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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

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U.S. EPR SUBCOMMITTEE

OPEN SESSION

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WEDNESDAY

SEPTEMBER 9, 2009

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, One White Flint North, Commissioner's Conference Room, 11555 Rockville Pike, at 8:30 a.m., Dr. Dana A. Powers, Chairman, presiding.

COMMITTEE MEMBERS:

DANA A. POWERS, Chairman

J. SAM ARMIJO, Member-at-Large

MARIO V. BONACA, Member

OTTO L. MAYNARD, Member

HAROLD B. RAY, Member

WILLIAM J. SHACK, Member

JOHN S. STETKAR, Member

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P R O C E E D I N G S

Time: 8:32 a.m.

CHAIRMAN POWERS: The meeting will now come to order. This is the meeting of the Advisory Committee on Reactor Safeguards.

ACRS members in attendance are Bill Shack, Sam Armijo, John Stetkar, Harold Ray, Otto Maynard. Derek Widmayer of the ACRS staff is the Designated Federal Official for this meeting, and I left out Mario. Mario showed up. The esteemed Chairman of the ACRS himself is here to watch and monitor and assess my importance.

Mike Ryan is here, but Mike -- did we ask you? Good, glad to have you here.

The Subcommittee will hear presentations and hold discussions with representatives of AREVA, NP, the NRC staff and interested persons regarding this matter. This is an information only briefing to the Subcommittee.

The Subcommittee will gather relevant information today and report to the full Committee later on this week, actually Friday, but will not be formulating any findings on these matters at the conclusion of this meeting. In fact, what we are just doing is beginning our process of going through design certification on the EPR.

In that regard, I am going to ask the speakers at the beginning of your presentations to give us a little background, because we are going to be with you for a protracted period of time, and it would be useful to know something about you here.

So if you would just do that at the beginning of your presentation, we usually say why are you qualified to speak before this august body, and just because we are going to be together for several committee meetings, I suspect, and it would be useful to the members to know. The members will not reciprocate, by the way. We have no intention of telling you why we make up such an august body.

The purpose of the meeting is to provide background information on two key technical areas which have been of interest to the staff during the review of the US EPR design certification.

The staff and AREVA both wish to introduce the ACRS Subcommittee members to these technical areas at this early date while the draft safety evaluation report chapters are being completed. The Subcommittee will review these matters again when the relevant chapters of the draft safety evaluation report come to the Subcommittee for formal review.

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

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

Copies of the meeting agenda and handouts are available, actually, in the front of the meeting room.

There is a telephone bridge line that

has been established for the meeting room today, and I understand that we have participants from AREVA and NRC staff on the lines. We do request that participants on the bridge line identify themselves when they speak, and to keep your telephones on Mute during the times when you are just listening and, if you can't figure out how to do that, Mr. Widmayer will be glad to explain the subtle details of *6 to you.

We can begin with the meeting now. I will first of all ask, are there any members of the Subcommittee that want to make opening statements? They are mute on this subject. They have pressed *6 apparently.

Again, it would be useful if speakers would give us a little bit of background when they talk. I will turn now to Mr. Tesfaye who will speak on behalf of the staff.

MR. TESFAYE: Good morning, Mr. Chairman. My name is Getachew Tesfaye. It is pronounced just like it is spelled here, Getachew. I will give you a little bit of background for myself.

I have been with the NRC for five years. Prior to coming to the NRC, I was a licensing engineer at Calvert Cliffs Nuclear Power Plant for 16 years -- Calvert Cliffs. And I was involved in several major project management activities at Calvert Cliffs.

Here at the NRC, I spent my first year doing containment evaluation, and have been the project manager since the application was submitted in December 2007. Is that enough for background, Mr. Chairman?

CHAIRMAN POWERS: That will do.

MR. TESFAYE: Thank you.

CHAIRMAN POWERS: Give us what you want.

