increment and the crack growth rate is so fast that it initiates and grows within that ten-year period.

MEMBER SIEBER: Okay.

MR. RUDLAND: This is just up here for graphics. You don't need to try to read this. But the point of it is that the models and inputs development will be complete in early 2013. So in about a year we'll be done with the models input development, and then the code release is scheduled for the end of 2013. So a good portion of 2013 will be focused on the V&V efforts.

Like I mentioned, really the purpose of xLPR Version 2 is to come up with a quantified solution to the LBB issue, and how we deal with that in terms of the GDC and the regulation is kind of unclear at this point. I think the thought is that we'll probably end up having a Reg Guide that will demonstrate and dictate how some of these analyses, an acceptable way of doing some of these analyses. We've talked about maybe doing an update to the SRP but again it hasn't quite been decided yet, but we're going to be doing something in that level in terms of the regulation.

Another aspect is that this development will hopefully be able to be used in a variety of different problems like we talked about. Some important ones are, a research tool for prioritization I think is important because it allows us to look at this problem and really pinpoint where we need to focus our research especially for the LBB issue. We can help reduce those uncertainties and understand the problem better by using a code like this.

It could also be used on some of the problems like the transition break size or risk-informed ISI or the GSI 191 problem especially since some of them are talking about going risk-informed with that. No guarantees. It's just an option that it might be able to be used.

MEMBER SCHULTZ: David, have you had, I know of at least one utility out there who was embarking on it.

MR. RUDLAND: Yes, I do too.

MEMBER SCHULTZ: Have you had discussions with them?

MR. RUDLAND: They've contacted me but I've been kind of, kept mum about anything right now especially during this part of the development. I'm not right now involved at all from the regulatory side with that.

MEMBER SCHULTZ: Okay.

MR. RUDLAND: I know there's some in the room that are.

MEMBER SCHULTZ: I was going to say, are other partners on your team more actively in --

(Simultaneous speaking)

MR. TREGONING: Rob Tregoning from research, I can address that. Their schedule is well accelerated compared to when it --

MEMBER SCHULTZ: That's why I was interested --

MR. TREGONING: -- would be ready, so they're looking at other methods for quantifying break frequencies in locations. And they're not doing a mechanistic approach. Their evaluation is complicated enough. So they're simplifying that aspect of their analysis.

MR. RUDLAND: But problems like that could be used. This program could be used for programs like that. It may not be applicable to that plant's problem but --

MEMBER SCHULTZ: But if they're ahead of your curve, have you had any input from them on things that they're doing that might --

MR. RUDLAND: I think the team members are gathering input, yes. They are gathering input because they're being directly involved with that kind of stuff.

MEMBER SCHULTZ: Good. Okay, good. Thanks.

MR. RUDLAND: And the hope is to go beyond piping also, is to take a look at maybe some CRDM ejections. Because there are problemistic models that the NRC's developed for upper head penetration ejection that could be included here. And the FAVOR code was developed by Oak Ridge for the PTS problem for RPV issues. And so it's possible that could be ported into this type of environment also. We've tried to keep this thing generic enough so that we can, with minimal effort, incorporate these other problems.

So where are we? Again, Version 2 is underway. Again we started, you know, in the fall with the QA development. We want to have these ongoing meetings. This meeting here, I want to make sure that we try to, at least yearly, talk to the subcommittee that we talked to in the fall. Possibly, and again not on the project as a whole but on individual topics like residual stress and crack initiation, maybe chemistry and things like that to get your ideas and input on those things.

We also have besides NRC and EPRI management, we're developing an external review team that's not associated with the project that's going to give this project a recurring review of our progress.

And we also have internal reviews, NRR technical staff that's not so involved in the development is also doing technical reviews on this ongoing basis. We're trying to keep this thing as vetted as possible through its development.

MEMBER BANERJEE: So you've presented this to our, what, fuels and materials subcommittee?

MR. RUDLAND: Yes.

MEMBER BANERJEE: And it involves all the gory details?

MR. RUDLAND: Four hours, four and a half hours.

(Simultaneous speaking)

MEMBER POWERS: Even more gory than this.

MR. RUDLAND: The problem is that there's a lot of gory details and I think that's why we need to --

(Simultaneous speaking)

MR. RUDLAND: Yes, we need to have topical meetings.

MEMBER SHACK: Initiation alone, four hours would be --

MR. RUDLAND: I'm going to be sick that day, I think.

MEMBER BANERJEE: And when you say external reviews, are these sort of like peer reviews?

MR. RUDLAND: Yes.

(Simultaneous speaking)

MEMBER SHACK: He's got everybody working on it.

MR. RUDLAND: Well, we're getting it into teams, we're gathering some professors and things like that from, some folks from other countries that are involved in problemistic fracture mechanics, and bringing them in to do an independent review, because they really haven't been involved in this development. And we're trying to find people that haven't been involved in the development.

MEMBER REMPE: That's a question I had, and I'm sure if you bring in the professors for peer review that the question will get asked. Is there any activities of this effort that could be done at a university, the V&V, or have you thought about that at all? Because I'm sure they'll be asking.

MR. RUDLAND: Yes, I'm sure there are. I mean we do have some, the industry side that there are some professors that are involved.

MEMBER REMPE: And so some of the work is actually being done by their students?

MR. RUDLAND: Yes, some of it's being done by students, and it's going to be more so in Version 2, yes. And, you know, the criteria for the external review, one of the criterias is just to not be involved in the project, and be, you know, find a technical area. So we're looking for, you know, fracture mechanics guys, we're looking for statistics guys. We're looking for plant guys. We're looking for people that are different categories so that we can put together a pretty comprehensive review team. And that's kind of why we like this review team, because it's very comprehensive in that aspect.

Version 2, then again like I said, is going to be released hopefully at the end of 2013. Right now we're on schedule but it's still quite a ways away. And then by 2015 we're going to have a technical basis and Regulatory Guide or some similar type of regulation effect in 2015.

MEMBER SHACK: Do you have some buy-in

from NRR that they'll accept the problemistic fracture analysis in lieu of?

MR. RUDLAND: Yes, I actually have a User Need Request from them to develop the software, so this is their idea and they want to go down this route. Tim can speak to that probably better than I.

MR. LUPOLD: Yes, but what we'll have to do is review the different modules that are developed as part of this code and decide how, the appropriate way to use them. We may decide that there's too much uncertainty in something like the initiation and then we might have to put some stipulation on how it's used. But that process has to come yet, we haven't worked out all the details.

MR. RUDLAND: One of the things that we're looking at because there's so much uncertainty in initiation, we're talking about putting in an option in the code to just run it conditional on initiation. Just take that out of the equation, you know, and we can deal with it afterwards, but at least then we can look at the effects of mitigation of things like that without having the uncertainty of initiation clouding of any answers.

CHAIR ARMIJO: Okay. Well, look, we're getting close to running out of time. I'd like to ask the members if there's any questions they'd like to raise or comments. Jack?

MEMBER SIEBER: None. I have none, thank you.

CHAIR ARMIJO: Sanjoy?

MEMBER BANERJEE: Interesting, but I'm not --

CHAIR ARMIJO: You've got to come to Subcommittee meeting.

MEMBER BANERJEE: Yes.

CHAIR ARMIJO: You're always welcome.

Steve?

MEMBER SCHULTZ: I'm good.

CHAIR ARMIJO: Dick?

MEMBER SKILLMAN: I'm good, thank you.

CHAIR ARMIJO: Dennis?

MEMBER BLEY: This will overwhelm --

(Off microphone comments)

CHAIR ARMIJO: Dr. Powers?

(Simultaneous speaking)

MR. RUDLAND: Well, we'll keep coming back and keeping you informed.

MEMBER POWERS: It's really interesting and it's really quite an expansive approach they're trying here. And I guess that all I can say is it's about time. It'll be very useful if you get it done.

You need to give careful thought about the use of the commercial software because that does have some downsides to it. It has some upsides too, and GoldSim is kind of almost commercial software. I mean it's one of the those things where if you wanted to look at the source code you probably can.

MEMBER BANERJEE: On that note though a lot of the CFD commercial software is being made available to NRC, the source code.

MR. RUDLAND: Source code.

MEMBER BANERJEE: Yes. And that we had a long fight over this before some years but now it's not --

MR. RUDLAND: And to think about this GoldSim Company, it's because they're small, they're focused and they realize the worth of their larger clients, and then maybe something like that could be worked out. I don't know, I mean we'll have to see where the evolution takes us.

CHAIR ARMIJO: Okay, Harold?

MEMBER RAY: No, sir.

CHAIR ARMIJO: Nothing? Nothing else?

MEMBER SHACK: I just can't figure out which topic we want to pick for that subcommittee meeting.

CHAIR ARMIJO: We'll talk about that.

Charlie?

MEMBER BROWN: I'm fine.

CHAIR ARMIJO: And Joy?

MEMBER REMPE: It sounds like a great effort. It's good to see industry and NRC cooperating on it.

CHAIR ARMIJO: Yes, I would like to add just a few things. I think it's a huge effort and you've made a lot of progress. But I think maybe you're putting too much on your plate on all the things you're attempting to cover, and focusing a little bit more wouldn't hurt.

I think there's something that you might want to think about in some confirmatory laboratory experiments that where you could make a prediction from your codes and actually try and confirm them by laboratory work. That seems to be missing from, you know, you're going from existing data and just using it.

(Simultaneous speaking)

CHAIR ARMIJO: Initiation thing is a real challenge, and I think I'd like to see, have a discussion on that somewhere in the future.

And you've always said you were going to incorporate the water chemistry variable, because in the BWR environment we could duplicate IGSCC and create it in our pipe test laboratory any time we wanted very predictably. We've tested pipes in tension, we've tested pipes in bending, and we're talking pretty big pipes.

And I don't know of any facilities similar to that exist anywhere anymore for PWSCC, but if there was some way that you could get some sort of confirmatory laboratory's PWSCC testing that you could set up to predict the behavior based on your code and have it happen in the lab that would be great, but I don't know if it's feasible. But I would like to --

MEMBER SHACK: Sounds like a national lab project.

MR. RUDLAND: Well, I think I mentioned this earlier that we are doing something some laboratory work on stability of these unusually shaped cracks, you know, this PWSCC gives us complex shaped cracks, and we're doing laboratory pipe experiments where we're taking dissimilar welds and putting cracks in them and then bending them to failure to try to understand that behavior. So some of it's happening --

MEMBER SHACK: Getting them to grow by PWSCC that's a different story.

(Simultaneous speaking)

MEMBER SHACK: And initiation's even worse because that's such a slow process.

(Simultaneous speaking)

MR. CSONTOS: -- certain materials with other types of stress-corrosion cracking versus nickle and PWSCC. Do you think there's an analogue --

(Simultaneous speaking)

CHAIR ARMIJO: If the model is good, yes. Is there good materials --

(Simultaneous speaking)

MR. CSONTOS: Like chloride stress-corrosion cracking --

(Simultaneous speaking)

MR. CSONTOS: Ammonia is another one. If you get something like that and if we put that into the code, you know, that may be a, it's just hard to get initiation of --

CHAIR ARMIJO: IGSCC, we know how to do that for BWR stuff.

MR. CSONTOS: That's what I'm asking. And so, you know, is that something that --

CHAIR ARMIJO: Well, you know, anything

that gives you some sort of, experimentally I've got, you know, I've predicted something and it actually happens.

MR. RUDLAND: They're good on a small scale, but there's no length to --

CHAIR ARMIJO: I'm not saying you have to test pipes, which we did, but just think about that.

MEMBER POWERS: Especially if you're going to go to any kind of surrogate testing and things like that then I think you definitely need to look at the CSAU methodology, and how you take that kind of an experimental result and get something that you can legitimately use in the validation and verification in the code.

MR. RUDLAND: Okay.

MEMBER POWERS: I don't know that you can take the CSAU methodology whole and just import it because it was developed for something else, but it would give you what, the thermohydraulic community has thought most deeply about this and they've developed something that gives you some real good ideas on what kinds of things you have to think about.

CHAIR ARMIJO: Rob?

MR. TREGONING: Just to clarify. You're probably aware of it but not everyone is. All the individual modules are verified quite extensively experimentally. Now it's true that we don't have one test to model the entire process from beginning to end nor do I think that type of test is necessary.

I think what we may find is once we learn enough about, and the beauty of the modular code is when you learn your risk and you learn what's driving your risk the first question you have is, well, is it really physically driving the risk or is it driving the risk just because I characterized that phenomenon so poorly and I had so much uncertainty?

Well, that allows us to go back and investigate that to see what's the case. There may come a point where it may make sense to do a couple of experimental tests, but planning and executing that in a way that we can convince ourselves that it's even representative of an actual situation is by no means trivial.

So while that's a possibility, I think we would only embark on that after very careful consideration and deliberation and planning. So that would not be something, at this stage I don't think we're ready to even envision what that test may look like. But down the road something like that may become not only, you know, feasible, but may be necessary to fully demonstrate conceptually. But all the pieces are definitely validated through 30 to 40 years of experimental and analytical work.

CHAIR ARMIJO: Okay, I hear you. But, you know, laboratory testing, you can control variables. And a lot of field stuff you don't know what those variables are.

MEMBER SHACK: Just think, Sam, try to do one crack growth test, which is an easy test, and think how difficult that is to get representative results out of one test. And then you go to a test like this where there's stresses, initiation, growth.

CHAIR ARMIJO: You know, maybe breaking it up into a test of a specific part of a model in the laboratory --

(Simultaneous speaking)

CHAIR ARMIJO: -- and maybe that's what Rob says already exists, but we'd like to look into that and see --

MR. RUDLAND: Yes, we could talk about that. We've talked about the validation of the models as one of the break-out talks.

CHAIR ARMIJO: Okay. Well, look, first I'd like to thank the staff for a very good presentation. And we ran a little bit longer, so I think we've got Turkey Point.

And I think we're going to try, do you want to start it at 3:00?

MEMBER SHACK: Let's take a 15-minute break.

CHAIR ARMIJO: Fifteen-minute, okay.

(Whereupon, the above-entitled matter went off the record at 2:52 p.m.)

CHAIR ARMIJO: Okay, we would like to reconvene and Dr. Shack will lead us through the Turkey Point EPU discussion.

MEMBER SHACK: Okay, we have had two subcommittee meetings now on Turkey Point EPU and a meeting of our full committee in January where we covered most of the issues associated with the EPU.

The big remaining issue of course if the handling of the thermal conductivity degradation and the we have heard from the licensee on that but what we are going to go through today again is a presentation from the staff on their evaluation, their confirmatory calculations and their conclusions.

From the licensee, we are going to have an overview of some topics that did come up in the January meeting; a loss of off-site power overview and their shared systems overview. Again, some of the features that are rather unusual about Turkey Point compared to other plants.

So I will start off with Allen Howe for the NRC.

MR. HOWE: All right, thank you. I am Allen Howe, Deputy Director, Division of Operating Reactor Licensing in the Office of Nuclear Reactor Regulations. We appreciate the opportunity to brief the ACRS this afternoon on the Turkey Point Extended Power Uprate Application.

As we mentioned, we did brief the ACRS Subcommittee back in December, the full committee in January and went back to the Subcommittee in February.

At our additional briefing of the Subcommittee, we addressed the resolution of several open items from the full committee meeting in January and our specific focus was on the thermal conductivity degradation issue.

In today's meeting the staff will present an overview of the result of our safety and technical review of the licensee's application regarding thermal conductivity degradation. There were no open items associated with the EPU review but we had a couple of clarifying questions from the February 12th or February 2012 Subcommittee meeting. The staff has provided responses to the ACRS staff and we are prepared to provide any clarification during today's meeting if needed.

Jason Paige will provide more details on the discussion topics for today's meeting. But before I turn it over to Jason, I just want to say that I am pleased with the depth and the breadth of the staff's review. In evaluating the Extended Power Uprate Application, the staff addressed a diverse set of technical issues, including the thermal conductivity degradation issue, which required extensive interaction with the licensee.

And at this point, I would like to turn over the presentation to Jason.

MR. PAIGE: Thank you, Allen.

Good afternoon. My name is Jason Paige. I am the Project Manager in the Office of NRR assigned to Turkey Point.

First, I would like to take this opportunity to thank the ACRS members for your effort in reviewing the proposed EPU application and revised safety evaluation.

I also want to express my thanks to the NRC staff for conducting a thorough review of a very complex application and also for providing support to these meetings.

During today's full committee meeting, you will hear from both the licensee and the NRC staff in providing you with specific details of the EPA application. The objective is to summarize the TCD issue and present the status evaluation supporting our reasonable assurance determination that the proposed EPU will not endanger public health and safety.

Before I cover the items that were discussed during the February 24th ACRS subcommittee meeting and agenda for today's meeting, I would like to provide some background information related to the proposed EPU.

On October 21, 2010, the licensee submitted its License Amendment Request for Turkey Point Units 3 and 4 EPU. The proposed amendment will increase each unit's license core power level from 2300 megawatt-thermal to 2644 megawatt-thermal. This represents a net increase in license core power of 15 percent, including a 13 percent power uprate and a 1.7 percent measurement uncertainty recapture. This is a 20 percent increase from the original license thermal power.

The staff's method of review was based on Review Standard 001, which is NRC's review plan for

EPUs. As you know, it provides a safety evaluation template, as well as matrices that cover the multiple technical areas that the staff is to review.

There are no associated or linked licensing actions associated with the EPU application. FPL previously submitted two license amendments for Turkey Point, the AST amendment in 2009 and the Spent Fuel Pool Criticality amendment in 2010.

The NRC staff approved the AST and Spent Fuel Pool Criticality Analysis amendments on June 23 and October 31st of 2011 respectively.

Finally, there were numerous supplements to the application, responding to multiple staff RAIs. Overall there were approximately 50 supplemental responses, which supported our draft safety evaluation. Also the staff completed several audits to complete its review and resolve open items.

As Allen mentioned, we briefed the ACRS Subcommittee and full committees on December 14, 2011 and January 19, 2012, respectively and followed up with additional briefings to the Subcommittee on February 24, 2012.

The follow-up Subcommittee briefing focused on the thermal conductivity degradation issue. This slide shows all of the items that were resolved and discussed during the ACRS Subcommittee meeting on February 24th. All open items have resolved by the staff and, as requested, the staff provided all licensee supplements to address the open items to the ACRS staff.

Also the staff provided a revised safety evaluation to ACRS staff on Tuesday February 21, 2012 with a list of sections that were revised to close out the open items.

At the conclusion of the February 24th Subcommittee meeting, a couple of clarifying questions were generated which were addressed by the staff and provided to the ACRS members and staff via email. The staff's presentation does not focus on these clarifying questions but, as Allen stated, the staff is prepared to provide clarification if requested.

Before I move on, as one point of clarification, Dr. Graham Wallace asked a question during the February 24th Subcommittee meeting regarding PAD4TCD license condition that was believed to be issued back in June 2002. I personally apologize for the confusion. We couldn't clarify it during the Subcommittee meeting but hopefully we will be able to do that right now.

If you look at the licensee supplement, I provided a handout to you, to everyone. It is dated January 245, 2012. The licensee in this supplement provided a marked-up page of the license, which included the license condition that Dr. Graham Wallace was referring to. If you look at the bottom of this supplement, you can see the date of June 6, 2002 and that is where the confusion came from. This date represents the last time that this page was revised with the issuance of a previous License Amendment Request. And I believe this date reflects the approval of license renewal for Turkey Point.

So once the EPU is issued with the PAD4TCD license condition, this date will change on this page to reflect the EPU issuance.

This slide is the topics for today's discussion. The topics were determined from the discussion topics from the February 24th Subcommittee meeting.

First, the licensee will provide an introduction, then FPL will provide additional details and clarifying information on discussion topics from the ACRS Subcommittee meeting on February 24th, which are loss of offsite powers and shared systems.

The NRC staff will then summarize the thermal conductivity degradation issue and present why the licensee's proposed approach is acceptable.

Even though the focus of today's meeting is on TCD, the staff is available to address any questions you might have on any other technical review area.

Lastly, a portion of the staff's TCD presentation will be a closed session, due to there being proprietary information on the slides.

Before I turn it over to FPL, I would like to also mention on behalf of the NRR staff members involved in this review, we would like to say a special thanks to the ACRS staff that helped in the preparation for the Sub- and full Committee meetings, specifically Weidong Wang.

Are there any questions at this point? Okay, I will turn it over to Mr. Mike Kiley. He is the site VP at Turkey Point.

MR. KILEY: All right, good afternoon. As Jason said, my name is Mike Kiley. I am the Site Vice President of Turkey Point. With me today is Steve Hale to my immediate left. Steve is the Director of EPU Licensing. To his left is Sam Shafer. He is the Assistant Operations Manager at the current SRO licensed facility with over 25 of experience at Turkey Point.

First of all, I would like to thank the ACRS for the opportunity to discuss the Turkey Point EPU License Amendment Request today. Since the last ACRS Subcommittee or I should say the last two ACRS Subcommittees and the last full Committee meeting, we have been working very, very closely with the NRC staff to close remaining open items.

Now we recognize and appreciate the importance of the staff's questions, particularly those centered on thermal conductivity degradation. The NRC staff performed a detailed review of our analysis, including a multi-day audit at the site, in which we presented our analysis to the staff. This audit and others performed by the staff have been invaluable and have led to a very thorough and comprehensive review of the proposed EPU.