MR. TESFAYE: Just a short presentation to give you a status of where we are at with the design certification review. As I said earlier, the application was submitted on December 22, 2007. This is a six-phase review process. Unfortunately, I don't have the slides on the screen, but I have a handout of the slides.

We have completed Phase 1 of the review, which is preliminary safety evaluation report with RAIs, and in that process we generated close to 2500 RAI questions, and that phase was completed on time.

We are currently in Phase 2. The target for completing Phase 2 is June 30, 2010. We have already completed two chapters of Phase 2 and issued SERs with open items. Those are Chapter 2 and Chapter 8. We plan to complete Chapters 11 and 10 within the next few weeks. As I will show you in the next slide, those four chapters will be the first one that will be formally presented to the Subcommittee and the full Committee in November.

Phase 3 is targeted to be completed September; Phase 4, Advanced SER with No Open Items in April of 2011; and then Phase 5 is ACRS review of Advanced SER with no open items, July 2011, and the final SER with no open items is scheduled to be completed in September of 2011, and the rulemaking in February of 2012.

Go to the next slide, please.

Our plan for Phase 3 ACRS review is: We have divided the chapters into four groups, four major groups. The first group, as I mentioned earlier, will be presented in November. Those are Chapters 2, 8, 10 and 12.

I guess, for the sake of people who don't have access to my slides, Chapter 2 is site characteristics. Chapter 8 is electric power. Chapter 10 is steam and power conversion, and Chapter 12, radiation protection.

The second group is a big group. So we have divided it into two subgroups. The first subgroup will be presented in February. That is going to be Chapter 4, Reactor; Chapter 5, Reactor coolant and connected systems; and Chapter 16, Tech Specs, and Chapter 17, quality assurance.

In group 2, we have Chapter 11 and Chapter 19. Chapter 11 is rad waste management. Chapter 19 is severe accidents and PRA.

In Group 3, which is currently tentatively scheduled for May 2010, we have Chapter 3, design of structures, components and equipment; and Chapter 7, instrument and control systems; Chapter 9, auxiliary systems, and Chapter 18, human factors.

The final group will be presented in July 2010, and that will be Chapter, general plant description; Chapter 6, engineered safety features; Chapter 13, conduct of operations; Chapter 14, initial test programs, and Chapter 15, safety analysis.

The last presentation to ACRS in this

Phase 3 will be in September. This will be a summary, and again this is tentative. If there is no need, we may not have that meeting in September. What we plan to do at that summary is give you the status of all open items, any cost cutting issues, and revisit earlier chapters as needed.

That is our plan for Phase 3.

MEMBER SHACK: The dates in parentheses, those are dates you completed the draft?

MR. TESFAYE: Yes, open items. No, no, no. The dates in parentheses in the tables?

MEMBER SHACK: Oh, yes. You've got a color. I've got a black and white. Okay.

MR. TESFAYE: You are right. The one next to the chapters?

MEMBER SHACK: Yes.

MR. TESFAYE: Yes. That is all I have. Is there any questions for me?

CHAIRMAN POWERS: I don't know that we have any questions.

MR. TESFAYE: Thank you.

CHAIRMAN POWERS: I appreciate the schedule, though. I don't guaranty that we will follow it, but it gets us started on this process. Thank you.

MS. SLOAN: All right. Thanks.

By way of introduction, my name is Sandra Sloan with AREVA. My current responsibilities are: I am the Manager of Regulatory Affairs for New Plants. What that means in practical terms is I am responsible for providing licensing support for all US EPR projects. Obviously, the focus of my group is US EPR design certification, but we also provide support to our combined license applicants as well.

By way of background, I started my career at the Idaho National Lab, spent six years there doing thermal hydraulics and safety analysis, and then went on to AREVA and its predecessor companies where I have been for the last 12 years, and transitioned to licensing related work about six years ago. And as I get told frequently, I am not very technical anymore as a licensing person, but I do like to think that I remember something from my thermal hydraulics and safety analysis background.

CHAIRMAN POWERS: We'll try to get rid

of that. How many COL applicants do you think you will have?

MS. SLOAN: You mean ever or do we have right now?

CHAIRMAN POWERS: As you see it, where is the world right now?

MS. SLOAN: Well, right now the active applicants, obviously, are Calvert Cliffs for Unistar, Bell Bend for PPO. Nine Mile Point has been submitted and accepted for review, but the start of the review has been sequenced or deferred, whichever word you choose to use, to September of next year.

Of course, Callaway was submitted and accepted for review, and has asked the staff for now to suspend the review, and the staff has agreed to suspend the Callaway review for the time being.

CHAIRMAN POWERS: Keeps you busy, presumably.

MS. SLOAN: Well, very busy, yes.

So our goal today, based on our interactions with the support staff, primarily with Derek, and in talking with the NRC staff -- Our goal was, as you said, to provide you some background information.

We understand this is the beginning of what we hope is a long and very successful relationship with the ACRS Subcommittee.

CHAIRMAN POWERS: We hope it is brief and successful.

MS. SLOAN: Briefer is better.

CHAIRMAN POWERS: We anticipate it will be warm, but we don't want to prolong it.

MS. SLOAN: Good. So in that vein, we decided we wanted to give you some background information. We realize this is not the end-all, and there will be other discussions on these topics, but what we had hoped to do was at least give you an overview in two key topic areas that have been of particular interest in the review, one of them being containment design and analysis, which we have had quite an extensive series of interactions with the NRC staff. So Marty Parece, who is our Vice President of Technology, will be talking about that.

Then also, based on some expressed interest, we decided to talk just a little bit in the afternoon about our safety analysis methodologies. Our objective, again, is to give you some background information that, hopefully, will be helpful to you as you get the SER with open items for the chapters from the staff.

So the way this is divided up, as shown on the agenda, what we planned is the morning session would focus on containment design and analysis. There will be an open session for the public, and then there will be a closed session where we will go into more details about the evaluation model that we are using for the mass and energy releases as well as the containment pressure and temperature response

Then after lunch we will come back and talk safety analysis methodologies, and that will be again formatted with an open session where we will give you a broad overview of the design and particular design features that are important, particularly important or unique, when it comes to safety analysis for EPR, and give you some insights on why we selected the methodologies we did, and at least an overview of how we demonstrated applicability of the methods for EPR.

Then in a closed session, we elected to focus on three key methodologies, and we put a lot of thought into which methodologies we would use this time. We realize the time is somewhat limited, but we used the criteria of did a methodology, as in the case of control rod ejection, need to be updated or changed on the basis of new regulatory expectations, in this case the new SRP acceptance criteria and guidance.

So there was a methodology developed specifically for EPR for control rod ejection, which reflects the new SRP acceptance criteria. Then we will spend some time talking about large break LOCA. We use a realistic large break LOCA methodology for EPR, and we submitted a topical report to the staff. It is a specific application of the realistic LOCA methodology for EPR.

So we would like to talk some about that, and then spend some time at the end talking about small break LOCA, simply because, as you will hear in the design overview, there are a couple of design features which, at least initially -- in particular, the partial cooldown of the generators and the use of medium head safety injection caused some slightly different response in the early phases of a small break LOCA.

So that's the topics that we picked and

why we picked it. I hope that helps. I did want to call attention to the fact that in your slide package, behind my slides there is a list of acronyms. We tend to talk in alphabet soup. So often we find it is helpful to have a decoder ring. So I would encourage you, if we use something and it doesn't make sense to you -- the acronym doesn't make sense -- obviously, stop us. But this is sort of the decoder ring. Hopefully, we covered all the acronyms that we will use today.

CHAIRMAN POWERS: A committee that goes by the name ACRS is not unfamiliar with abbreviations.

MS. SLOAN: Good. Unless there are any questions, generally speaking, about what we have done in design certification, I will turn it over to Marty Parece.