Our culture at FPL and Turkey Point is grounded by a strong commitment to nuclear safety. It is uncompromising. This is our top priority and we do not take that responsibility lightly. So at this time, I would like to turn this over to Steve Hale and we look forward to being able to answer any remaining questions.

MR. HALE: Okay, as Jason mentioned, the focus of our presentation really is to hopefully clarify some discussion we had at full Committee in January. I wanted to talk a little bit about loss of offsite power. There were a number of questions. We walked through this with Subcommittee and it was thought this would be a good summary to go over with the full Committee.

With regards to dual-unit loss of offsite power, we have one event where both units lost offsite power. This occurred in August of 1992 with Hurricane Andrew. Of course, in advance of hurricane force winds, we had put both units in a safe shutdown condition. So from that standpoint, we have never had a case where we have had a dual-unit loss of offsite power with both units at 100 percent power.

There have four single-unit loss of offsite power events; two on each unit. These were internal or associated with the switchyard not specifically related to the grid.

There was one event that happened in 2008, February of 2008, which was a partial blackout in the State of Florida. This event did cause a trip of both units but neither unit lost offsite power. We were able to do the fast bus transfer to the startup transformers and we maintained offsite power.

When you look at it strictly from a risk standpoint, the risk of a or the probability of a dual-unit LOOP is about 0.02 per year. When you look at it in terms of Core Damage Frequency, we are well down into the low probability eight times ten to the minus eight per year.

The transient analysis that we did perform did demonstrate the specific criteria for Cat 2 events as we not fill the pressurizer and we demonstrated for a regular loss of offsite power that that does not occur.

And another time that we wanted to clarify, I think there was a question. We are not adding any new operator actions as a result of loss of offsite power with the EPU. The results are very similar. It is a fairly quick transient and we are not adding any new additional operator actions as a result of the EPU.

MEMBER SCHULTZ: Steve?

MR. HALE: Yes.

MEMBER SCHULTZ: Don't flip so quickly there. I guess I am, for a plant design like Turkey Point, I am a bit surprised by a number that is as small as eight times ten to the minus eight per year for Core Damage Frequency for a loss of offsite power.

Do you have any information on that?

MR. HALE: Yes, we had -- and it really kind of leads into the next topic talking about shared systems.

MEMBER SCHULTZ: Okay.

MR. HALE: I wanted to walk through that.

MEMBER SCHULTZ: Okay, thanks.

MR. HALE: We did get in -- okay.

MEMBER SCHULTZ: Thanks, I'll let you do that.

MEMBER BROWN: Before you go on, can you enlighten me, because I am not enlightened right now, on how you come up with a 0.02 per year dual LOOP when you have had one in 20 years?

MR. AVERETT: I can talk to that.

MEMBER SCHULTZ: Thank you.

MR. AVERETT: My name is Mark Averett. I am the lead PRA analyst for Turkey Point.

And 0.02 per year -- you are referring to the Hurricane Andrew event?

MEMBER BROWN: Well I just looked at 20 years and one drop out and I divide one by 20. I'm not an analyst and I am not a PRA guy. That just worked out to be five percent to me.

MR. AVERETT: Well, I understand. Well like a lot of things in PRA, we not only use plant-specific data, we also use industry data as well. It is just one event isn't really that statistically significant. So we used Bayesian-update the plant-specific data along with industry data to come up with that number.

MEMBER SCHULTZ: The plant-specific data with what industry data?

MR. AVERETT: I beg your pardon?

MEMBER SCHULTZ: Plant-specific data with what particular industry data?

MR. AVERETT: U.S. nuclear industry data.

MEMBER SCHULTZ: If you could give me a reference, it would be helpful in terms of knowing what that means. I am looking at a tabulation of data that has been specialized from U.S. nuclear industry data to the Turkey Point site that has generic mean frequency of 0.047 per year. And if I update that with one event in 20 years, I don't think I get 0.02. I think I get a much higher frequency. So I am curious what data you use.

If I use data for Southern Florida, for example, it probably doesn't look like other sites. So if you did a Bayesian-update with using as a prior data from a lot more beneficial sites, you are optimistic.

MR. AVERETT: Well what we did is we typically take like the last ten years of data.

MEMBER SCHULTZ: If you took last year's worth of data, you might get like zero. This looks at data for like 40 years.

MR. AVERETT: Well what we do is we take the last ten years' worth of data. EPRI publishes, every year they publish a compendium of loss of offsite power events throughout the industry. We look at those industry events. We examine them to see if they are applicable to Turkey Point, if it could happen at Turkey Point. Things like snowstorm events we don't count. Things like hurricane events we weight more heavily. And we -- I can't compare it to what you are looking at without having both sets of data.

MEMBER SCHULTZ: This is NUREG/CR-6890 from Table B-2.

MR. AVERETTE: So we used the EPRI data.

MEMBER BLEY: Did you use the national average? Is that what you started with?

MR. AVERETTE: We used national data, yes.

MEMBER BLEY: Okay, so you mixed in data from clients with much more favorable offsite power configurations than you have?

MR. AVERETTE: Yes, but the Bayesian process should take care of that.

MEMBER BLEY: And you did it for the last ten years, not --

MR. AVERETTE: I believe it is about the last ten years.

MEMBER BLEY: I thought you were going to tell us that you weren't going to count this hurricane-related one because you shut down. And I could almost see that story. But this one is a little harder to buy into.

MR. AVERETTE: Well there is that argument, too, because it had been shut down for some hours before the loss of offsite power occurred.

MR. HALE: Well there is one area also I would like to mention is the transmission system is not static. As part of our discussions in getting ready for this meeting, we spent quite a bit of time with our transmission distribution department. They have done quite a bit of upgrades since the last 20 years as well. Hardening, one of the things certainly is better performance during hurricanes, things of this sort. So you know, from that perspective, the transmission system has been improved significantly over the last 20 years as well.

But I think the intent here is to show that we have never experienced a dual-unit loss of offsite power with both units at 100 percent power. Okay? The one case we did have dual-unit LOOP was a hurricane where we are procedurally required to reduce power and then go into a safe shutdown condition before the onset of hurricane force winds. And the results of our analysis indicate that we are very similar to where we were before the EPU and we are not adding any additional operator actions as a result of that.

MEMBER SCHULTZ: Yes, you are -- Okay.

MEMBER BANERJEE: But you have a higher heat load to get rid of.

MR. HALE: Sorry?

MEMBER BANERJEE: But you have a higher heat load to get rid of.

MR. HALE: That is true but the response is pretty quick to loss of offsite power event. The heat up event happens very quickly. And then it stabilizes fairly quickly.

MEMBER BANERJEE: Well what happens if you use 0.1 there instead of 0.02?

MR. HALE: You are still relatively low.

MEMBER BANERJEE: Okay, so why are you?

If you use 0.1, there is no effect. What would happen if you used 0.1 instead of 0.02?

MR. HALE: It would be affected by --

MEMBER BANERJEE: I mean it is not a big issue.

MR. HALE: Yes, we were strictly communicating what is used in our PRA analysis for this and we thought it would be worthwhile to describe what our history has been over the last 20 years.

MEMBER BLEY: But in a few slides you are going to tell us what takes you from 0.02 down to ten to the minus seven.

MR. HALE: Yes, that's true.

MEMBER BLEY: Okay, that is what I want to hear.

MR. HALE: All right. One of the topics we had for the Subcommittee, we did run through our PRA and discuss that. And of course one of the questions that came up was why is our probability of CDF so low. One of the main reasons is because of some of the design features at Turkey point that facilitate or help us in those risk scenarios.

Our Emergency Core Cooling System, we have a shared Emergency Core Cooling System. This consists of two high-pressure pumps per unit, two RWSTs per unit. If we get one SI signal, it initiates SI on both units. In other words, you get all four pumps starting and supplying flow.

We have four emergency diesel generators on-site and each pump is powered by a separate emergency diesel generator.

The Aux Feedwater System is fairly unique in that we have three turbine-driven pumps, any one of which can supply the auxiliary feedwater needs for both units. We are able to accommodate and address AFW needs without any AC power.

In addition to the turbine-driven pumps, they also have a diesel-driven standby feedwater pump which has its own diesel and it drives just that pump. It is not an electrical diesel. It is strictly for driving the standby feedwater pump.

As mentioned, we have four EDGs at the site and an SI on either unit will initiate the emergency diesel generators.

Also, as we mentioned at the Subcommittee as part of our license renewal effort at Turkey Point, the ACRS Subcommittee actually came to the Turkey Point site and we demonstrated the ability to cross tie the emergency diesel generators for station blackout. In other words, we can actually cross tie emergency diesel generator from the control room and one diesel can satisfy the station blackout needs of both units.

MEMBER BROWN: Simultaneously?

MR. HALE: Simultaneously, yes.

So the combination of these things all contribute significantly to our relatively low Core Damage Frequency at Turkey Point.

Any other questions?

MEMBER SCHULTZ: Core Damage Frequency then would apply to a dual-unit core damage event.

MR. HALE: Yes. Mark, if you could --

MR. AVERETT: What is the question?

MEMBER SCHULTZ: That number that you -- Because of the extent of cross ties, you pretty much have to lose everything to lose either unit. Which means, you have probably lost both of them. Is that right?

MR. AVERETT: Well, the Core Damage Frequency you see up there is typical of a unit, per unit Core Damage Frequency. We take into account dual-unit --

MEMBER BLEY: I understand. But if you had a dual-unit -- If you have loss of offsite power, that probably -- because of the way you are crosswired here, if you lose one, you probably lost them both is what John said. So that is probably your dual-unit Core Damage Frequency under --

MEMBER SCHULTZ: Given the fact that your shared systems can support both units, I am assuming you are taking credit for anything you can share in protecting the unit that you focused on. Meaning, if it is gone, the other one probably doesn't have a lot left.

MR. AVERETT: That's true. If you have a dual-unit loss of offsite power and you have a station blackout core damage sequence, then you have probably lost both units.

MEMBER SCHULTZ: Okay. And you say your turbine-driven aux feedwater pumps are capable without AC power. Are they capable without DC power?

MR. AVERETT: No.

MEMBER SCHULTZ: Okay.

MR. AVERETT: But it is a full power train. And then in addition to that, we have the backup of the diesel driven auxiliary.

MEMBER BLEY: Does that require any power?

MR. AVERETT: What's that?

MEMBER BLEY: Does that require DC power as well?

MR. AVERETT: No, not the standby pump.

MEMBER BLEY: The diesel pump just by itself.

MR. AVERETT: Stand-alone diesel. Right.

MEMBER RAY: The cross-tie of the Emergency Diesel Generators, is that part of the original design, do you know?

MR. AVERETT: No, it wasn't. The original Turkey Point design was two diesels for two units.

MEMBER RAY: Yes, I tried mightily in another life to achieve that and I never could get the NRC to approve it. So I wound up getting permission to manually affect the cross-tie. Because of the benefit it has to the PRA. I mean, it is obvious. But the potential, at least as was explained to me at the time for a single failure to take out the diesel from both sides was the reason I couldn't never get it approved.

MR. AVERETT: It is key-locked in the control room. Sam can walk through the process.

MEMBER RAY: You don't need to do that. I just wondered.

MR. AVERETT: But no, you are right. It wasn't part of the original plant design we installed. In fact, I was the engineering director at the site.

When we did this, we installed two new diesels at Turkey Point in the late '80s and early '90s. And as part of that design, we were also addressing station blackout and we incorporated the cross-tie, swing the kV bus into the design.

MEMBER RAY: Well, like I say, that is the right thing to do in my judgment but I never could get them to let me do it that way. I had to have cables that I want to plug in to the switch gear.

MR. AVERETT: Understood. So anyway, that was what we wanted to cover and hopefully we clarified some of the questions that came up.

MEMBER SCHULTZ: I can see where the cross-tie capabilities just --

MR. AVERETT: I would like to revisit one thing on the gentleman's question about whether if you have a station blackout core damage sequence if indeed you take out both units. In other words, a core damage event for both Unit 3 and Unit 4. Sometimes that is true, sometimes it is not. You could actually have a situation where you could have a core damage and a station blackout on say Unit 3 because you forgot, the operator failed to implement station blackout cross-tie.

MEMBER SCHULTZ: Yes, so there are some --

MR. AVERETT: Right, but if you have the hardware there, all the diesel --

MEMBER SCHULTZ: Hardware -- then you are out of luck.

MR. AVERETT: Yes.

MEMBER SCHULTZ: Okay, thanks, Mark.

MR. AVERETT: Okay, that really completed ours.

Similar to what we did, I just did want to mention that at Point Beach we did keep a small group of people here to facilitate any renews you need of the letter that ACRS was going to produce. I know we provide a comments and review on that. So we will maintain a small group here after.

MEMBER SHACK: We do plan to write a letter.

MR. AVERETT: Okay and we will provide any support we can in that regard. Okay? All right, thanks. We appreciate it.

MR. ULSES: Actually if I may, Mr. Chairman, I would like to just kind of give my -- Well, if we wait for Paul to settle here, just share a few thoughts with you on TCD. And for the record, my name is Tony Ulses. I am the Branch Chief for the Reactor Systems branch.

I'm going to start up talking about TCD today. This has been a challenging issue for the staff. And you know, we are waiting for the responses to the 50.54(f) letters. Those are not due until 19 March. And so quite literally, there are some generic aspects of this issue that we just do not have the information all in front of us yet. So at this point, we are not prepared to discuss all these issues.

But as you recall, when we were talking to you at the Subcommittee meeting, we do make a commitment to come and talk to you after we have had a chance to digest all the information and draw all our conclusions after we review the information. And we will do that after we have a chance to complete the review. As soon as we possibly can we will talk to Weidong and we will set this up.

But in the context of looking specifically at the work that FP&L did to address the issue for the Turkey Point station, I think what you are getting here today from the staff is that the licensee very aggressively approached this issue. They went back and they essentially re-did their accident analysis where it was necessary in order to complete this action.

So we have drawn the conclusion of

reasonable assurance of safe operation. That is what we are going to leave you with here today. I just kind of wanted to frame the issue before we get started. So thank you very much.

MR. CLIFFORD: Thank you, Tony.

Hello, my name is Paul Clifford, NRR Division of Safety. And you will have to use your imagination to see Ben here. He will be joining us on the phone when it is his turn to discuss LOCA.

MEMBER BANERJEE: Is he lying on the beach somewhere?

MR. CLIFFORD: Let's imagine he is.

MEMBER BANERJEE: At Point B.

(Laughter.)

MR. CLIFFORD: Okay, I will be describing the interim solution to the generic TCD issue that FP&L has put forth to support the EPU for Turkey Points 3 and 4.

We will be discussing the staff's review of the PAD4TCD Thermal Conductivity Model. And finally discussing some independent calculations which were performed with the audit tool FRAPCON-3.4.

The original Turkey Point License Amendment Request that came in in 2010 was based upon a currently approved PAD 4.0 model. This is the approved Westinghouse fuel rod thermal mechanical code that has been in use, I believe, since 1999.

It wasn't until relatively late in the staff's review that it became evident that we needed to address the thermal conductivity degradation issue. In response to RAIs which were issued by the staff, FP&L developed an interim solution to address this generic issue. It involved revising the fuel thermal conductivity model within PAD and using this revised model in all the downstream calculations, including the fuel mechanical design.

MEMBER BANERJEE: Paul, what was not completely clear to me is that interface between PAD and say your analysis using COBRA/TRAC or whatever that is used in your ASTRUM analysis. Is it just the stored energy or how does it sort of filter through? When I asked that question at the last Subcommittee meeting, it wasn't 100 percent clear to me what all those interfaces and interactions were.

MR. CLIFFORD: My understanding is, and I'm not an ASTRUM guy, is that they use the fuel performance code and they run various power histories and various burnups and they come up with table sets of stored energy, which would be average fuel temperature as a function of burnup and as a function of power level. So they have these tables of stored energy for different rods.

MEMBER BANERJEE: So there is a table look-up procedure?

MR. CLIFFORD: I believe so. And then when they do the ASTRUM analysis and they pick a rod out of the 124 cases to run, they would then, based upon whoever picks their rod, it would then go to the stored energy for that particular characteristics of burnup and power level.

MEMBER BANERJEE: And how does the fission gas part of it enter? Does it enter through something like the --

MR. PAIGE: Excuse me, Paul. If you are going to answer that question, you have got to make sure it is not proprietary information. This is still open right now. So if we need to close it out.

MR. CLIFFORD: Yes, I would suggest that we close the meeting.

MEMBER SHACK: Well, let's just go into closed session now because the next slide starts the closed session and we can do it right now.

MEMBER BANERJEE: I can hold the question until the next slide.

MR. CLIFFORD: Well, we're already here.

MEMBER SHACK: We are there. So we are in closed session now.

(Whereupon, at 3:40 p.m., the foregoing proceeding went off the record for a closed session and went back on the record at 4:49 p.m., continuing the open session.)

MR. CLIFFORD: Okay, Ben, I think you are all set to go. I'm on slide 19.

MR. PARKS: Okay, thank you, Paul. I am ready to begin my presentation. Is the room all set?

MR. CLIFFORD: I believe so.

MR. PARKS: Okay, thanks. I'm Ben Parks with the Reactor Systems Branch. I am going to talk about the thermal conductivity degradation issue and the licensee's assessment on the downstream safety analyses.

On slide 19 here we have the licensee provided a disposition for most events and did some explicit analysis to account for the effects in the TCD-sensitive events. I have listed those events here. And for the next two slides, I will explain what I mean by disposition and why we ultimately accepted that.

So for these five events, the licensee did an explicit analysis and they concluded that those events would continue to meet their acceptance criteria.

So their disposition I have details on the next slide. So if we could go to slide 20 which is disposition.

In the RETRAN model, there are some conservative features in RETRAN to offset the effects of thermal conductivity degradation. The assumption of an elevated average fuel pellet temperature that is greater than what would be calculated by PAD 4.0 and then on top of that, the entire core is initialized at a peak fuel temperature. So even the fresh high-powered fuel is initialized at this elevated fuel temperature. And that is expected to account for the TCD effects and the transient analyses.

So the more detailed VIPRE models that the licensee uses to calculate DNBR, the steady-state rods that they calculate in VIPRE, they actually model the fuel to coolant heat flux and not necessarily the explicit fuel rod internal conditions. And in those cases, those models are insensitive to TCD.

The transient VIPRE models use a maximum fuel temperature input. The licensee performed assessments to determine whether that would impact the calculated departure from nuclear boiling ratio and the licensee determined that the DNBR would not be impacted. I believe that was for two reasons. When the NRC stopped review of those, we realized that the transient analyses relied on conservative assumptions. For those events the licensee determined would be affected by a TCD, they performed explicit analysis to confirm that there was margin relative to the applicable acceptance criteria.

And basically because the transient analysis relied on conservative and bounding assumptions, what they have shown even as corroborated by their explicit analysis that the events are less sensitive to the effects of thermal conductivity degradation than the realistic ECCS evaluation.

And so for us, what that means is the licensee's disposition is acceptable while we wait for the generic resolution to the TCD effects because they did the explicit analysis where the analysis was needed.

On side 22, I will get into the realistic Emergency Core Cooling System evaluation. The licensee addressed TCD and introduced additional operating restrictions to offset its effects in the LOCA analysis. The offsetting operating restrictions included a reduction in the power peaking, an increase in the reactant coolant system average temperatures, and additional investigation of the accumulator volume uncertainty. Basically what I mean by that is the licensee looked into the way that the accumulator volume uncertainty was being treated, both in its tech specs and the surveillance requirements, and in the ASTRUM analysis. What the licensee determined was that they were double counting their accumulator volume uncertainty because the tech specs already included an allowance for that uncertainty. So they tightened up that number to be more reflective of the way that the plant is required to operate by tech specs.

MEMBER BANERJEE: Ben, how sensitive are the results to this accumulator volume uncertainty, as well as there was one more, which was the number of steam generator tubes that could be plugged. Right? I'm just wondering --

MR. PARKS: I've got a couple plants in my head. I believe Turkey Point's solution was to reduce steam generator tube plugging from ten percent to five percent. They do not explicitly quantify the effect of each of these parameters and basically by I will say convolving, by convoluting these together, the previous sector re-scattered all the results. So it is hard to say explicitly how sensitive the overall results are to one of these parametric changes. They didn't do that sensitivity for me.

MEMBER ABDEL-KHALIK: How much was the power peaking reduced?

MR. PARKS: There was a change in the SQ. I think it came from 2.5 to about a 0.2 reduction, if I recall. If you need the explicit number, I will have to look it up. I can't remember.

MEMBER ABDEL-KHALIK: Can the licensee provide that information, please?

MR. O'FARRILL: Yes, I can. This is Carl O'Farrill with Florida Power and Light. We changed the transient FQ from 2.4 is what we had in the original submitted to 2.3. We also changed the F delta-h value from 1.65 to 1.60. The study FQ was changed from 2.0 to 1.9

MEMBER ABDEL-KHALIK: Thank you.

MEMBER BANERJEE: How much in the way of tube clogging do you have at the moment? Is it very low?

MR. O'FARRILL: It is well below that five percent value. I don't have those numbers right in front of me. It is on the order of a percent or two.

Maybe Rudy does.

MR. GIL: Yes, Rudy Gil, FPL. One of the generators is just at two percent and the majority of the generators are well below that. It is not a -- that one is not a mechanism that we would expect would grow much more.

MEMBER BANERJEE: Can you also just recount the accumulator volume uncertainty treatment? It was done at the Subcommittee meeting but it wouldn't harm to get clarification on that once more. I think it was Steve Hale who told us about it.

MR. HALE: Yes, Steve Hale, Florida Power and Light.

What we found is the accumulator volumes that were addressed in the ASTRUM analysis were based on a taking the tech spec values and then applying uncertainty to the tech spec values. What we found is that the tech spec values already accommodated the uncertainty. So essentially we were subtracting the uncertainty twice as a result of that. So we just corrected that one aspect of it. But we basically used the tech spec values which already incorporated uncertainty, rather than counting it twice.

MEMBER BANERJEE: Thanks.

MEMBER SCHULTZ: Steve, for completeness, can you provide the RCS T-ave increase?

MR. HALE: Yes. If you will recall, when Westinghouse performs these safety analysis, they do it over a range of minimum to maximum T-ave.

For ECCS or for peak clad temperature analyses, the lower T-ave is more conservative. So we raised the lower bound. I can't remember what the specific value was. Maybe Carl can speak to it.

MR. O'FARRILL: Carl O'Farrill again, FP&L. We changed that value from 570 degrees to 577 degrees at hot full power and we operated 580 nominally at hot full power.

MEMBER BANERJEE: Okay, thanks.

MR. PARKS: Okay, can I continue with the presentation?

MEMBER BANERJEE: Yes.

MR. PARKS: Okay. So once those parameters were identified and adjusted and the new TCD model incorporated into ASTRUM, all of the WCOBRA/TRAC analyses were re-executed. So they used 124 cases to figure out what the upper tolerance on PCT and oxidation, or I'm sorry upper tolerance limits on those parameters are. And they re-executed all 124 of those cases to get their results.

The previous random numbers were retained, which was acceptable. It enables comparison of the previous distribution of ASTRUM cases through the corrected set of cases. And I presented to the Subcommittee the effects of the changes that the licensee made by comparing the previous analysis to the current analysis. And basically generally what you would see is there is just a shift upward in the peak cladding temperatures and the results were, as I said, re-scattered. And there is a burnup dependence introduced into the analysis, based on the elevated fuel centerline temperature.

MEMBER ABDEL-KHALIK: How was the increase in T-ave justified?

MR. PARKS: I'm sorry, what was the question?

MEMBER ABDEL-KHALIK: I guess the applicant stated that T-ave, the lower limit in the band of the analysis or the sampling I guess from 570 to 577 --

MR. O'FARRILL: This is Carl O'Farrill at FP&L. The 570 was there to accommodate a temperature coast down at end of cycle, should we need to do that.

MEMBER ABDEL-KHALIK: Right.

MR. O'FARRILL: So we are removing that ability to do the temperature coastdown or limiting the ability to do a temperature coastdown at the end of cycle.

MEMBER ABDEL-KHALIK: Because the lower the T-ave you use the higher the PCT that you calculate.

MR. O'FARRILL: Yes.

MEMBER ABDEL-KHALIK: Okay. So this limitation on your ability to coast down, is that a license constraint or is that a condition that you are imposing on them?

MR. O'FARRILL: It is an analysis constraint that is using what we had. That was not a licensing constraint or it is an analysis -- It is an analysis constraint on our operation.

MEMBER ABDEL-KHALIK: An analysis constraint on your operation?

MR. O'FARRILL: We don't typically coast down at the end of cycles. It is there if we have change. We have to design our cycles like 18 months ahead of time. So we make assumptions on capacity factors and outage duration and they are scheduling for the scheduling for the operating cycle. So if any of those things changes, there could be a mismatch. We could be shutting down the cycle early or we could be saying we are going to have to go beyond our point of full energy capability.

And we like to have some capability to do a temperature coastdown because that allows us to take advantage of the negative MTC and maintain full power longer for a few days.

MEMBER ABDEL-KHALIK: But this constraint is on the first power upright cycle. Is this something that you intend to do for the foreseeable future?

MR. O'FARRILL: I think in my experience we have only had two coastdowns at Turkey Point.

MR. CLIFFORD: Well, Carl, maybe it would help. Does Turkey Point have an LCO on T-inlet, which would then be used to infer what T-average was?

MR. O'FARRILL: We have a maximum for T-ave.

MR. CLIFFORD: A maximum?

MR. O'FARRILL: Right and it is DNB-related.

MR. CLIFFORD: Obviously there is a program value for T-ave.

MR. O'FARRILL: There is a program value that goes from hot zero power to hot full power on temperature. And it includes a dead band and that is where the plus or minus three degrees that we have in there; 580 minus is 577 and on the plus side is 583. So we do the analysis on that to allow that accommodation for a dead band for any kind of power maneuvers that you might have to have, we can go off the temperature.

MEMBER ABDEL-KHALIK: This is the dead band for automatic rod control.

MR. O'FARRILL: This is the dead band that the operators have as their target for T-ave when they operate.

MEMBER ABDEL-KHALIK: Okay, it should be roughly the same.

MR. O'FARRILL: Yes. It could be viewed with what we had -- It is typically to have this dead band in there for operations. And it is just the flexibility that we had to accommodate a potential temperature coastdown at the end of cycle. We are just taking away that flexibility.

MEMBER SKILLMAN: And Carl, how hard is that limit at 577 T-ave? Is that going to be a restriction until PAD 5.0 is brought onto your license or do you retain the ability to drop back to 570?

MR. O'FARRILL: We would have to do the re-analysis of the large-break LOCA. So if we do that and when we do that analysis, it is our hope that when

PAD 5.0 comes out, that we are going to gain margin back and we will get to alleviate some of these restrictions, particularly on peaking factor and potentially on T-ave as well. We are going to prioritize what we can or can't do.

MEMBER SKILLMAN: So your T-ave, your minimum T-ave is going to be 577.

MR. O'FARRILL: Yes. For now it will be 577. That is the bottom end of the dead band at hot full power.

MEMBER SKILLMAN: Okay, now is that a tech spec requirement or an admin requirement? How is that --

MR. O'FARRILL: Not, it is a procedural requirement.

MEMBER SKILLMAN: It's a procedural requirement.

MR. O'FARRILL: Right. It is an operating procedure requirement.

MEMBER SKILLMAN: Thank you, Carl.

MEMBER BANERJEE: I guess I don't really understand why you think -- Well maybe I should ask the staff the question.

The way this has been done, if I understand it, it shouldn't be too different from what

PAD 5.0 will come out with. If it was wildly different, we wouldn't feel so happy about it. So I am assuming that PAD 5.0 will come up with essentially the same answers. The thermal conductivities and the full power seconds or whatever is in the core at the start of LOCA is going to be about the same because you have got the right thermal conductivity.

With fission gas, you are roughly right, based on the old model. So why is there hope that PAD 5.0 will alleviate anything? Or is there any hope?

MR. CLIFFORD: I have no expectation that there is going to be some margin recovery.

MEMBER BANERJEE: All right. That makes much more sense. There should be none. If there is, then there is something very strange, counterintuitive. And if there is, you would imagine that there is some pencil sharpening which has produced that margin, not physical phenomenon.

MEMBER SHACK: There may be more room to sharpen the pencil when we have PAD 5.0.

MEMBER BANERJEE: Well obviously. Every time we do an ASTRUM run, there is more pencil sharpening possible.

MEMBER SHACK: No, no, you are not allowed to do that.

MEMBER SCHULTZ: Carl, the way I understand it, you had an opportunity and a need to use these four features to compensate for the new results that were derived. And so you put all of these four into the analysis. You didn't look at each one independently and say we have to add one more. You put four in. And so when you do have the next model to review, you may have an opportunity to gain back one of the pieces.

MEMBER BANERJEE: To put more in.

MR. O'FARRILL: That is correct. We consulted with Westinghouse and we determined what parameters could give us some margin. And we looked at where we had the margin available. And so we made changes to all of them and then the analysis was redone.

MEMBER SCHULTZ: Thank you.

MR. ULSES: I just want to briefly add here and if obviously the change to PAD 5.0 would lead to unexpected results, those changes would be captured by 50.46 reporting requirements. So the staff and I will have an opportunity to review those and to make sure that we agree with why that would happen.

I think generally, however, that we agree with you that we wouldn't expect any major changes but the rule will capture those because the rule covers both an error and a change, not just an error.

MR. CLIFFORD: Ben, do you want me to go on slide 23?

MR. PARKS: Twenty-three, please. Yes, Paul, thank you.

The next two slides deal with the results of the Emergency Core Cooling System evaluation. The first is the observation and then finally I will get to what the actual results were.

Here we see the TCD increased the peak cladding temperature significantly. Significant is a defined term in 50.46(a)(3). This means more than 50 degrees. In this case, it went up a little over 100 degrees. I will show that comparison.

The TCD in my observation made the fuel assembly burnup uncertainty more important. So one of the sample parameters in ASTRUM is the fuel assembly burnup. They seemed not to have from what I could see a big effect on the results of the analysis and then once you put in the corrected fuel performance model, you see the burnup actually does have an important result. And the more burnt fuel assemblies went up in TCD a lot more.

The licensee is compensating margins and did recapture some PCT benefits. And the PCT and oxidation in the net result were higher.

So if we could go to slide 24, I will show you what the actual results were.

In terms of the actual explicit analysis, you can see that the predicted PCT went from 2064 degrees Fahrenheit up about 90 degrees to 2152. And the oxidation results changed up to 10 percent for local and they came own a little bit for the core-wide.

That ASTRUM result does not include a 12 degree Fahrenheit mixed-core penalty. And so the reported PCT is 2164 Fahrenheit.

MEMBER ABDEL-KHALIK: So there is a fuel from another vendor that will be expected in this core?

MR. CLIFFORD: I can answer that. They are both Westinghouse fuel designs but different assembly designs.

MEMBER ABDEL-KHALIK: Different vintage.

MR. CLIFFORD: But they are both Westinghouse.

Do you want slide 25?

MR. PARKS: Sure, bring on 25.

Basically the conclusion here is that the

ASTRUM re-analysis demonstrated compliance with the 50.46 acceptance criteria when accounting for TCD and including those compensatory changes.

And that is all the formal presentation material that I have. So if there are any questions, I can answer those.

MEMBER ABDEL-KHALIK: If we go back to page 22, have these three changes not been implemented, do you know what that calculated peak clad temperature would have been? And did you do sort of a sanity check?

MR. PARKS: The peak clad temperature I would have expected it to go over 2200. I expect that one of the cases around 1950, I think, would have gone up a lot higher. So not necessarily the currently limiting case would have been a lot different but another one would have been more limiting.

MEMBER BANERJEE: Ben, the currently the limiting case and you told us about this at the Subcommittee meeting but please refresh my memory. What were the full parameters that were adjusted for that case? Was it --

MR. PARKS: Dr. Banerjee, the two high PCT cases were similar in some respects that they didn't have a particularly high burnup and they didn't have anything particularly out of the ordinary with respect to the power distribution. They were high PCT because from what I could tell, they had a pretty assumption regarding ECCS performance. Their accumulator volume may have been a little low. The accumulator pressure was low so they weren't getting the accumulator injection quickly. And the overall SI performance temperatures would have been high, too much NOP in the SI cooling water.

And so based on that, the two running cases switched places and went up in PCT a bit but it was really the SI that caused them to be so high.

MEMBER BANERJEE: Thanks. That was more or less the situation, if I remember, in the previous, before the TCD correction was made as well. It was the degraded SI that caused the problem. Right? Is that consistent with my memory?

MR. PARKS: Yes, and then there were a population of cases that were down to two limiting ones. They had been in the neighborhood of 16 to 1850 and they came up higher. A lot of them came up to about 1950. And those were the ones that were affected by burnup and then they started coming down again once you put in the power distributions. So some of the oddly pieced ones, they came back down.

MEMBER BANERJEE: Thanks.

MEMBER SHACK: Any further questions from the Committee?

Any comments from anyone in the audience?

MR. PAIGE: Just one comment. I just want to give a special thanks to Ben Parks. He is actually on vacation right now and he took time out to present during today's meeting.

MEMBER BANERJEE: So he is really on a beach. Right?

MR. PARKS: I am not on a beach. I am in the office.

MEMBER BANERJEE: You are where, Ben?

MEMBER POWERS: He is in an office.

MEMBER SHACK: Is there anybody on the phone line?

MEMBER SIEBER: There shouldn't be. It's a closed session.

MEMBER SHACK: Thank you, very much to the licensee and the staff for a very good presentation. Again, I think we went through all this so I think it will be very helpful to the Committee in reaching their decision.

CHAIR ARMIJO: Okay, well thank you very much. I would like to add my thanks to Paul and FPL for their efforts. I know we have got a lot of work to do between now -- I think we need at least a ten minute break. Let's do that and we are off the record.

(Whereupon, at 5:15 p.m., the foregoing proceeding was adjourned.)

Presentation to the ACRS Full Committee - 592nd Meeting

Briefing on EPR Design Certification Application Safety Evaluation Report with Open Item for Chapters 6, 7, 11, 13, 15, 16, and 18

Getachew Tesfaye
Project Manager

March 8, 2012

- **Additional Presenters:**
  - Shie-Jeng Peng, Chapter 6
  - Jean-Claude Dehmel, Chapter 11
  - Shanlai Lu, Chapter 15

Major Milestones Chronology

United States Nuclear Regulatory Commission
Protecting People and the Environment

12/02/2004	Pre-application activities began
12/11/2007	Design Certification Application submitted
02/25/2008	Application accepted for review (docketed)
03/26/2008	Original review scheduled published
01/29/2009	Phase 1 review completed
04/08/2010	ACRS full committee briefing on Chapters 2, 4, 5, 8, 10, 12, and 17
08/10/2011	U.S. EPR FSAR, Revision 3 submitted
02/09/2012	Phase 2 review completed
02/23/2012	Phase 3, ACRS Subcommittee presentation completed
03/08/2012	ACRS full committee briefing on Chapters 6, 7, 11, 13, 15, 16, and 18
05/10/2012	ACRS full committee briefing on Chapters 3, 9, 14, and 19 is planned

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Review Schedule

Task	Target Date
Phase 1 - Preliminary Safety Evaluation Report (SER) and Request for Additional Information (RAI)	Completed
Phase 2 - SER with Open Items	Completed
Phase 3 – Advisory Committee on Reactor Safeguards (ACRS) Review of SER with Open Items	July 2012
Phase 4 - Advanced SER with No Open Items	Schedule under review
Phase 5 - ACRS Review of Advanced SER with No Open Items	Schedule under review
Phase 6 – Final SER with No Open Items	Schedule under review
Rulemaking	Schedule under review

Review Strategy

- Pre-application activities
- Frequent interaction with the applicant
  - Teleconferences
  - Audits
  - Public meetings
- Use of Electronic RAI System (eRAI)
- Phase discipline

Chapter 6 - Engineered Safety Features

SRP Section/Application Section		Number of OI
6.1.1	Metallic Materials	0
6.1.2	Organic Materials	2
6.2.1	Containment Functional Design (with exception of 6.2.1.2)	12
6.2.1.2	Subcompartment Analysis*	Not delivered in Phase 2
6.2.2	Containment Heat Removal*	Not delivered in Phase 2
6.2.3	Secondary Containment Functional Design	3
6.2.4	Containment Isolation System	2

*The safety evaluation for these Sections was not delivered in the Phase 2 SE for Chapter 6.

Chapter 6 - Engineered Safety Features (continued)

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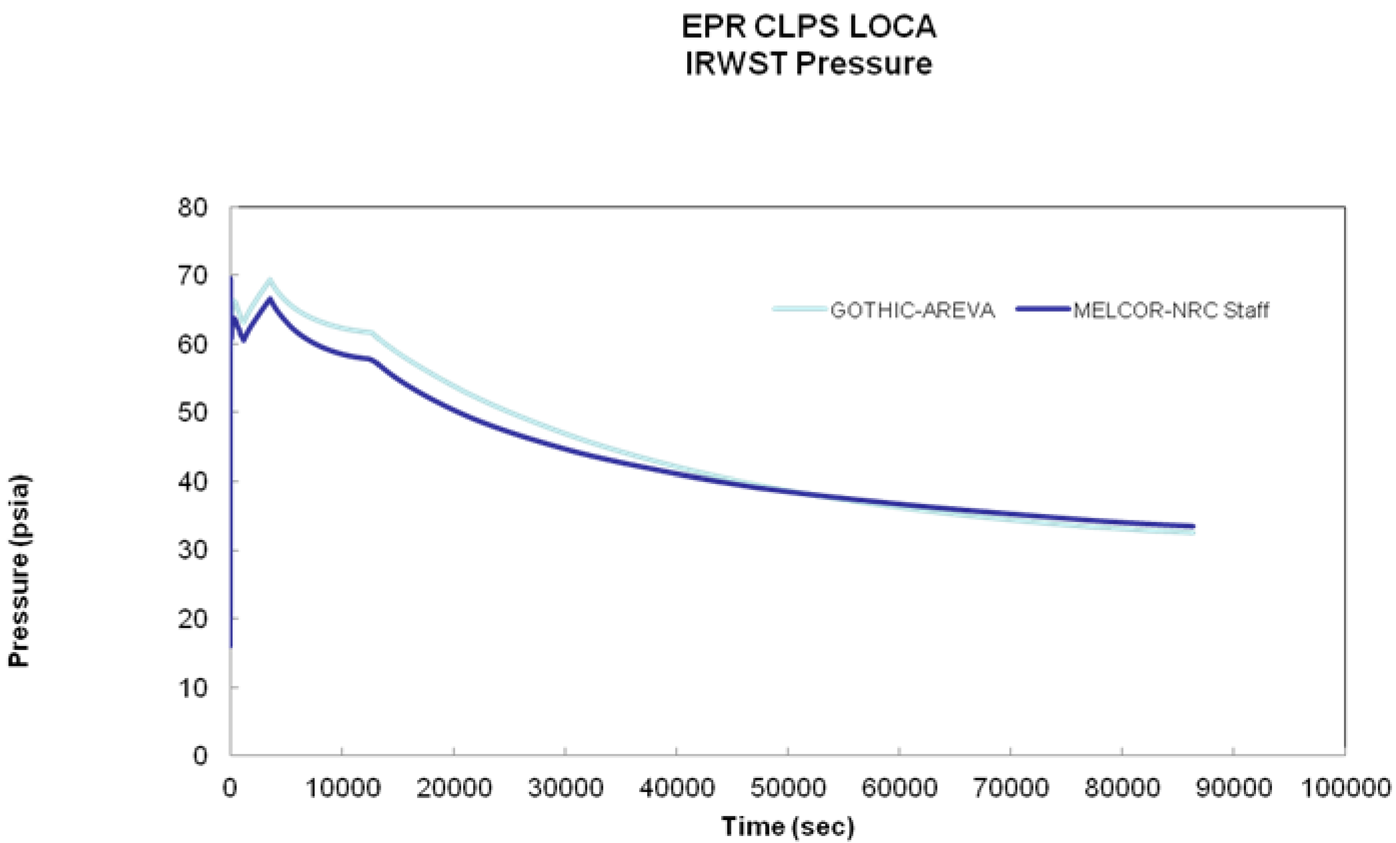
SRP Section/Application Section		Number of OI
6.2.5	Combustible Gas Control in Containment	6
6.2.6	Containment Leakage Testing	0
6.2.7	Fracture Prevention of Containment Pressure Vessel	0
6.3	Emergency Core Cooling System	2
6.4	Habitability Systems	4
6.5	Fission Product Removal and Control Systems	4
6.6	Inservice Inspection of ASME Class 2 and 3 Components	0
Totals		**35**

Chapter 6 - Key Confirmatory Analysis Performed by the NRC staff



Chapter 6 - Key Confirmatory Analysis Performed by the NRC staff (Continued)