CHAIRMAN POWERS: I would just inject that we will go into closed sessions a couple of times today, and the pressure is on you to see that everybody in the room is qualified to be here. Derek will take care of the mechanics, but you've got to vet the people.

MS. SLOAN: Okay, got it.

CHAIRMAN POWERS: And you would probably kick me out on that basis, and that would be good. Marty?

MR. PARECE: Okay. As you have heard, my name is Marty Parece. I am the Vice President of Technology. In that position, my organization is responsible for the configuration control of all new reactor products in North America, including EPR and work we are doing on our gas reactor product with BEA and the DOE.

In that vein, many of the design features we are going to talk today is part of our goal of keeping converged with the worldwide design. We would like to standardize as much as possible with the European fleet.

My background: I started with a precursor company of AREVA, Babcock-Wilcox, in Lynchburg, Virginia, 27 years ago, and my background started in safety analysis and plant analysis, including all types of PWRs, and I have had a very broad background.

So I tend to be more of a generalist these days than a specialist in any one thing, but I was the architect of our power up rate and steam generator replacement licensing technologies and approaches, and I have worked on emergency operating procedures, component design, large licensing projects, small projects, and I have always kept my toe in the technology, the R&D, the codes and methods that we are applying.

That is one of my responsibilities now as Vice President of Technology. I am responsible for coordinating our R&D programs.

So that is my background. is that sufficient?

CHAIRMAN POWERS: It is a start, Marty.

MR. PARECE: Okay. Next slide, please. So in this open session, we are going to talk about the containment design features and the layout of the containment, so that you get a good feeling for what the containment looks like and where things are and how it works during a postulated event, and we are going to do a summary of the evaluation model.

Then in the closed session, we will go into more details on the evaluation model.

CHAIRMAN POWERS: One of the great philosophical issues ACRS now wrestles with is that this isn't the first EPR in the world, and presumably other regulatory bodies have examined this containment in some detail. Particularly, I know the Finns looked at it in some detail.

One of the great philosophical issues that the Commission certainly wants to wrestle with -- I don't know whether the ACRS does or not; I look at Mr. Bonaca on this -- is how much of that do we just take on faith? I mean, they looked at it. Why are we looking at it in detail?

As it comes up, it would be useful to know where things are going to stand, and in great detail, and what your feelings are on the need to pursue some of these things in depth. It is an issue that we've got to wrestle with.

We understand that the nuclear business is becoming very international in character, and you know, there's been some things that an American plant, or a European plant or a Japanese plant -- they are all kind of like everybody in the world's plant. So how much duplication of effort do we have to go through on these things, and how much can we say, okay, well, you know --

For instance, Sloan, in your area of thermal hydraulics, I bet you the French go through thermal hydraulics, that they are fairly detailed, and I know for sure the Finns looked at containment a little bit.

MR. COLACCINO: Dr. Powers, if I could interject here. I would just like to introduce myself. My name is Joe Colaccino.

I am the Chief of the EPR Projects Branch. Just to follow through with what you asked for, my responsibilities are to manage the branch that is doing the design certification, the four COLAs that we have, Calvert, Bell Bend, and the two suspended ones that we were mentioned, Nine Mile and Callaway.

I just wanted to bring up the activities that the staff is involved with in the Multi-national Design Evaluation Program or MDEP.

EPR is a particularly active as an active sub-working group. We are meeting with regulators from Finland, France, the UK, Canada, and just recently China, biannually, every six months, to discuss where the regulators are in their reviews. We are exchanging information.

There are three technical expert subgroups that are within that expert group. One of them is on accent analysis, which does include containment. One is digital INC which the NRC is the lead for. One of the NRO branch chiefs is

Terry Jackson who, I believe, has appeared before the Committee before. The third group is the -- I believe it is the PRA Subgroup.

So those technical expert subgroups meet separately with the exception of the containment one. The containment one is really meeting in conjunction with a main working group.

So there is significant discussion that is going on between the regulatory bodies of al the nations that are either in the process of deploying EPRs or thinking of deploying EPRs.