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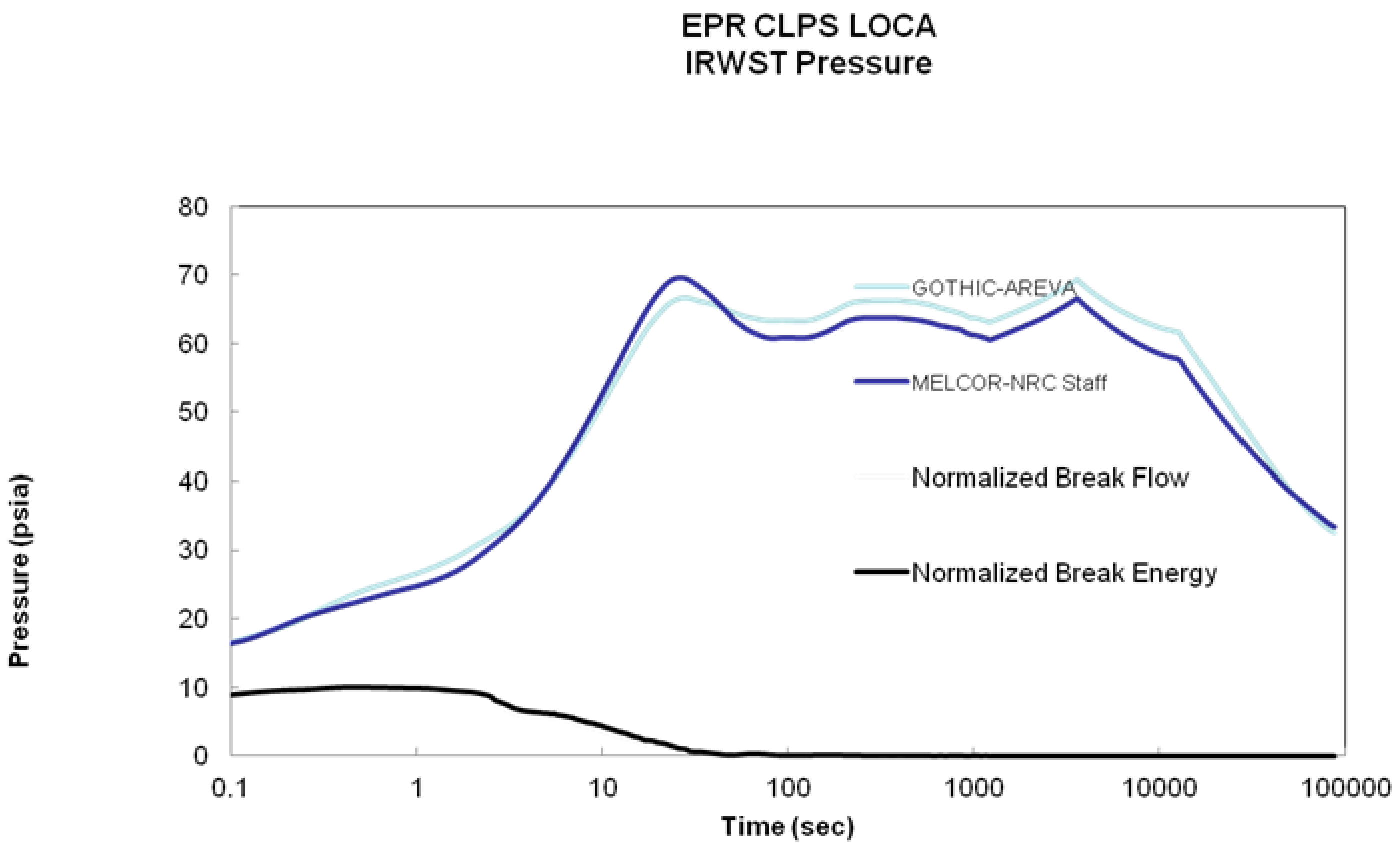
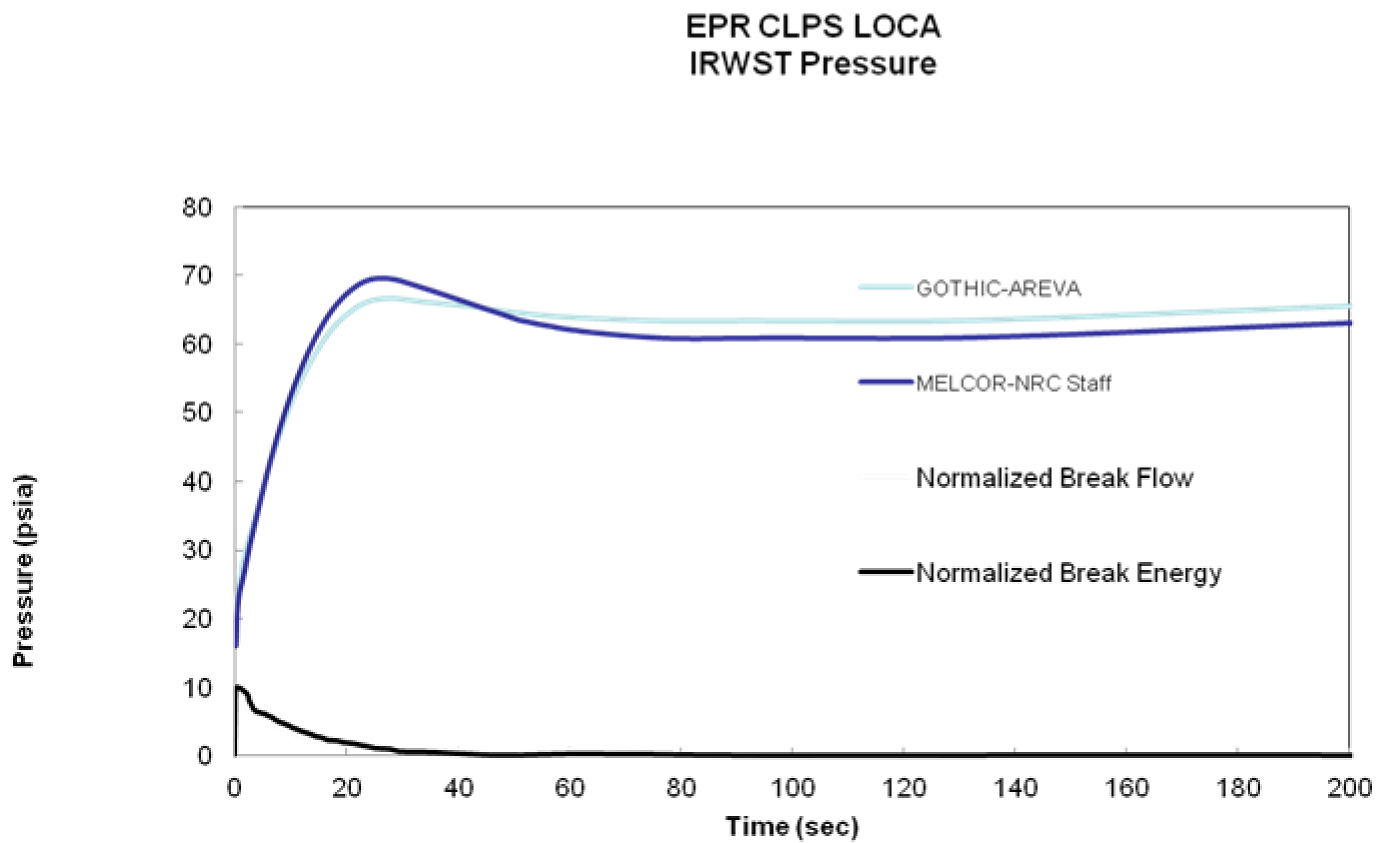


Chapter 6 - Key Confirmatory Analysis Performed by the NRC staff (Continued)



Chapter 7 – Instrumentation and Controls

SRP Section/DCA Section		No. of Open Items
7.1	Introduction	19
7.2	Reactor Trip System	0
7.3	Engineered Safety Features Systems	3
7.4	Systems Required for Safe Shutdown	1
7.5	Information Systems Important to Safety	2
7.6	Interlock Systems Important to Safety	0
7.7	Control Systems Not Required for Safety	1
7.8	Diverse Instrumentation and Control Systems	8
7.9	Data Communication Systems	2
Totals		**36**

Chapter 11 – Radioactive Waste Management

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SRP Section/Application Section		Number of Open Items
11.1	Source Terms	0
11.2	Liquid Waste Management System	8
11.3	Gaseous Waste Management System	4
11.4	Solid Waste Management Systems	6
11.5	Process and Effluent Radiological Monitoring and Sampling Systems	6
Totals		24

Chapter 11 - Key Confirmatory Analysis Performed by the NRC staff

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- FSAR Section 11.2 – Liquid Effluents

  - Annual average liquid effluent releases (curies per year)
  - Annual average liquid effluent concentrations (uCi/ml) and compliance with Part 20, App. B, Table 2, Column 2 limits
  - Annual average offsite doses (mrem per year) and compliance with limits of Part 20.1301 and 20.1302 and Part 50, App. I design objectives.
  - Assessment associated with the assumed failure of a tank containing radioactive materials and radiological impact on groundwater

Chapter 11 - Key Confirmatory Analysis Performed by the NRC staff (Continued)

- FSAR Section 11.3 - Gaseous Effluents

  - Annual average gaseous effluent releases (curies per year)
  - Annual average gases effluent concentrations (uCi/ml) and compliance with Part 20, App. B, Table 2, Column 1 limits
  - Annual average offsite doses (mrem per year) and compliance with limits of Part 20.1301 and 20.1302 and Part 50, App. I design objectives
  - Assessment associated with the assumed failure of a power cycle offgas treatment system component containing radioactive materials and radiological impact at the EAB

Chapter 13 - Conduct of Operations

SRP Section/Application Section		Status Number of OI
13.1	Organizational Structure of Applicant	0
13.2	Training	0
13.3	Emergency Planning	0
13.4	Operational Program Implementation	0
13.5	Plant Procedures	0
13.6	Security	3
13.7	Fitness for Duty	0
Totals		3

Chapter 15 - Transient and Accident Analyses

SRP Section/Application Section		Number of OI
15.0.1 and 15.0.2	Radiological Consequence Analysis and Computer Codes Used in Transient and Accident Analysis	3
15.0.3	Radiological Consequences of Design Basis Analysis	2
15.1	Increase in Heat Removal by the Secondary System	0
15.2	Decrease in Heat Removal by the Secondary System	0
15.3	Decrease in Reactor Coolant System Flow Rate	0
15.4	Reactivity and Power Distribution Anomalies	1

Chapter 15 - Transient and Accident Analyses (continued)

SRP Section/Application Section		Number of OI
15.5	Increase in Reactor Coolant Inventory	0
15.6.1	Inadvertent Opening of a Pressurizer Safety Relief Valve	0
15.6.3	Steam Generator Tube Failure (PWR)	0
15.6.5*	Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	10
15.8	Anticipated Transients Without Scram	0
Totals		16

*The Phase 2 safety evaluation for Section 15.6.5 does not contain the GSI-191 evaluation of in-vessel downstream effects. This topic will be delivered in Phase 4 for the U.S. EPR design certification application.

Chapter 15 - Key Confirmatory Analysis Performed by the NRC staff

- As part of Chapter 15 review, staff performed confirmatory analyses to identify and resolve issues

- The analyses are divided into the following categories:

  1. LOCA – Thermal Hydraulic Analysis
  2. Transient And Accident Analysis – Neutronics analysis coupled with thermal hydraulic analysis
  3. Fuel Thermal Performance Analysis

Chapter 15 - Key Confirmatory Analysis Performed by the NRC staff (Continued)

1. LOCA – Thermal Hydraulic Analysis
   - LBLOCA and SBLOCA were analyzed using TRACE, RELAP-5 and S-RELAP5
   - Staff's confirmatory analyses showed that there is margin to the 50.46 limits
2. Transient And Accident Analyses
   - Confirmed the applicability of the on-line low DNBR and High Linear Power Density protection system
   - For the rod ejection accident, compared the peak power value TRACE/PARCS was used to perform the analysis and SCALE code package was used to generate the cross-section library
3. Fuel Thermal Performance Analysis
   - FRAP-CON and RODEX-4 were used to quantify the initial stored energy calculation error. As the result of this effort, two generic Information Notices (IN 2009-23 and IN 2011-21) were issued

Chapter 16 – Technical Specifications

SRP Section/Application Section		**Number of SER Open Items**
16.4.1	General	0
16.4.2	Use and Application	0
16.4.3	Safety Limits	0
16.4.4	Limiting Condition for Operation and Surveillance Requirement Applicability	0
16.4.5	Reactivity Control System	0
16.4.6	Power Distribution Limits	0
16.4.7	Instrumentation	11
16.4.8	Reactor Coolant System	9
Continued on next page		

Chapter 16 – Technical Specifications (Continued)

SRP Section/Application Section		Number of SER Open Items
16.4.9	Emergency Core Cooling System (ECCS)	2
16.4.10	Containment Systems	1
16.4.11	Plant Systems	4
16.4.12	Electric Power Systems	2
16.4.13	Refueling Operations	0
16.4.14	Design Features	1
16.4.15	Administrative Controls	0
Totals		30

Chapter 18 – Human Factors Engineering

SRP Section/Application Section		Status Number of OI
18.1	HFE Program Management	0
18.2	Operating Experience Review	0
18.3	Functional Requirements Analysis and Function Allocation	0
18.4	Task Analysis	0
18.5	Staffing and Qualifications	0
18.6	Human Reliability Analysis	0

Chapter 18 – Human Factors Engineering (Continued)

SRP Section/Application Section		Status Number of OI
18.7	Human-System Interface Design	0
18.8	Procedure Development	0
18.9	Training Program Development	0
18.10	Verification and Validation	0
18.11	Design Implementation	0
18.12	Human Performance Monitoring	0
Various Sections	Human Factors Engineering	0
Totals		**0**

END

NEXT STEP

ACRS Full Committee Phase 3 Briefing on the Remaining Chapters:
Chapters 3, 9, 14, and 19
May 10-12, 2012

Presentation to ACRS U.S. EPR^{TM} Design Certification Chapters 6, 7, 11, 13, 15, 16, 18

March 8, 2012

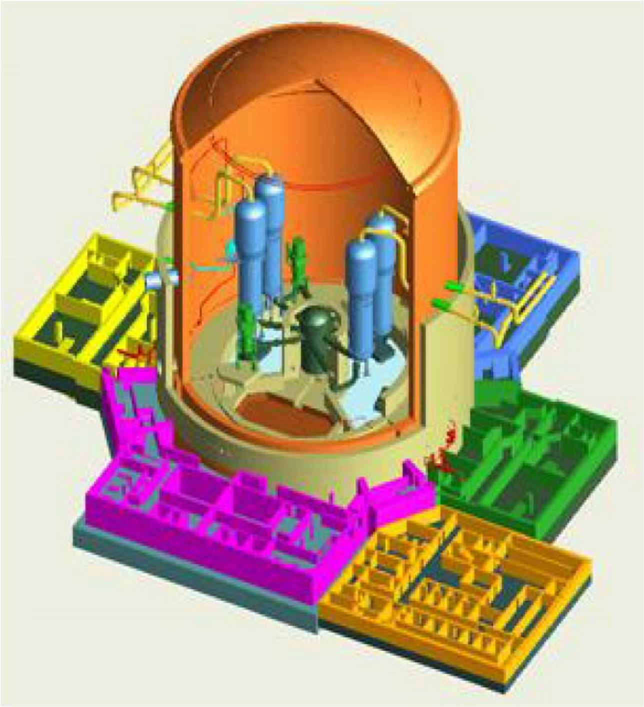
AREVA NP

Outline

- Introduction
- Overview of U.S. EPR Design
  - EPR Development Objectives
  - Major Design Features
  - Main Safety Systems
  - Protection From External Hazards
  - Severe Accident Mitigation
- Overview of U.S. EPR Design Certification Application
  - Chapters 6, 7, 11, 13, 15, 16, 18

EPR Development Objectives

- **Evolutionary design based on existing PWR operating experience, construction experience and Research & Development**
  - **Improved economics**
    - Reduce generation cost by at least 10%
    - Simplify operations and maintenance
    - 60-year design life
  - **Improved Safety**
    - Increase design margins
    - Increase redundancy and physical separation of safety trains
    - Reduce core damage frequency
    - Accommodate severe accidents and external hazards
    - Reduce occupational exposure and low level waste



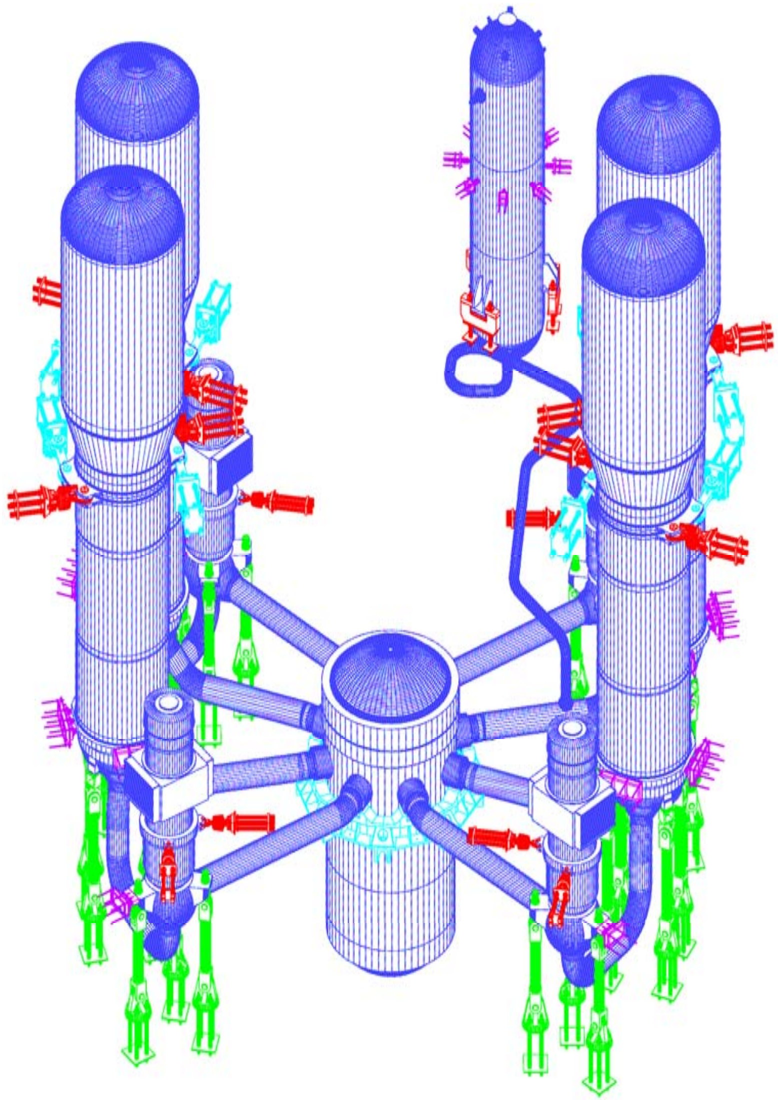
Major Design Features

- **Nuclear Island**
  - Proven Four-Loop RCS Design
  - Four-Train Safety Systems
  - Double Containment
  - In-Containment Refueling Water Storage
  - Severe Accident Mitigation
  - Separate Safety Buildings
  - Advanced 'Cockpit' Control Room

- **Electrical**
  - Shed Power to House Load
  - Four Emergency Diesel Generators
  - Two Smaller, Diverse Station Blackout Diesel Generators

- **Site Characteristics**
  - Airplane Crash Protection (military and commercial)
  - Explosion Pressure Wave

Reflects full benefit of operating experience and 21st century requirements



- Conventional 4-loop PWR design, proven by decades of design, licensing and operating experience
- NSSS component volumes increased compared to existing PWRs, increasing operator grace periods for many transients and accidents

A solid foundation of operating experience

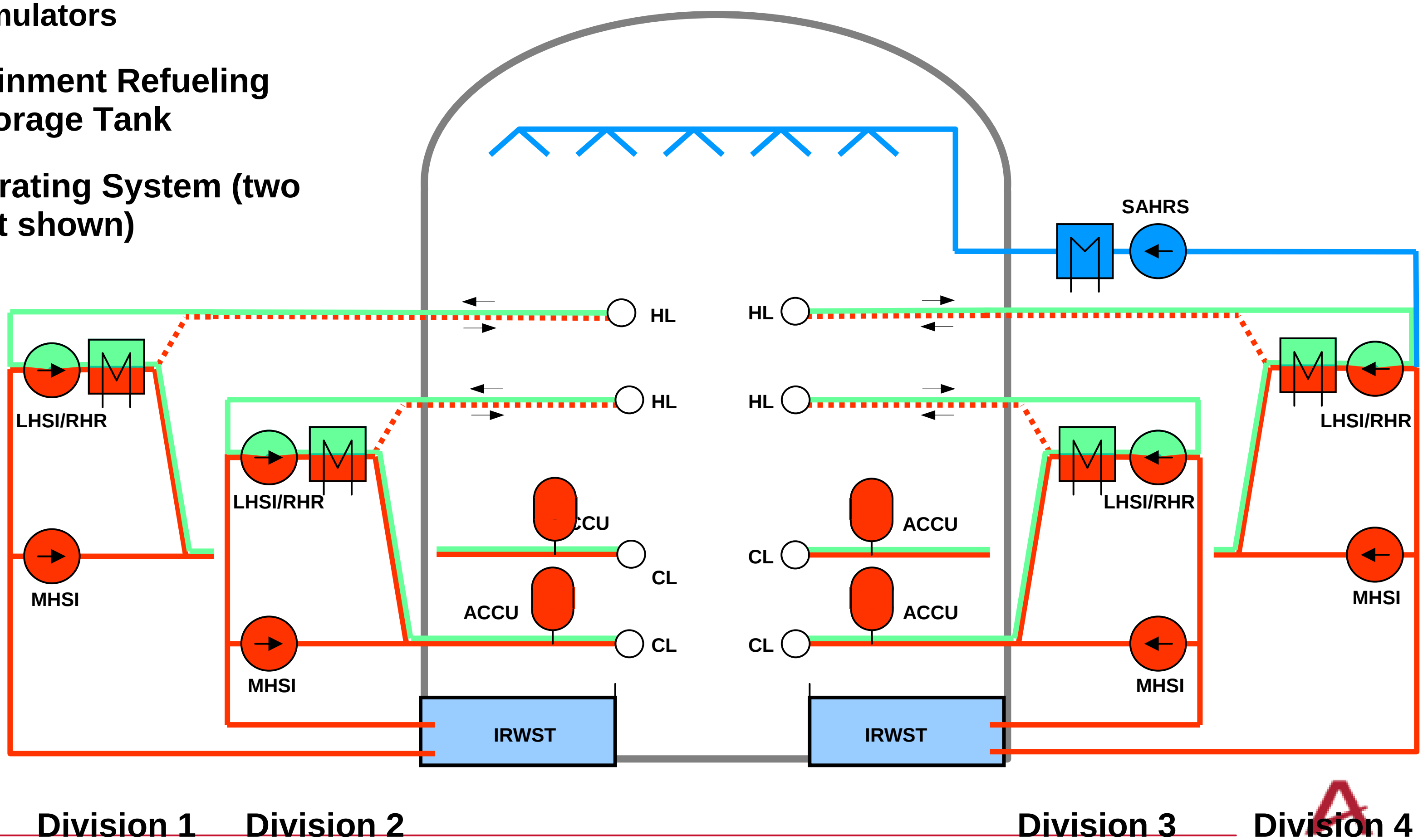
The Four Train (N+2) Concept

Each safety train is independent and located within a physically separate building

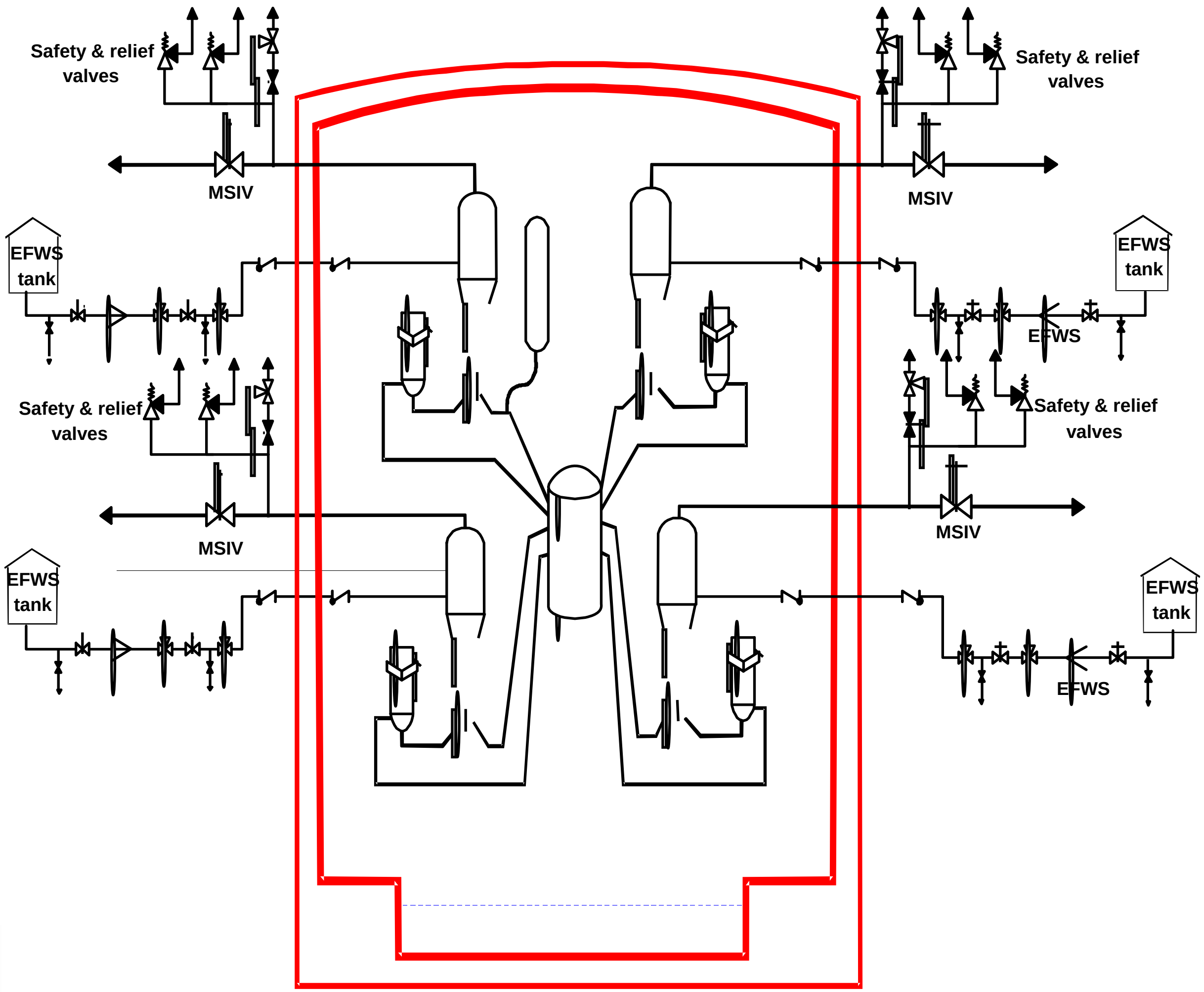
Main Safety Systems

- Four train Safety Injection (SI) system
  - Medium head SI pumps
  - Combined Residual Heat Removal System / Low Head Safety Injection
  - Accumulators
- In-Containment Refueling Water Storage Tank
- Extra Borating System (two trains not shown)

- Non-safety containment spray for severe accident mitigation



Main Safety Systems
Secondary Side

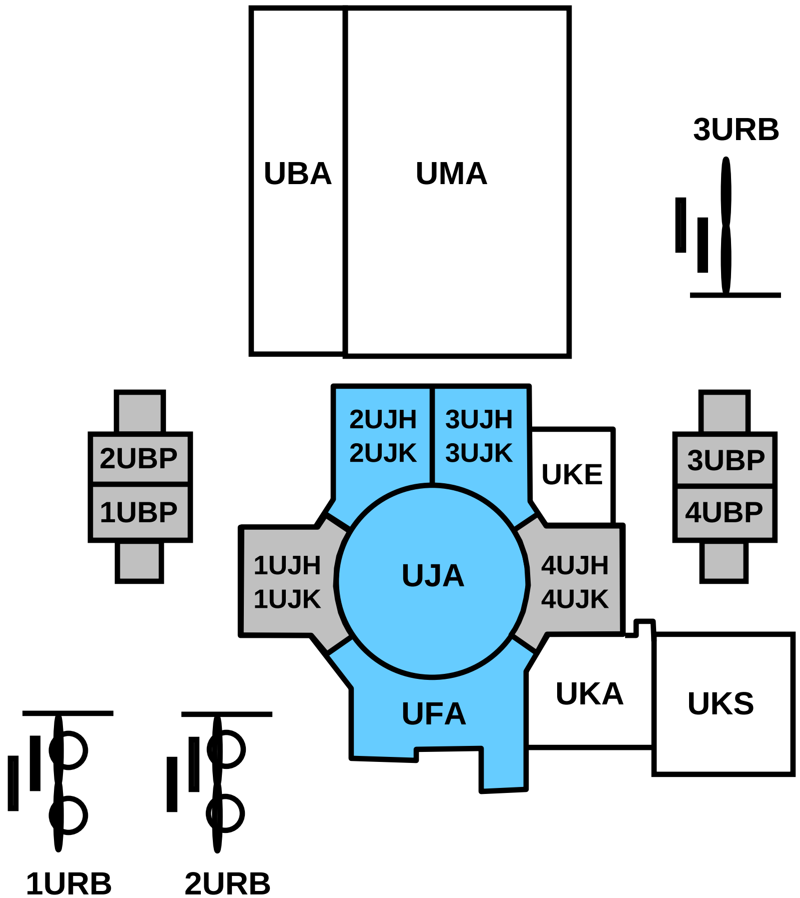


- Safety-related main steam relief train
- Four separate Emergency Feed Water Systems (EFWS)
- Separate power supply for each
- 2/4 EFWS also powered by Station Black Out (SBO) diesels
- Interconnecting headers at EFWS pump suction & discharge

Protection From External Hazards
Shielded Containment

- Inner wall post-tensioned concrete with steel liner
- Outer wall reinforced concrete
- Protection against airplane crash
- Protection against external explosions
- Annulus filtered to reduce radioisotope release

Protection From External Hazards



UBA	Switchgear Building
UBP	Emergency Power Generating Building
UFA	Fuel Building
UJA	Reactor Building
UJH	Safeguard Building Mechanical
UJK	Safeguard Building Electrical
UKA	Nuclear Auxiliary Building
UKE	Access Building
UKS	Radioactive Waste Processing Building
UMA	Turbine Building
URA	Cooling Tower Structure
URB	Essential Service Water Cooling Tower Structure

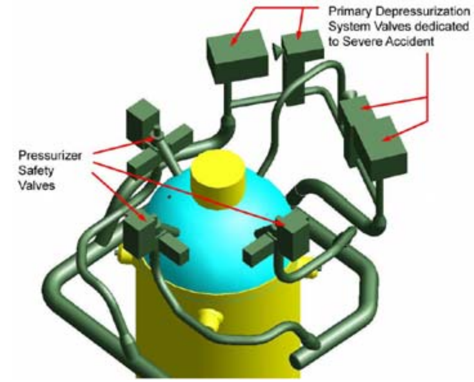
PROTECTED BY SHIELD BUILDING

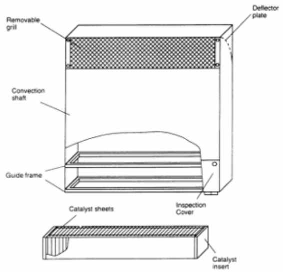
PROTECTED BY PHYSICAL SEPARATION

NOT PROTECTED

Severe Accident Mitigation

- Prevention of high-pressure melt using Primary Depressurization System
- Passive ex-vessel melt stabilization, conditioning and cooling
- Long-term melt cooling and containment protection using active cooling system
- Control of H_2 concentration using passive autocatalytic recombiners





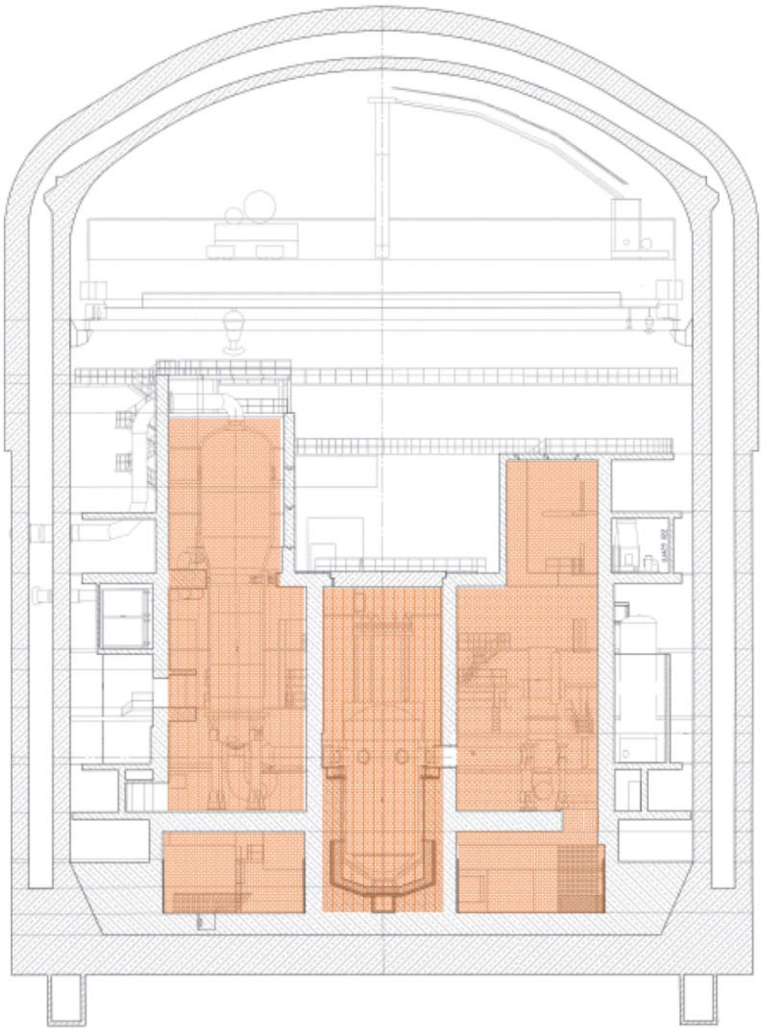
U.S. EPR Design Certification Application

- **U.S. EPR design reflects an evolutionary, active plant design**
- **U.S. EPR applies proven analytical methodologies**
- **FSAR consistent with key NRC guidance documents**
  - **Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants (Light Water Reactor Edition)”**
  - **NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants“**
  - **Exemptions and exceptions minimized**
  - **No RTNSS**

Chapter 6: Engineered Safety Features

- **Topics**
  - **Engineered Safety Features Materials – 6.1**
  - **Containment Systems – 6.2**
    - (excluding 6.2.2 Containment Heat Removal Systems – later)
  - **Emergency Core Cooling System – 6.3**
    - (excluding GSI-191 – later)
  - **Habitability Systems – 6.4**
  - **Fission Product Removal and Control Systems – 6.5**
  - **Inservice Inspection of Class 2 and 3 Components – 6.6**
  - **Extra Borating System – 6.8**

Containment Design Features



- Post-tensioned concrete containment with steel liner
- Reinforced concrete Shield Building wall
- Containment Free Volume = 2.8 Mft^3
- Containment Inside Diameter = 153.5 ft.
- Containment Wall Thickness = 4.3 ft.
- Design pressure = 62 psig
- In-Containment Refueling Water Storage Tank (~500,000 gallons)
- Two-zone containment (equipment area and service area)
- CONVECT system of rupture and convection foils and dampers connect containment zones during high energy line breaks (HELBs)
- Passive hydrogen reduction system
- Filtered, vented annulus to prevent uncontrolled releases to environment

U.S. EPR does not rely on safety related fan coolers or containment sprays

ECCS Design Features

- **Four independent, front line Safety Injection Systems**
  - **Accumulators**
  - **Medium head safety injection**
  - **Low head safety injection**
- **Automatic partial cooldown of steam generators (SGs) on safety injection system (SIS) actuation signal reduces primary pressure to below discharge head of the medium head safety injection (MHSI) pumps**
- **In-Containment Refueling Water Storage Tank (IRWST)**
  - **Single source of emergency core cooling system (ECCS) water**
  - **Eliminates need to switch to a recirculation injection mode**
  - **Sufficient static head to the suction of the SIS pumps (no piggyback operation)**
  - **Sufficient inventory during shutdown to fill reactor cavity, internal storage pool, reactor building transfer pool and the reactor coolant system**
  - **Sufficient inventory for flooding a core melt during a severe accident**
- **Manual alignment of low head safety injection (LHSI) to hot leg nozzles at 60 minutes to suppress core boiling**

Chapter 7
Instrumentation and Controls

- **Topics included in U. S. EPR Chapter 7**
  - **U.S. EPR I&C Systems – 7.1**
  - **Reactor Trip System – 7.2**
  - **Engineered Safety Features Systems – 7.3**
  - **Systems Required for Safe Shutdown – 7.4**
  - **Information Systems Important to Safety – 7.5**
  - **Interlock Systems Important to Safety – 7.6**
  - **Control Systems Not Required for Safety – 7.7**
  - **Diverse I&C Systems – 7.8**
  - **Data Communication Systems – 7.9**

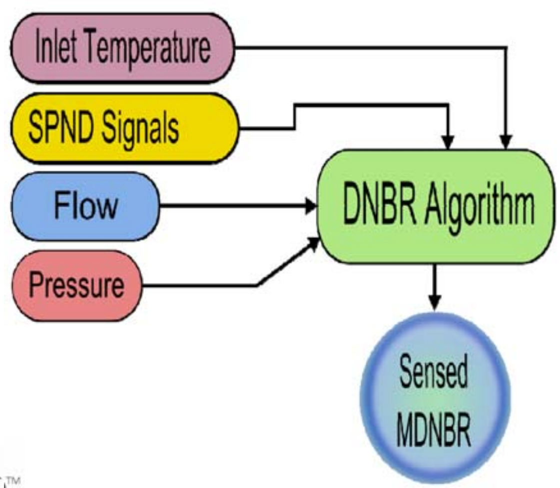
Chapter 7
Instrumentation and Controls

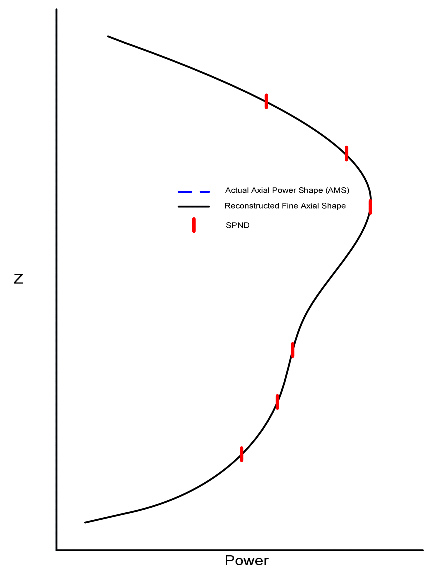
System	Basic Functions	Safety Classification	Number of Channels	Technology
Safety Information and Control System (SICS)	**Backup HMI for accident mitigation, safe shutdown, and severe accident mitigation**	**S**	**4**	**Hardwired/TXS (QDS)**
Protection System (PS)	**Actuation of safety systems (reactor trip/ESF)**	**S**	**4**	**TXS (microprocessor PE)**
Safety Automation System (SAS)	**Control of safety systems**	**S**	**4**	**TXS (microprocessor PE)**
Signal Conditioning and Distribution System (SCDS)	**Signal conditioning and distribution of field input signals**	**S**	**4**	**TXS (Electronic and PLD based PE)**
Priority Actuation and Control System (PACS)	**Prioritizes commands from various systems, actuates and monitors actuators**	**S**	**4**	**TXS (100% Tested PLD based PE)**
Diverse Actuation System (DAS)	**Diverse actuation of safety systems (reactor trip/ESF)**	**NS-AQ**	**4**	**Not microprocessor PE**
Process Information and Control System (PICS)	**Primary HMI for all plant operations**	**NS-AQ**	**4**	**Industrial platform (microprocessor PE)**
Reactor Control, Surveillance and Limitation (RCSL)	**Control of reactivity**	**NS-AQ**	**4**	**TXS (microprocessor PE)**
Process Automation System (PAS)	**All other process control functions**	**NS**	**4 NI 2 TI**	**Industrial platform (microprocessor PE)**

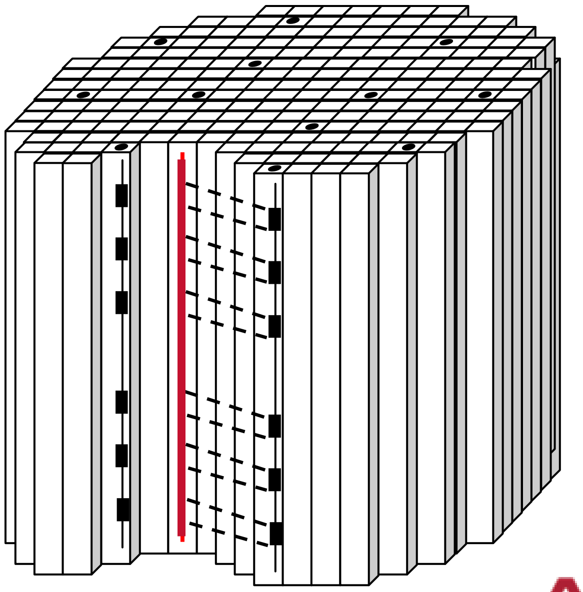
- **Safety Related DCS Design Features**
  - **Digital I&C technology that includes continuous online self-testing and diagnostics that allow early detection of failures and improved human-machine interfaces (HMI)**
  - **Robust I&C architecture that optimizes plant safety through defense-in-depth, diversity, redundancy, independence and priority setting**
  - **High degree of automation that improves plant operation, reduces operator burden, and improves situational awareness during normal and accident conditions**
  - **PACS - Priority module utilizes programmable logic device (PLD) that is 100% combinatorially tested to protect against common cause failures of PACS**
  - **DAS – Diverse Actuation System utilizes non-microprocessor based platform to protect against software common cause failures of the Protection System**
  - **Communication**
    - PS and SAS connections are unidirectional and electrically isolated from non-safety systems
    - Interdivisional safety function processor interface
      - PS divisions are physically separated with interdivisional communication for actuation voting using optical communication medium
      - SAS divisions are physically separated with interdivisional communication when necessary to perform a safety function (e.g., CCW interlocks to RCP thermal barrier coolers). Connections are via optical communication medium.
    - Service Unit provides maintenance access to PS and SAS
      - Not continuously connected
      - Switch provided that physically restricts Service Unit to only allow connection to one division at a time

U. S. EPR Unique Design – Self-Powered Neutron Detector (SPND) Based Reactor Trips:

- **Low DNBR trip and High Linear Power Density trip are only reactor trips that rely on SPNDs**
- **Same as standard reactor trips, except that all 72 SPND signals are needed in all four PS divisions to recreate complete flux distribution**
- **Benefits of Design**
  - **Provides more direct measurement of neutron flux (Clause 6.4 of IEEE 603)**
  - **Reduces uncertainties and assumptions with excore based trips**







Chapter 11: Radioactive Waste Management

- Topics
  - Source Terms – 11.1
  - Liquid Waste Management System – 11.2
  - Gaseous Waste Management Systems – 11.3
  - Solid Waste Management Systems – 11.4
  - Process and Effluent Radiological Monitoring and Sampling Systems – 11.5