So there is quite a bit of discussion that is going on, and so I understand your comments. It is completely understandable. I just want you to know that the staff is working with the other regulatory bodies that are working on doing EPR licensing.

I would just make one observation. We are kind of all in an interesting time -- I think I would say "all" -- the Finns, the French and the U.S., because even though we are in different stages of our licensing processes, we are actually all converging at about the same time. We are making decisions.

It is kind of interesting. Our Part 52

licensing process, driving decisions earlier in the decision making, is kind of gelling with where the Finns and the French are. They are in the middle of construction right now, but they haven't received the operating license applications.

So they are looking at things, and we are looking. It is actually kind of an opportune time. So we are trying to take as much advantage as we can out of these interactions.

CHAIRMAN POWERS: I appreciate your comments. In fact, it would probably be useful to have a presentation from you at sometime on exactly the activities that you speak of.

MR. COLACCINO: I was anticipating that you would have that, and I would offer that at some point. I would suggest probably sometime next year, if we had our meeting in December.

The NRC staff has initiated several calls on other topics, because as Getachew said, we are producing -- I want to emphasize what we are producing right now. It is a safety evaluation report with open items. It is not a draft safety evaluation report.

In fact, one of the safety evaluations that we have issued, Chapter 8, has no open items in it. I consider that final. You know, unless someone makes a change or, in the context of doing a review, that we find something else, I don't expect to look at it again.

CHAIRMAN POWERS: It is really quite interesting, and maybe you guys can figure out some time and we can hear about what all they are doing. That is useful.

MR. COLACCINO: Okay. I would expect in the spring of next year is probably a more appropriate time to come with that.

CHAIRMAN POWERS: I would think that is probably the earliest we can schedule it anyway.

MR. COLACCINO: Yes, sir.

CHAIRMAN POWERS: But I appreciate your comments, and we are going to try to follow up on that.

MR. COLACCINO: Thank you.

CHAIRMAN POWERS: Similarly, I would like to get AREVA's thoughts on these subjects, as they come to mind.

MR. PARECE: Next slide, please. So we are on slide 11 in the package now, and this just gives you an overview of the containment design parameters.

You may have seen this in our last presentation of the overview of the plant, but we just want to point out: The reactor building system is comprised of a containment building, a post-tensioned containment building, concrete containment building with tendons, and also has a quarter-inch steel liner, and it is surrounded by a shield building that is reinforced concrete to protect the containment from external hazards.

The volume is about 2.8 million cubic feet, which makes it similar in size to large containments here in the U.S., and we give some dimensions there: About 153 feet in diameter on the inside. We will be talking about the containment building and its performance today, not the shield building in particular.

The design pressure is 62 psi gauge, which is a little higher than some units, and that design pressure was selected purposely based on the design basis events and certain beyond-design basis events.

So this design: We have in-containment refueling water storage tank, and that is typical of a lot of advance reactors. It is down in the bottom of the containment so that liquid collects in the bottom of the containment during events as the condensation occurs. I am pointing it out down here.

Also we have features for issues regarding GSI 191 in this floor. We are going to talk about this floor. It is called the heavy floor, and we are going to discuss that today.

The reason we call it the heavy floor is that the steam generators, the pumps, the large components -- the supports actually sit on that floor. So that floor supports the steam generators and the pumps, and thereby supporting also the coolant lines.

In that floor, we have large holes for water to drain down to the IRWST from a postulated break inside that containment, and we have racks over those holes to prevent large debris from getting into in-containment refueling water storage tank.

Also, below those holes we have baskets so that water and debris that goes in the basket fills the basket and spills over the top. So large debris gets collected in the basket.

Then inside the IRWST we have strainers for the emergency core cooling system that takes suction off the IRWST, and those strainers have a very small mesh, and we have features, back end angled strainers, so that caking tends to fall off, and we can also -- We have flushing that we can have, active flushing from the ECCS system to flush the strainers.