- Highlights of US EPR Radioactive Waste Management design:
  - Design basis and realistic source terms defined using typical industry practice
  - Liquid waste processing systems consist of an evaporator/vapor compressor package, centrifuge and demineralizer to provide operational flexibility
  - Solid waste volume reduction provided with shredding device, solid waste drying, compaction and sorting box
  - Main steam radiation monitoring provided as primary indication of a steam generator tube rupture

Liquid, gas and solid waste effluent concentrations compliant with 10CFR20 Appendix B

Chapter 13:
Conduct of Operations

- **Topics**
  - **Organizational Structure of Applicant – 13.1**
  - **Training – 13.2**
  - **Emergency Planning – 13.3**
  - **Operational Program Implementation – 13.4**
  - **Plant Procedures – 13.5**
  - **Security – 13.6**
  - **Fitness for Duty – 13.7**
- **Highlights:**
  - **Conduct of Operations is primarily responsibility of COL applicant**
  - **Emergency Operating Procedures (EOPs) will be developed based on symptom-based accident management guidelines**

Chapter 15: Transient and Accident Analyses

- **Topics**
  - **Increase in Heat Removal by the Secondary System – 15.1**
  - **Decrease in Heat Removal by the Secondary System – 15.2**
  - **Decrease in Reactor Coolant System Flow Rate – 15.3**
  - **Reactivity and Power Distribution Anomalies – 15.4**
  - **Increase in Reactor Coolant Inventory – 15.5**
  - **Decrease in Reactor Coolant Inventory – 15.6**
  - **Radioactive Release from a Subsystem or Component – 15.7**
  - **Anticipated Transients Without Scram – 15.8**
  - **Spent Fuel Pool Criticality and Boron Dilution Analysis – 15.10**

Chapter 15:
Transient and Accident Analyses

- **U.S. EPR design features are similar to previous PWR designs**
- **Unique features important to transient and accident analyses**
  - **Front-line safety systems**
    - Four train systems (Safety Injection/Residual Heat Removal, Emergency Feedwater and Main Steam Relief Trains)
    - Two train systems (Extra Borating System and Annulus Ventilation)
  - **In-Containment Refueling Water Storage Tank (IRWST)**
    - Source of ECCS water
    - No switchover needed
  - **Operator action times for Chapter 15 events**
    - 30 minutes for actions from inside of Main Control Room
    - 60 minutes for actions outside of Main Control Room
  - **Automatic partial cooldown of Steam Generators on SI signal for LOCA mitigation**
  - **Safety-related alarm on high activity in steam line for SGTR mitigation**
  - **Automatic trip of reactor coolant pumps on coincident SI actuation signal and low Delta-Pressure across the reactor coolant pumps for LOCA mitigation**
  - **Low DNBR and High Linear Power Density reactor trip functions utilizing in-core measurements of local core power distributions with SPNDs**
  - **Alternative Source Term used in radiological consequences analyses**

Chapter 16: Technical Specifications

- Topics – US EPR Generic Technical Specifications (GTS) follow the format and content of the Improved Standard Technical Specifications
  - 1.0 Use and Application
  - 2.0 Safety Limits
  - 3.0 LCOs
  - 3.1 Reactivity Control
  - 3.2 Power Distribution Limits
  - 3.3 Instrumentation
  - 3.4 Reactor Coolant System
  - 3.5 Emergency Core Cooling Systems
  - 3.6 Containment Systems
  - 3.7 Plant Systems
  - 3.8 Electrical Power Systems
  - 3.9 Refueling Operations
  - 4.0 Design Features
  - 5.0 Administrative Controls
  - Bases
- Since the U.S. EPR is an active, 4-loop PWR, Standard Technical Specifications (STS), NUREG 1431 Revision 3.1 was chosen as the primary model for the GTS (other Improved STS NUREGs and precedents were used as appropriate)
- "N+2" safety concept utilizing four (4) trains of safety systems:
  - One train is assumed to be lost due to postulated single failure
  - One train may be out of service for maintenance/surveillance
  - One train may be lost due to initiating event (e.g., ECCS injection line break)
  - One train is available to perform accident mitigating function

Chapter 16:
Technical Specifications - continued

- **Differences reflect U.S. EPR specific design and supporting safety analysis**
  - **Section 3.3, Instrumentation - U.S. EPR's digital I&C System incorporates reactor trip and ESF functions credited in safety analysis thereby reducing the number of subsections**
    - LCOs and Actions are component-based rather than function-based since single components may support several functions
    - Includes Diverse Actuation System (DAS)
  - **Section 3.4, Reactor Coolant System - Revised to allow limited 3-loop operation (consistent with U.S. EPR safety analysis)**
  - **Section 3.6, Containment Systems - Did not include containment spray (not credited in U.S. EPR safety analysis)**
    - Passive heat sinks (concrete walls and steel structures) inside containment are credited post-LOCA
    - Containment heat removal is performed by recirculation of reactor coolant from the IRWST, through the LHSI heat exchangers, to the RCS, and through the postulated break back to the containment and IRWST
    - Doses mitigated using Annulus Ventilation System
  - **Section 3.8, Electrical Power Systems - utilizes four EDGs**
    - Alternate feeds can be established between Division 1 and 2 (one divisional pair) or Division 3 and 4 (another divisional pair)
    - Alternate feeds provide power to required safety-related SSCs that do not have the required redundancy when the divisional EDG is out of service (e.g., annulus ventilation)

Chapter 18:
Human Factors Engineering

- **Topics**
  - Human Factors Engineering Program Management – 18.1
  - Operating Experience Review – 18.2
  - Functional Requirements Analysis and Functional Allocation – 18.3
  - Task Analysis – 18.4
  - Staffing and Qualifications – 18.5
  - Human Reliability Analysis – 18.6
  - Human System Interface Design – 18.7
  - Procedure Development – 18.8
  - Training Program Development – 18.9
  - Verification and Validation – 18.10
  - Design Implementation – 18.11
  - Human Performance Monitoring – 18.12

Chapter 18: Human Factors Engineering

- **Human Factors Engineering (HFE) Highlights:**
  - **HFE program is described in nine (9) implementation plans reviewed by the NRC staff**
    - Program is implemented by COL applicant
    - Tier 1 Design Acceptance Criteria (DAC) are provided for HFE program
    - HFE program is consistent with NUREG 0711, Revision 2
  - **Scope of HFE Program – MCR, RSS, TSC, and risk significant local control stations**
  - **Task Analysis considers:**
    - **Select sample of representative and important tasks for operations, maintenance, test, inspection and surveillance**
    - **Tasks that support critical safety functions (abnormal, emergency, transient low-power and shutdown conditions)**
    - **Risk-significant human actions based on Chapter 19 PRA**
    - **Unique tasks that support system designs not used in existing operating PWRs**
  - **Human System Interface evaluations are performed on virtual and physical mock-ups in combination with part task simulation**
  - **Integrated System Validation, as a part of Verification & Validation, is conducted on a Full Scope Simulator using EPR plant scenarios**

Acronyms and Abbreviations

Acronym	Definition
ACCU	Accumulator
BTP	Branch Technical Position
CFR	Code of Federal Regulations
CL	Cold Leg
COL	Combined License
CCW	Component Cooling Water
DAC	Design Acceptance Criteria
DAS	Diverse Actuation System
DCS	Distributed Control System
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFWS	Emergency Feedwater System
EOP	Emergency Operating Procedure
ESF	Engineered Safety Features
FSAR	Final Safety Analysis Report
FT	Feet
GDC	General Design Criteria
GL	Generic Letter
GSI	Generic Safety Issue
GTS	Generic Technical Specifications
HELB	High Energy Line Break
HFE	Human Factors Engineering
HL	Hot Leg
HMI	Human Machine Interface

Acronyms and Abbreviations (Cont'd.)

Acronym	Definition
I&C	Instrumentation and Controls
IEEE	Institute of Electrical and Electronics Engineers
IRWST	In-Containment Refueling Water Storage Tank
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LHSI	Low Head Safety Injection
MCR	Main Control Room
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MHSI	Medium Head Safety Injection
MSIV	Main Steam Isolation Valve
MSRT	Main Steam Relief Train
NI	Nuclear Island
NRC	Nuclear Regulatory Commission
NS	Non-Safety
NS-AQ	Non-Safety, Supplemented Grade
NSSS	Nuclear Steam Supply System
PE	Programmable Electronics
PLD	Programmable Logic Device
PACS	Priority Actuation and Control System
PRA	Probabilistic Risk Assessment
PS	Protection System
PSIG	Pounds Per Square Inch, Gage
PWR	Pressurized Water Reactor
QDS	Qualified Display System
RAI	Request for Additional Information

Acronyms and Abbreviations (Cont'd.)

Acronym	Definition
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RSS	Remote Shutdown Station
RTNSS	Regulatory Treatment of Non-Safety Systems
S	Safety
SAHRS	Severe Accident Heat Removal System
SAS	Safety Automation System
SBO	Station Blackout
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIS	Safety Injection System
SPND	Self-Powered Neutron Detector
SRP	Standard Review Plan
SSC	Structures, Systems and Components
STS	Standard Technical Specifications
TI	Turbine Island
TSC	Technical Support Center
TXS	Teleperm XS

Source Terms for Small Modular Reactors

ACRS Meeting

March 8, 2012

NUCLEAR
ENERGY
INSTITUTE

Overview of Industry Objectives

- Establish a sound technical basis for evaluation of radionuclide inventories in SMRs and their postulated release and transport mechanisms
- Focus on integral pressurized water reactors (iPWRs)
- Identify ST related design and operations attributes that are different than that of large LWRs
- Propose potential regulatory positions in light of existing regulatory requirements
- Identify areas for focused research
- Deliverable: NEI Position paper

NEI

Guiding Principles

- Rely on established or prior work to the extent feasible
- Focus on integrated pressurized water reactors (iPWRs)
  - Maintain consideration of potential development of other SMR technologies
- Identify iPWR attributes and commonalities in design or concerns related to source terms and, where practical, propose common regulatory positions

Overview of Approach

- Broad assessment of source term identification and assessment
  - Plant designs and operations
  - Accidents and beyond design basis events
- Review of regulatory requirements and guidance to identify:
  - Commonalities with current approaches for large LWRs
  - Differences with current approaches for large LWRs
- Evaluate potential regulatory applications
- Identify areas where research may be beneficial
  - Define specific research topics when appropriate

NEI

iPWR Commonalities

- Design
  - Extension of proven technologies
  - Use of modified PWR fuel assemblies
  - Increased use of passive features
  - Multi-module considerations
- Operations
  - Control room
  - Surveillance and maintenance
  - Fuel handling and storage

Highlighted iPWR Attributes

	Generation mPower	NuScale Power	Westinghouse
Containment	Below Grade	Below Grade, Submerged	Below Grade, Submerged
Control Rod Drives	In Vessel	Outside Vessel	In Vessel
Reactor Coolant Pumps	Pumps inside RPV with external motors	No pumps	Seal-less pumps inside RPV with external motors
Integral Steam Generators	Once Through	Helical Coil	Once Through
Refueling Frequency (Relative Fuel EOL Burnup)	4 years	2 years	2 years

NEI

Technical Considerations

- Definition of SMR licensing basis events
- General Areas
  - Magnitude of releases
  - Timing of releases
  - Credit for Passive Design
  - Credit for Other Design Features
- Source Term Treatment
  - Non-mechanistic
  - Mechanistic
  - Mixed or hybrid approach (some of each)

NEI

Technical Considerations

- Specific Technical Considerations
  - Fission/Activation Product Inventory
  - Release Fractions
  - Timing of Release [Key Discriminator for iPWRs]
  - Radionuclide Composition
  - Physical and Chemical Form
  - Release thermal energy
  - Fuel Damage Mechanisms
- Non-Safety Considerations
  - Minimize Consequences / Protect Investment
  - Operator Actions
  - Active Design Features

NEI

Scope of Regulatory Evaluation

- Regulations
  - Part 52 and Part 50 (as applicable)
  - Accident analysis definitions and approaches
- Regulatory Guides
  - RG 1.206
  - RG 1.183, Alternate Source Terms
  - Standard Review Plan
  - NUREG-0800
  - NUREG-1555
  - Interim Staff guidance

Applications of Source Term Modeling

- Plant design
  - Worker safety
  - Component design and operational life
- Normal operations
  - Operational leakage
  - Fuel handling/refueling
  - Maintenance, ISI/IST
  - Component/system repairs and replacements
  - Radioactive waste generation and management
  - Environmental analysis
- Decommissioning

NEI

Applications of Source Term Modeling (continued)

- Accident analysis
  - Design basis – including control room dose
  - Beyond design basis – including site boundary analysis
- Security (Design Basis Threat source term)
- Equipment qualification
- Emergency preparedness

Potential Research Areas

- Potential SMR research topics
  - Beyond design basis computer code development and verification
  - Passive removal mechanisms in small containment and RCS volumes
  - Atmospheric dispersion at close distances (<100 meters)
- Potentially applicable existing research
  - EPRI studies of removal mechanisms for escape of radionuclides through cracks
  - PNL χ/Q study

NEI

Industry Research Strategy

- Potential sponsor organizations
  - DOE
  - National laboratories
  - EPRI
  - Universities
- Identification of possible funding
- Retain focus on near-term technical and regulatory needs

NEI

Path Forward

- Dialogue with NRC
  - Identification of Involved Branches
  - Contact Personnel
- Development of Position Paper
- Pursue identification and scoping of research activities having potential benefit to SMR deployment

NEI

Extremely Low Probability of Rupture (xLPR) Project

David Rudland
Senior Materials Engineer
RES/DE/CIB

Advisory Committee on Reactor Safeguards
March 8, 2012
Rockville, MD

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Welcome

- Purpose of meeting
  - To brief the ACRS on the Extremely Low Probability of Rupture (xLPR) program
- Objective
  - Achieve a common understanding of xLPR status, objectives, priority and planned path forward
  - ACRS review and advice on project
  - Letter from the ACRS Full Committee on the efficacy of the project with respect to the NRC safety goals
- Due to the complexity of this project, we seek ACRS (Subcommittee on Materials) review/advice at least once a year to ensure that we're on the right track.

Outline

- Background and Regulatory Need
- xLPR project plan
- Version 1.0 technical details
- Pilot study goals and results
- Version 2.0 plans and path forward

GDC-4 and LBB

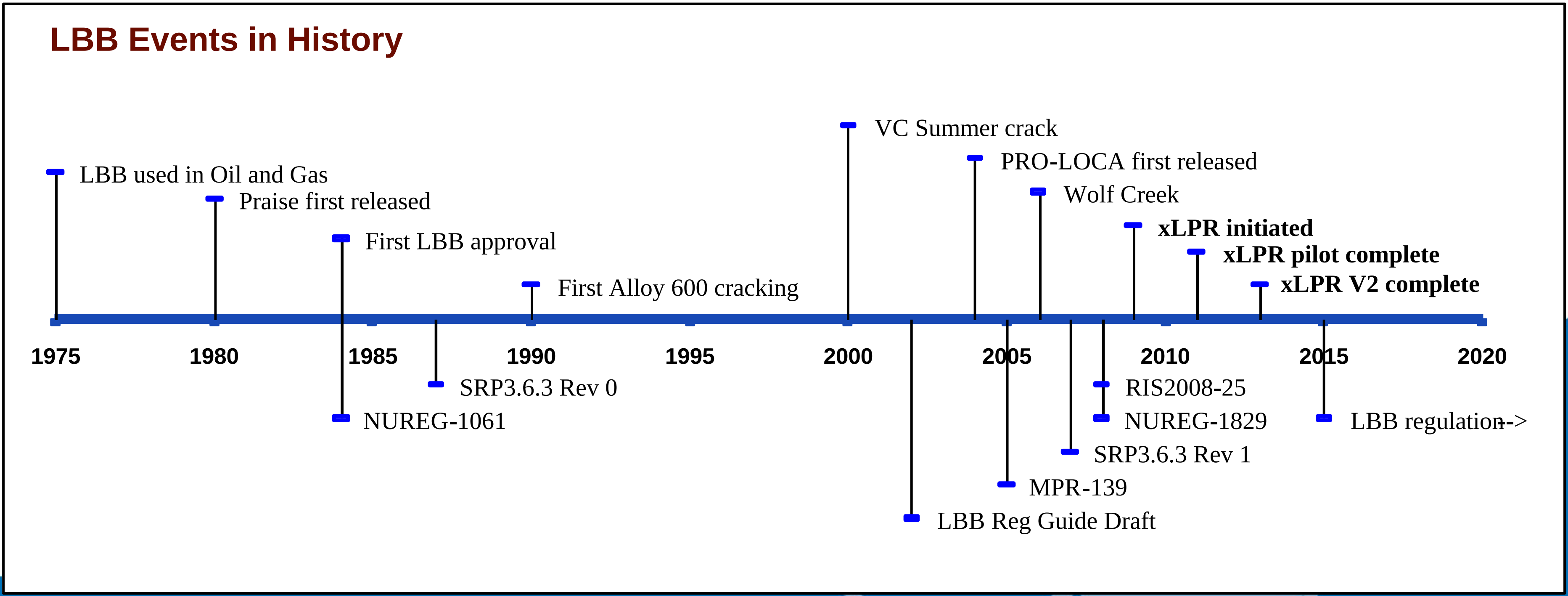
- 10CFR50 Appendix A GDC-4 allows local dynamic effects of pipe ruptures to be excluded from design basis if pipe ruptures have extremely low probability of occurrence

- Local dynamic effects include pipe whipping and discharging fluids. Effect is to eliminate need for whip restraints and jet impingement shields

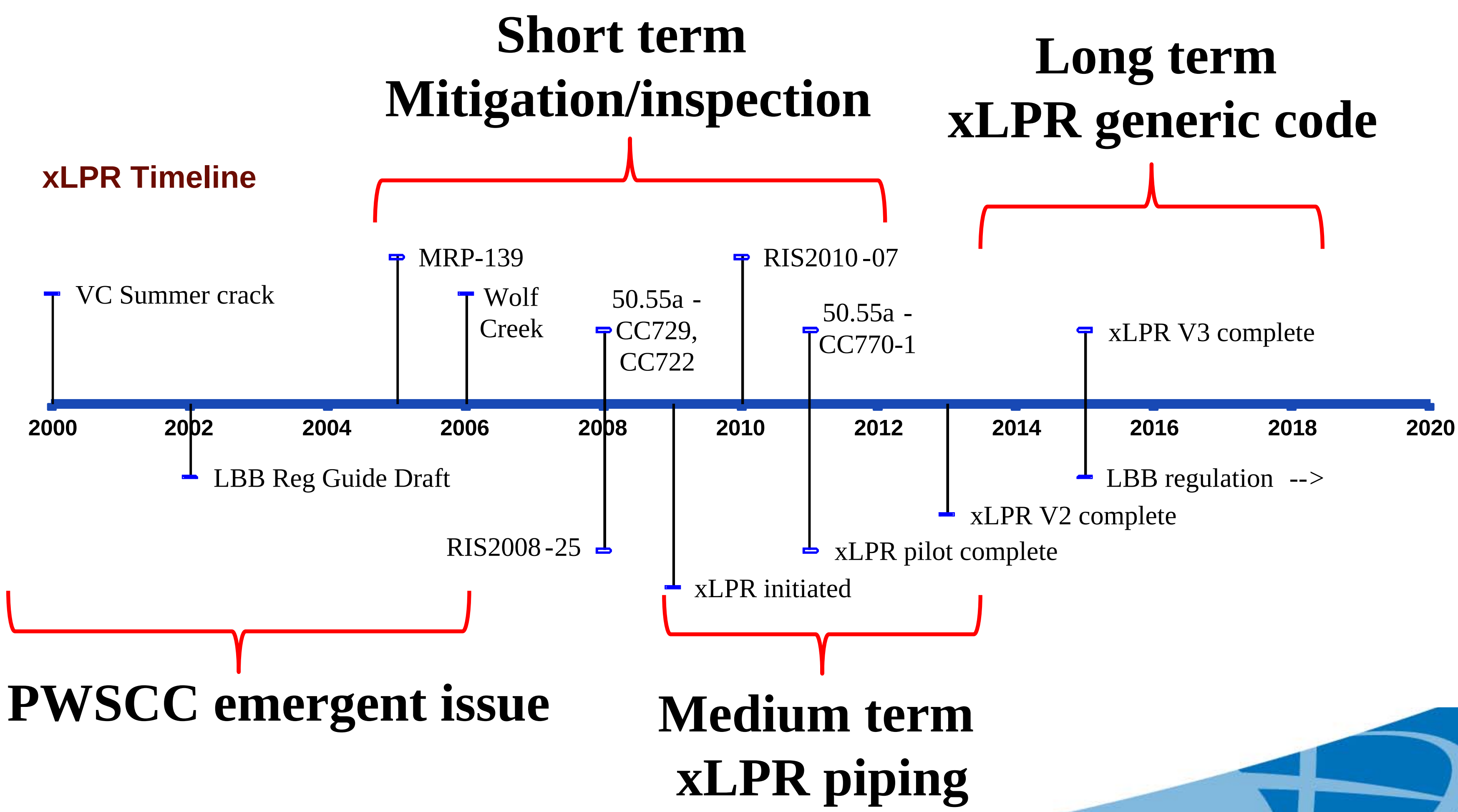
- Conservative flaw tolerance analyses developed and incorporated in SRP3.6.3 to demonstrate leak-before-break (LBB) and satisfy GDC-4

LBB Historical Review

- PWRs have LBB approvals for reactor coolant loop (RCL) piping
  - Some PWRs have LBB for RCL branch piping
- SRP 3.6.3 stipulates no active degradation. PWSCC is active in LBB approved lines
- Qualitative: mitigations and inspections – **Short Term**
- Quantitative: probabilistic evaluation – **Long Term**



xLPR Timeline



Longer Term

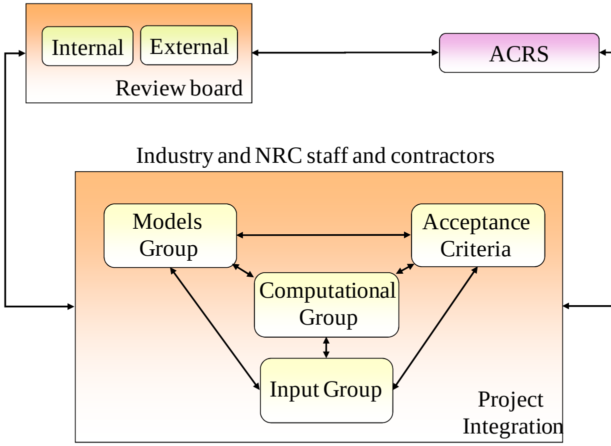
- Develop a ***probabilistic*** assessment tool that can be used to ***directly*** assess compliance with 10CFR50App-A GDC-4

- Tool will be
  - **Comprehensive** with respect to known challenges and loadings
  - **Vetted** with respect to scientific adequacy of models and inputs
  - **Flexible** to permit analysis of a variety of in service situations
  - **Adaptable** – able to accommodate
    - evolving / improving knowledge
    - new damage mechanisms

xLPR Development

- NRC goal to develop “modular” code for evaluating the risk of pressure boundary integrity failure

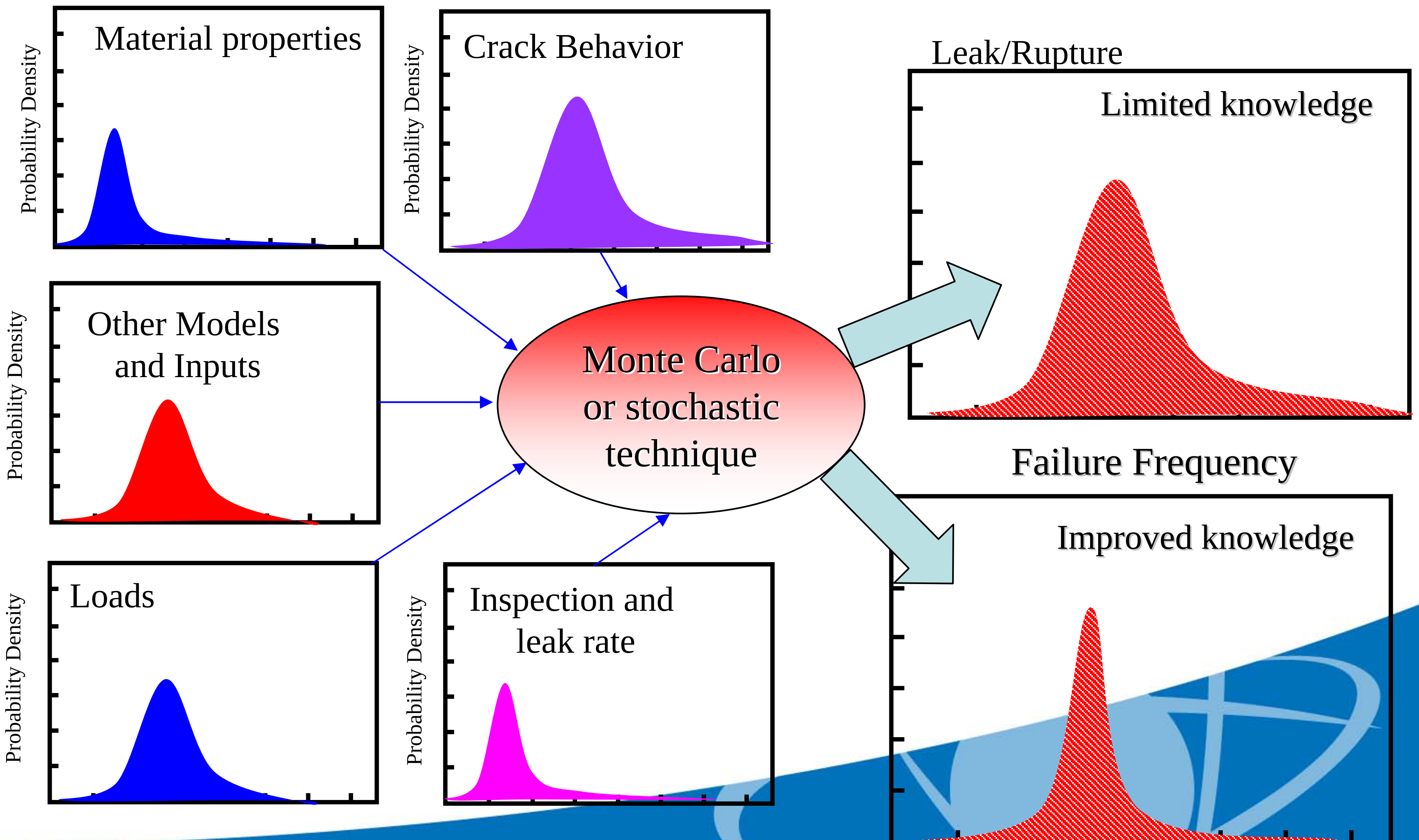
- Currently focusing on piping issues
  - LBB
  - <u>May be applicable to other needs</u>



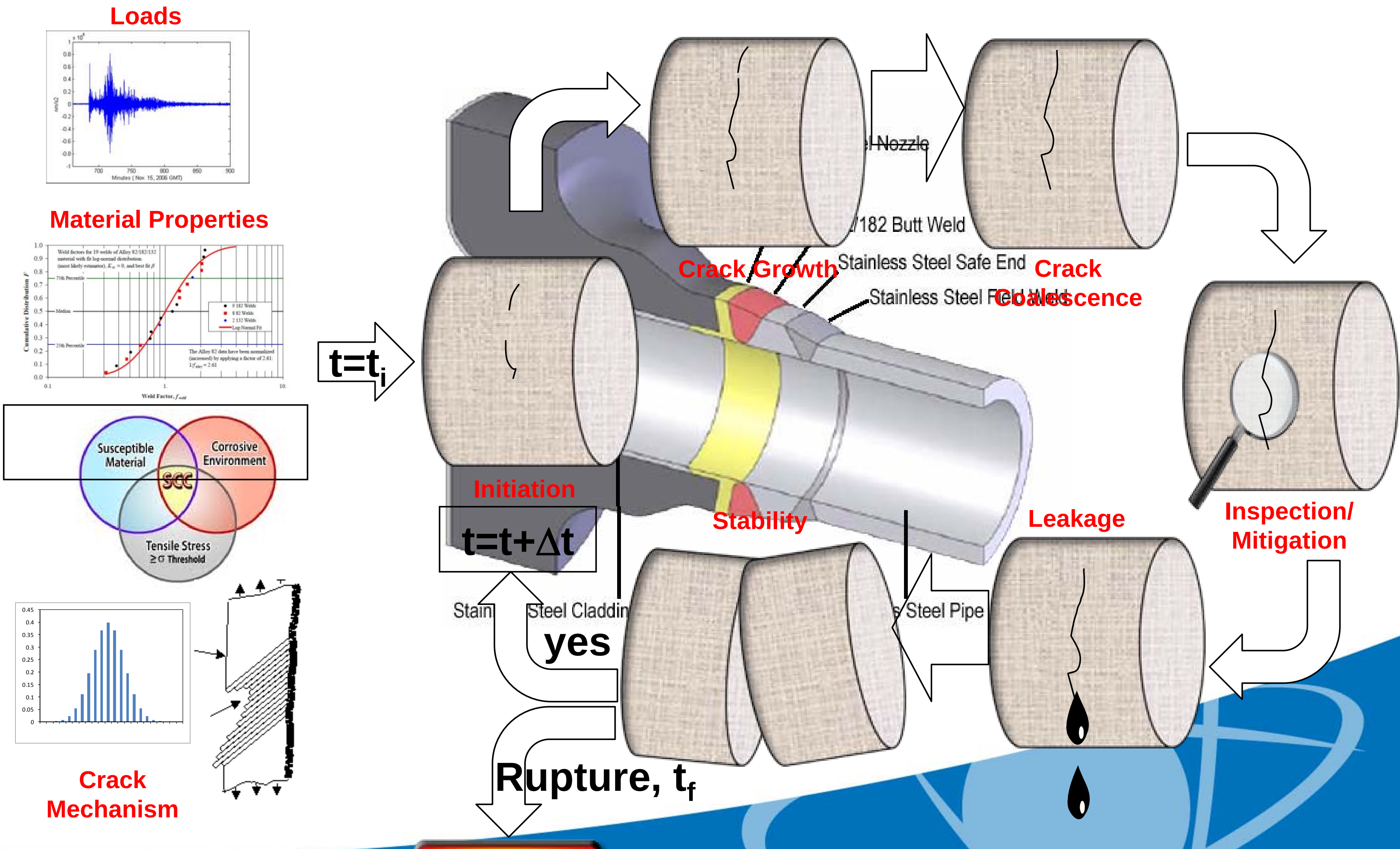
- Working cooperatively with EPRI through MOU addendum

- Initial pilot study to assess effectiveness of approach

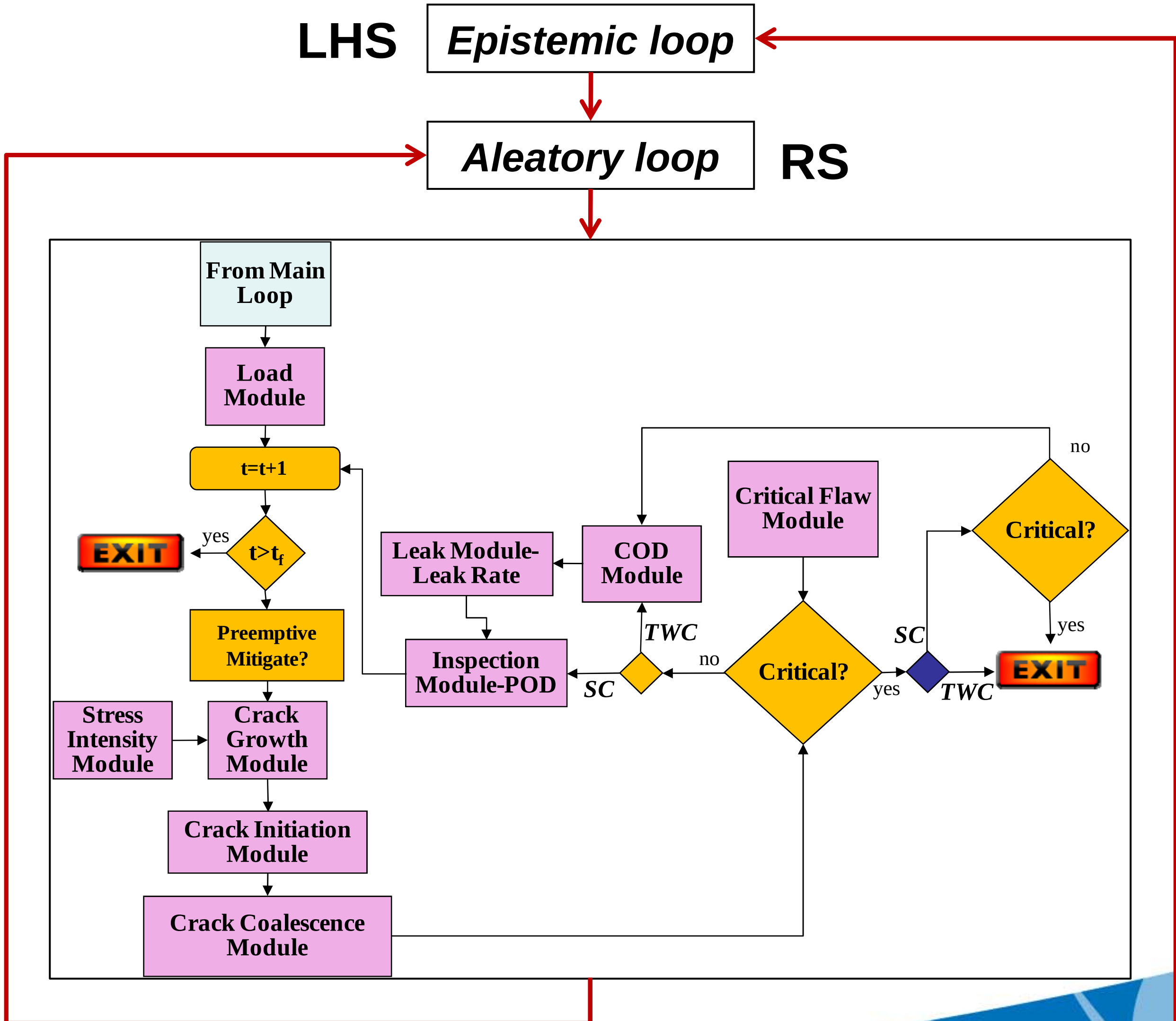
xLPR Process



xLPR Technical Flow



xLPR Process



Purple boxes represent self-contained, independent modules

Epistemic – Lack of Knowledge uncertainty

Aleatory – Irreducible uncertainty

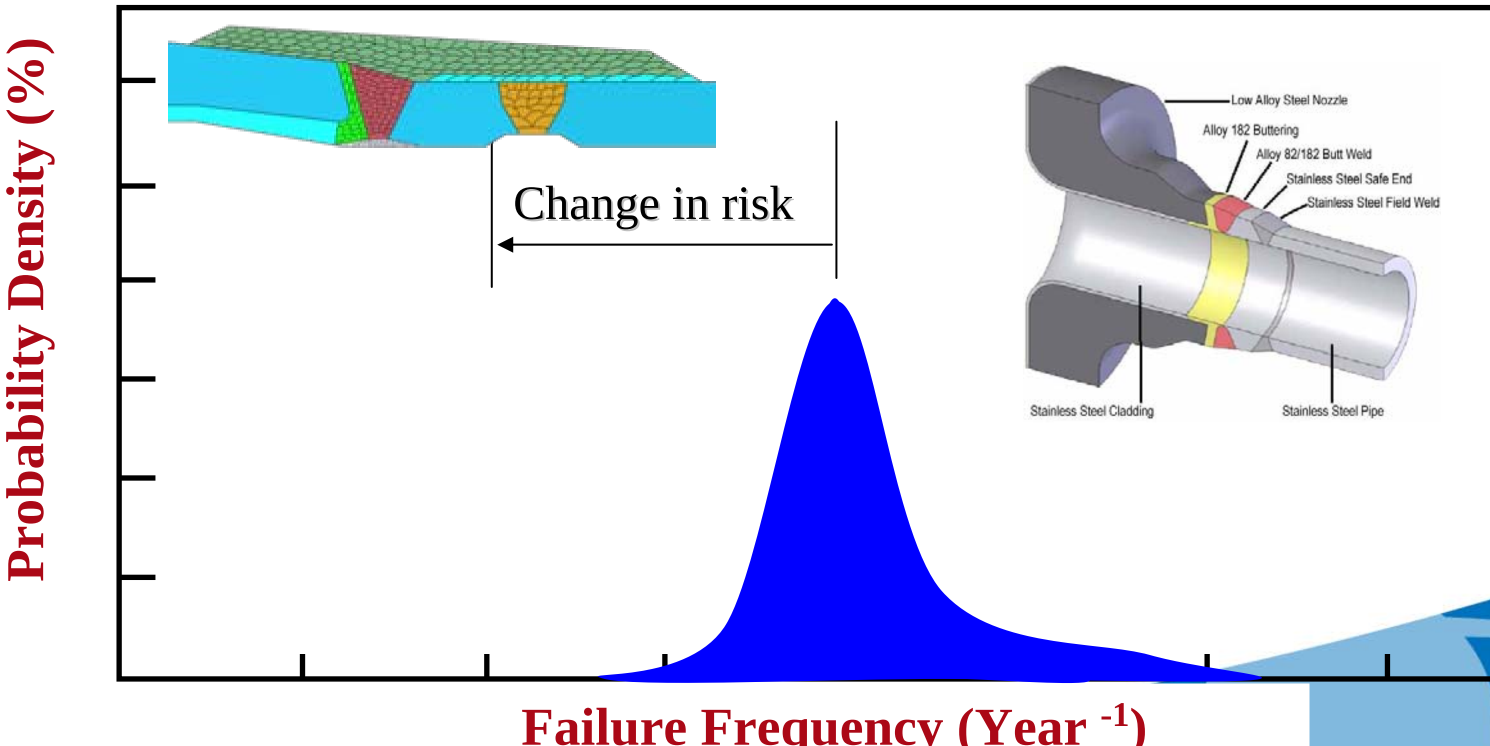
Probability of leak/rupture

Importance sampling was demonstrated

Using xLPR

Conduct analyses with typical parameters

Conduct analyses with typical parameters and overlay



Team Members

Computational Group

David Rudland – U.S. NRC
Bruce Bishop – Westinghouse
Nathan Palm – Westinghouse
Patrick Mattie – Sandia National Laboratories
Cedric Sallaberry – Sandia National Laboratories
Don Kalinich – Sandia National Laboratories
Jon Helton – Sandia National Laboratories
Hilda Klasky – Oak Ridge National Laboratory
Paul Williams – Oak Ridge National Laboratory
Robert Kurth – Emc^2
Scott Sanborn – Pacific Northwest National Laboratory
David Harris – Structural Integrity Associates
Dilip Dedhia – Structural Integrity Associates
Anitha Gubbi – Structural Integrity Associates

Inputs Group

Eric Focht – U.S. NRC
Mark Kirk – U.S. NRC
Guy DeBoo – Exelon
Paul Scott – Battelle
Ashok Nana – AREVA NP Inc.
John Broussard – Dominion Engineering
Nathan Palm – Westinghouse
Pat Heasler – Pacific Northwest National Laboratory
Gery Wilkowski – Emc^2

Acceptance Group

Mark Kirk – U.S. NRC
Glenn White – Dominion Engineering Inc.
Aladar Csontos – U.S. NRC
Robert Hardies – U.S. NRC
David Rudland – U.S. NRC
Bruce Bishop – Westinghouse
Robert Tregoning – U.S. NRC

Models Group

Marjorie Erickson – PEAI
Gary Stevens – U.S. NRC
Howard Rathbun – U.S. NRC
David Rudland – U.S. NRC
John Broussard – Dominion Engineering
Glenn White – Dominion Engineering
Do-Jun Shim – Emc^2
Gery Wilkowski – Emc^2
Bud Brust – Emc^2
Cliff Lange – Structural Integrity Associates
Dave Harris – Structural Integrity Associates
Steve Fyfitch – AREVA NP Inc.
Ashok Nana – AREVA NP Inc.
Rick Olson – Battelle
Darrell Paul – Battelle
Lee Fredette – Battelle
Craig Harrington – EPRI
Gabriel Ilevbare – EPRI
Frank Ammirato – EPRI
Patrick Heasler – Pacific Northwest National Laboratory
Bruce Bishop – Westinghouse

Program Integration Board

Craig Harrington – EPRI
Aladar Csontos – U.S. NRC
Robert Hardies – U.S. NRC
Denny Weakland - Ironwood Consulting
David Rudland – U.S. NRC
Bruce Bishop – Westinghouse
Eric Focht – U.S. NRC
Guy DeBoo – Exelon
Marjorie Erickson – PEAI
Gary Stevens – U.S. NRC
Howard Rathbun – U.S. NRC
Mark Kirk – U.S. NRC
Glenn White – Dominion Engineering Inc.

xLPR Pilot Study

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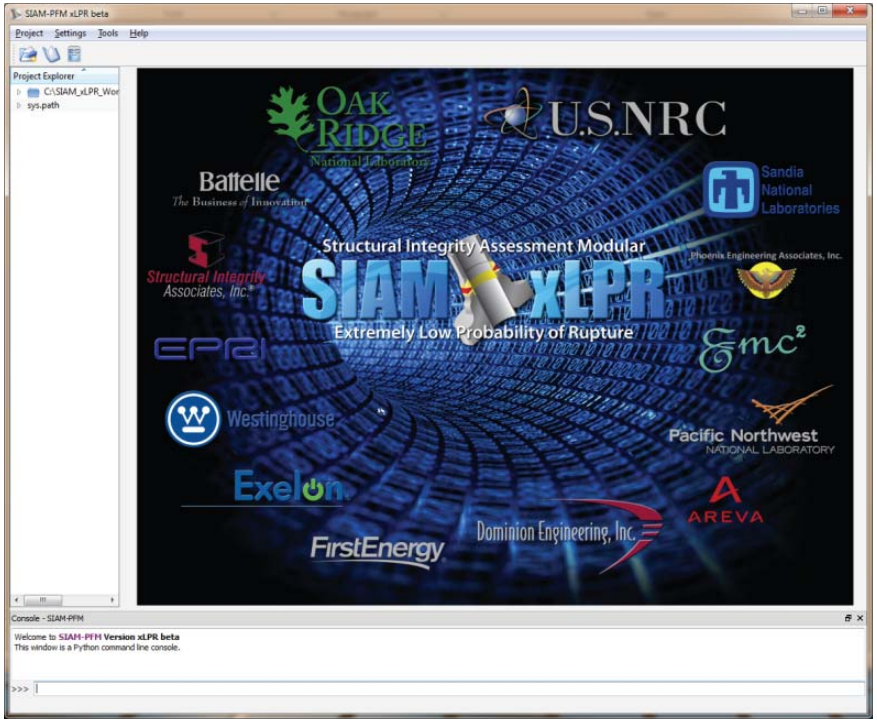
- Pilot study objectives
  - Develop and assess xLPR management structure
  - Determine the appropriate probabilistic framework
  - Assess the feasibility of developing a modular-based probabilistic fracture mechanics computer code

- Focused on pressurizer surge nozzle DM weld with PWSCC

- Development of Version 1.0 code using comprehensive configuration management

- Developed detailed program plan (objective, schedule, deliverables, budget, communications) for Version 1.0 and Version 2.0 code

xLPR Version 1.0 Framework



Fully Open Source

GoldSim Commercial Code

Two framework structures considered
Same calculation modules used
Both gave similar results

Uncertainty

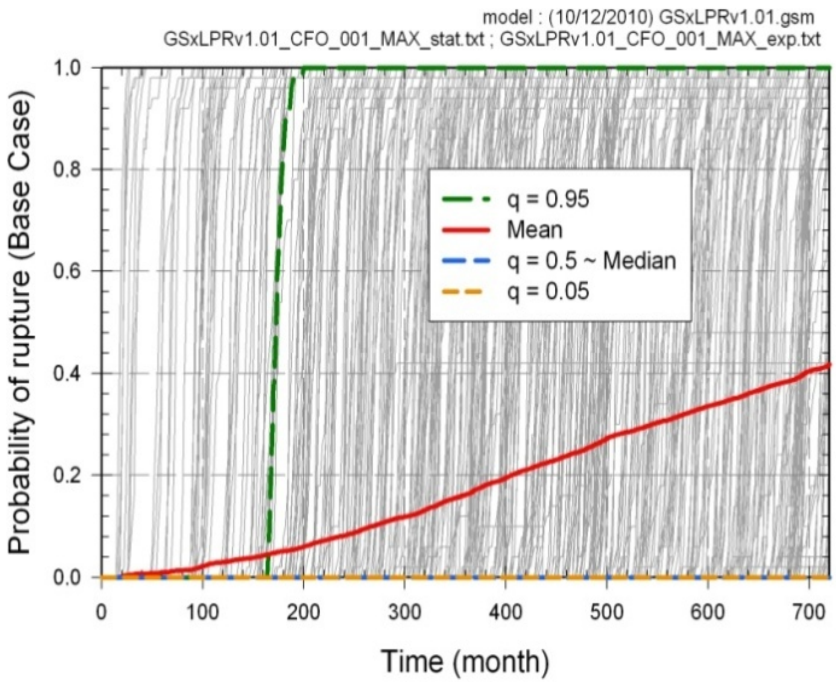
- Uncertainties were classified by models/inputs group
- More discussion needed, but satisfactory for pilot study

Epistemic (Lack of knowledge)	Aleatory (Irreducible)
• Loads • WRS • Crack growth (fweld) • Crack initiation parameters • POD parameters	• Crack size • POD detection • Material properties • Crack growth parameters (Q/R,c,P)

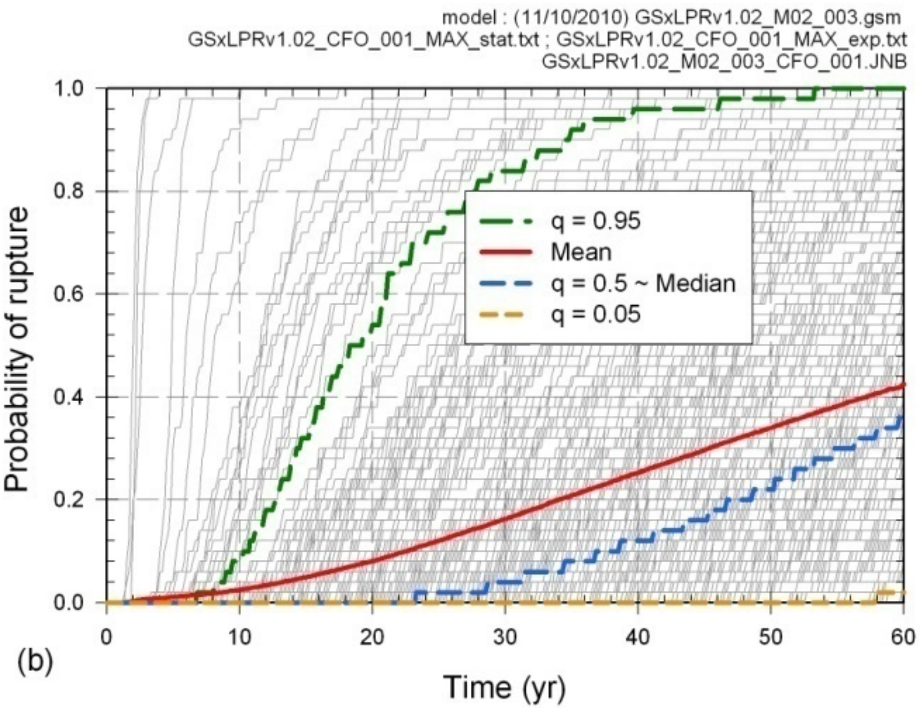
- Currently uses LHS (epistemic) and MC (aleatory)
- Discrete probability distributions also available.
- Importance sampling was demonstrated

Base Case Results

No leak detection, no inspection, high WRS



Crack Initiation categorized as epistemic by models group



Crack Initiation categorized as aleatory

Problem is driven by crack initiation!!

Grey lines represent individual epistemic realizations

xLPR Code Feasibility

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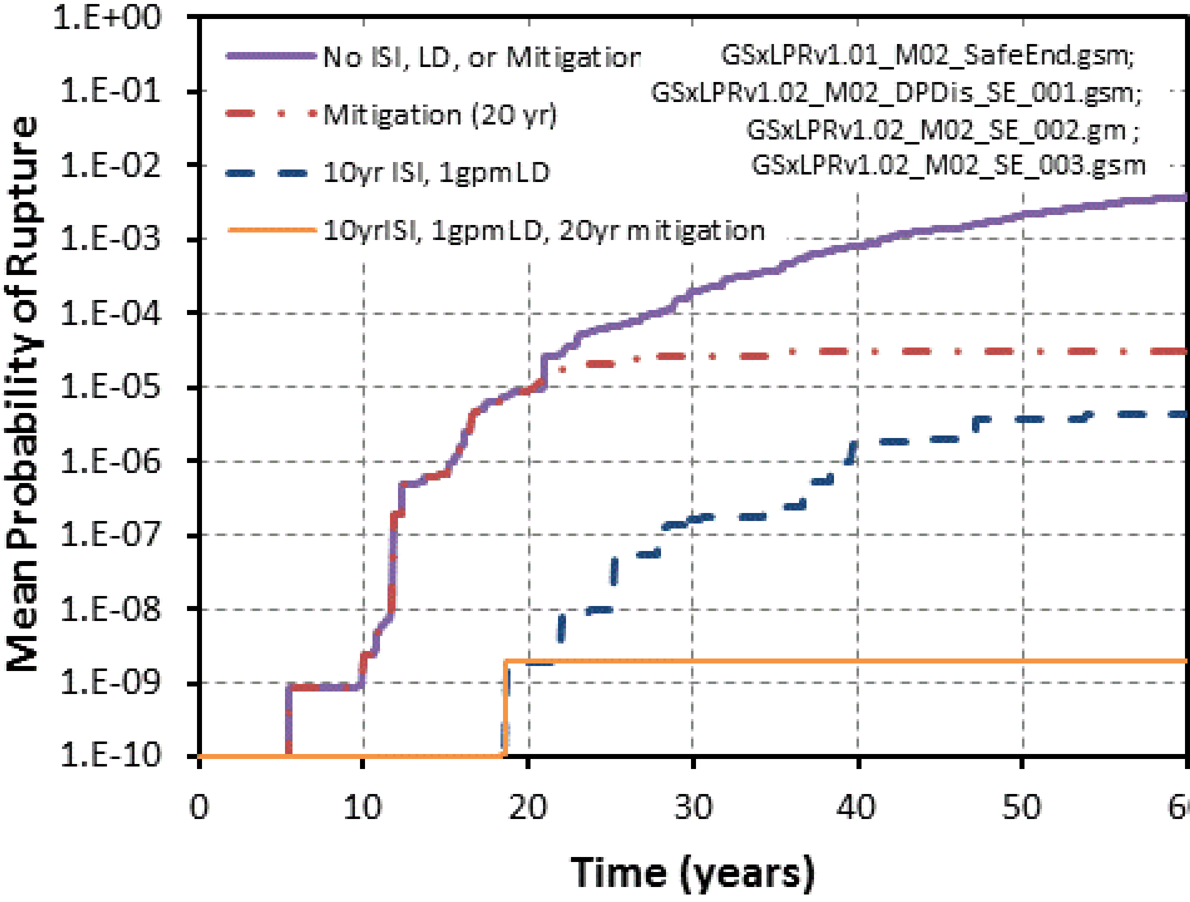
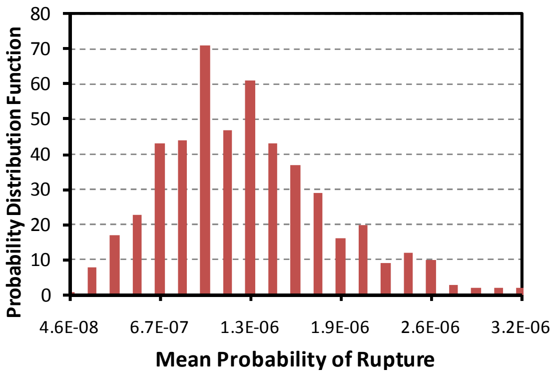
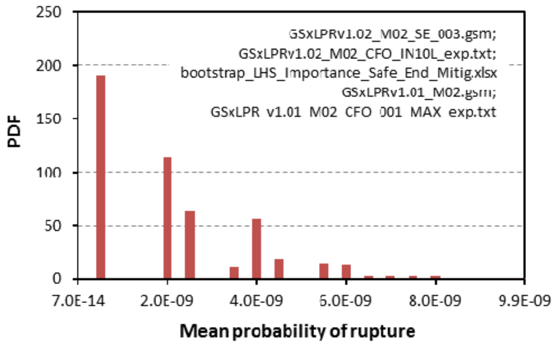
Westinghouse-type pressurizer surge nozzle dissimilar metal weld







Pilot Study Results

- The project team demonstrated that **<u>it is feasible</u>** to develop a modular-based probabilistic fracture mechanics code within a cooperative agreement while properly accounting for the problem uncertainties

- Identified potential efficiency gains in the program management structure

- Selected commercial software as the computational framework

xLPR Version 2.0 Scope

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- xLPR Version 1.0 was developed as part of a feasibility study and focuses on PWSCC in a Westinghouse-style pressurizer surge nozzle DM weld
- Version 2.0 is being expanded to handle welds within piping systems approved for LBB
- Capabilities of Version 2.0 will meet requirements for LBB lines, but <u>must stay</u> within available cost and schedule limitations
- The lessons learned from the pilot study provided many areas where improvement was needed

Version 2.0 Scope

- Pilot study demonstrated several shortcomings in Version 1.0 scope

- xLPR Groups have developed work plans that selected scope recommendations that fit within available resources and overall xLPR timeframe – Scope decided by majority vote of team leads and PIB

- Model inclusion in xLPR Version 2.0 does not guarantee regulatory approval. Process for obtaining approval of xLPR models is under discussion

Version 2.0 Scope Additions

- Framework
  - Microsoft Access dB for inputs
- Models
  - Environmental fatigue
  - Axial cracks
  - IGSCC
  - Surface crack-to-through wall crack transition
  - Manufacturing defects

Version 2.0 Scope Modifications

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- Framework
  - Investigate advanced methodologies to improve sampling efficiency and solution accuracy
  - Revisit uncertainty propagation methodology
  - Modify code output structure
  - Update post processing
  - Modify GoldSim for additional user capability
- Models
  - Revisit PWSCC initiation – Expert panel
  - Update Weld Residual Stress model – more generic, better uncertainty
    - Weld repairs
  - Update K-solution to be consistent with updated WRS model

Red font represents high priority items

Version 2.0 Scope Modifications

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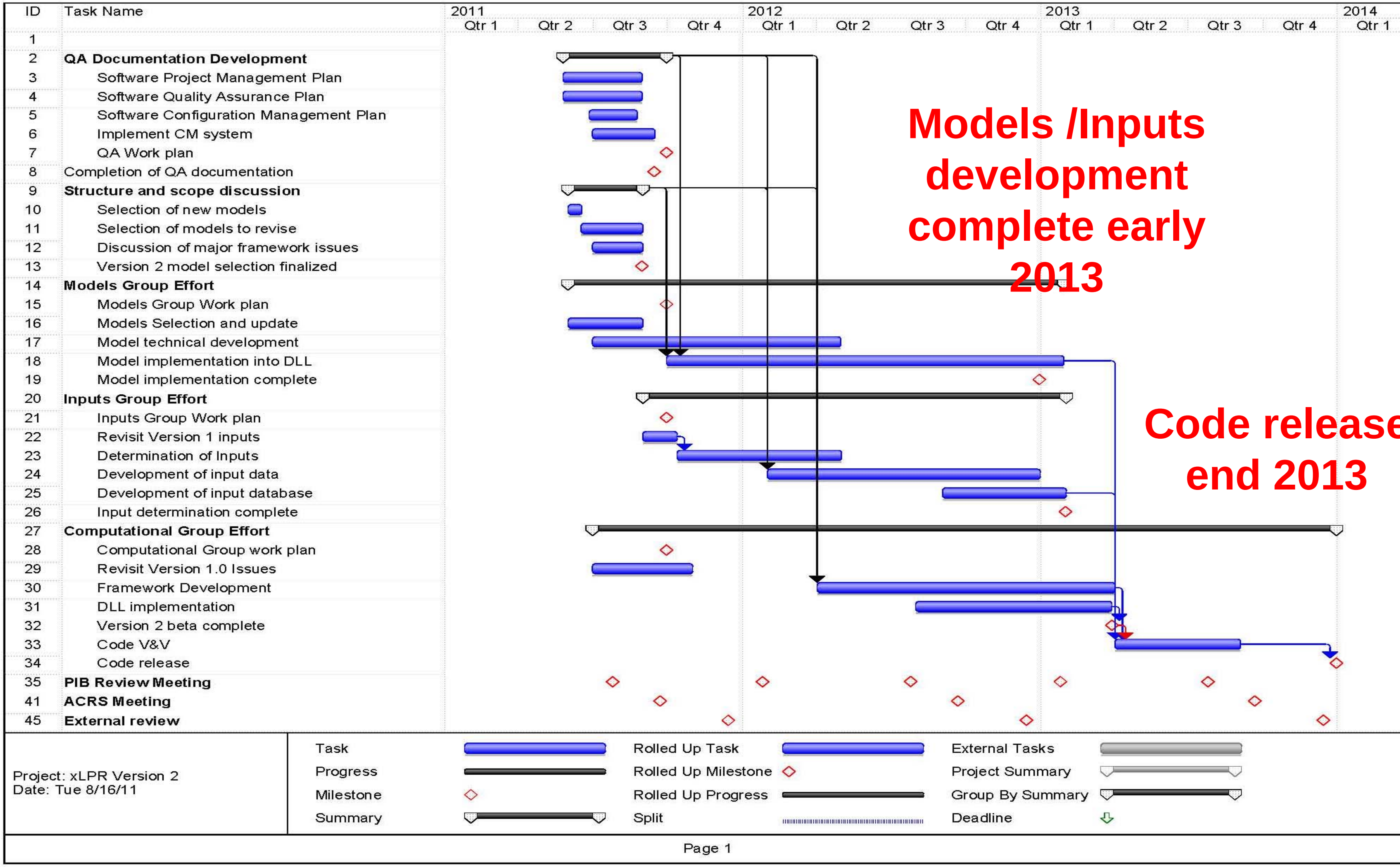
- Models
  - Update mitigation to include FSWOL,OWOL, Inlay, surface treatment, and other chemistry
  - Update Inspection model – sizing, POD, simplified model
  - Update crack stability – Surface crack EPFM
  - Update leak rate model –SQuIRT, bound leak rate calc
  - Update crack-opening displacement - tension and bending blended solution.
- Inputs
  - Update load definition to include transients
  - Retrieve all relevant data for
    - One reactor coolant loop - Westinghouse PWR
    - One reactor coolant loop - Babcock & Wilcox (B&W) PWR
    - Others may be considered

xLPR QA

- Version 1.0 was controlled by a Configuration Management plan but not associated with a detailed QA structure
- Conducted QA workshop with appropriate Regulatory/Industry QA experts
- Consensus agreement that the top level requirements in ASME NQA-1 are sufficient to meet xLPR program, NRC, Industry, and DOE requirements for software development
- QA audits will occur and be aligned with key milestones

Version 2.0 Schedule



Benefits of xLPR

- Quantified solution to LBB issue
  - Regulation guide
  - Update to SRP3.6.3
- Fully QA'ed modular probabilistic fracture mechanics code for reactor pressure boundary integrity
  - LBB including evaluation of mitigation for DM welds
  - Research tool for prioritization
  - TBS – 50.46a
  - Risk informed ISI
  - GSI 191
  - Easily adaptable to other applications
    - CRDM ejection probabilities
    - RPV

Path Forward

- Version 2.0 Development underway
- Ongoing meetings
  - ACRS meeting - March 2012 (yearly updates to subcommittee)
  - NRC and EPRI Management (as needed)
  - External reviews
  - Internal reviews
- Version 2.0 release – End 2013
- Technical basis and Regulatory Guide for LBB - 2015

Turkey Point Units 3 & 4
Extended Power Uprate (EPU)
ACRS Full Committee

March 8, 2012

Agenda

- **Introduction .... Mike Kiley**
- **Discussion Topics from ACRS Full Committee**
  - Loss of Off Site Power Overview............ Steve Hale
  - Shared Systems Overview.................... Steve Hale

FPL

Turkey Point

- **FPL appreciates the opportunity to discuss the EPU License Amendment Request for Turkey Point with the ACRS**
- **Since the ACRS Subcommittee and Full Committee meetings, FPL and NRC Staff worked diligently to close the open items**
- **FPL recognizes and appreciates the importance of the Staff's questions, particularly those on Thermal Conductivity Degradation (TCD)**
  - NRC Staff performed a detailed review of FPL's analysis including a multi-day audit where FPL provided Staff with its analyses
  - This audit and the others performed by NRC Staff have been invaluable to the result of a thorough and comprehensive review of the proposed EPU
- **FPL's top priority is safety. We are proceeding with caution through the remaining steps of the EPU**
- **FPL looks forward to answering any remaining questions**

FPL

Agenda

- Introduction .... Mike Kiley
- **Discussion Topics from Full Committee**
  - Loss of Off Site Power Overview............. Steve Hale
  - Shared Systems Overview..................... Steve Hale

FPL

Although highly unlikely, dual unit loss of offsite power (LOOP) will not result in the pressurizer filling

- **Since 1990, only one dual-unit LOOP**
  - August 1992, as a result of Hurricane Andrew
  - Both units placed in a safe shutdown condition prior to onset of hurricane force winds
- **4 single unit LOOP events, 2 on each unit**
  - Caused by unit internal or switchyard faults/failures
  - Not initiated from the grid
- **Risk associated with a dual-unit LOOP**
  - Dual-unit LOOP frequency = 0.02 per year
  - CDF (dual-unit LOOP) = 8E-08 per year
- **Transient analysis demonstrates that the pressurizer will not fill with a LOOP**
  - Results are comparable to current licensing basis
- **No new Operator actions added for LOOP as a result of EPU**

FPL

Agenda

- **Introduction .... Mike Kiley**

Discussion Topics from Full Committee

  - Loss of Off Site Power Overview......... Steve Hale
  - Shared Systems Overview.................. Steve Hale

FPL

The shared systems provide flexibility and safety benefits

Shared Systems for Turkey Point Units 3 and 4

- **Emergency Core Cooling System**
  - 4 pumps which all start on SI signal from either unit
    - -- 2 required for LOCA, only 1 required for other events
    - -- Each pump aligned to separate emergency diesel generator
- **Auxiliary Feedwater System**
  - 3 turbine driven pumps
    - -- 1 required to satisfy heat removal requirements of both units
    - -- Capable of performing design function without AC power
  - 1 diesel driven standby feedwater pump
- **Emergency Diesel Generators (EDGs)**
  - 4 EDGs, 2 for each unit; all start on SI signal from either unit
  - EDGs can be cross tied from the Control Room
  - Each EDG has sufficient capacity to maintain both units in a safe shutdown condition

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