So it is a very robust approach from that point of view, and that is all in this area here, inside the equipment space.

MEMBER SHACK: Do you have a rough square footage of strainer area or are you depending on the black flushing?

MR. PARECE: No. We count on the size to keep the delta P low, and those -- We had to revise the design a bit because of the seismic requirements. At OL3 the seismic is 0.1g, and we are designing 0.3g. So we wanted to beef up the design from a seismic point of view, because these are safety related.

So we have made some adjustments to the design, and we've got testing at Alden Labs later this year to finish that design up. The OL3 design has been tested with and without debris to characterize the delta P. We are going to do the same types of testing, but on this beefed up design.

The area is generally significantly more area than we need, and that is generally the design approach. The holes in the heavy floor and the baskets and all of that are based on our 4 by 100 design.

So we tend to oversize and keep delta Ps low and keep the flow areas large, because during severe accident we also count -- and we are going to talk about that in a moment. In severe accident we count on hydrogen mixing and air mixing through circulation through the large holes in that floor. So we've got water draining down. We've got vapor coming up. So we oversize all those holes.

MEMBER SHACK: Plus you are mitigating the large break LOCA. Is that unique to the EPR or is that a carryover?

MR. PARECE: I would say that it is -- I wouldn't call it a carryover. I would say it is a combination. The design philosophy for the EPR was to take the best of the French and German designs, and at the time this started, that was the N4 and the Convoy.

So the German and French engineers

worked together on trying to take the best from the different units, and when you look at individual design features, you can see that.

This is truly a combination of approaches and, as we talk about it later, we will talk about hot leg injection during the LOCA, and that is a uniquely German approach to mitigating the event, and we will discuss that. But it has been adjusted based on the French approach as well.

The design leak-rate for the containment is about a quarter of a percent per day. That is, in fact, set up based on the radiological approach, and we are not going to talk about the radiological today very much, if at all, but I just wanted to point out that any leakage through the liner into this annulus is filtered.

So it goes through iodine filters before it can be released, and any leakage around the penetrations into the surrounding safeguard buildings go into radiological controlled areas where those are also filtered.

The fuel building, if you have a fuel handling accident, there is a safety related system to filter that. So all radioactivity releases from EPR are filtered by a safety related system during any postulated accidents.

Mouse Click. The EPR is set up with a two-zone containment, two-room containment. This two-room containment is how the plant normally operates. So the red area that you see on your slide is what we call the equipment space, and for obvious reasons, including radiation and temperature, no one is allowed in that space during operation, but the space around it that is white -- you see this area here, the white spaces -- that is called the service compartments.

The service space, you can have access at anytime during operation, at power or not. This environment is maintained so that you can have access. So what this means then is, when the plant is operating, it is a two-room containment so that we can control the two areas separately.

You can imagine -- and we have -- These compartments are closed on top, and you can imagine, if they were open, the convection -- you would be cooling the whole containment and trying to maintain the whole containment. In this way, we can have one HVAC system cool the equipment spaces and a separate HVAC system cool the service spaces.

Those are done with non-safety related HVAC system using chillers with operational chilled water. So the cooling of the containment is a non-safety related function for operation of the unit, and those chillers are inside these service compartments.

We also have a -- Well, what we are going to talk about today is how we transition to a single-room containment when there is a loss of coolant accident or other pipe rupture, and we do that using a subsystem with call a CONVECT system, and it has rupture foils and convection foils on top of those steam generator compartments that will open, and we also have dampers in the bottom here near the IRWST that opens, and by opening they basically then connect both sides, both rooms.

That allows water vapor and hydrogen and other gases to circulate based on the mechanics of buoyancy due to the warm energy and cooling on the liner and the other containment structures.

As part of that then, there is no

safety related fan coolers on this unit, and there is no safety related containment sprays. What we will talk about today is how, during an event, the energy from the core and from the sensible heat of the reactor coolant is basically condensed as hot water vapor in the in-containment refueling water storage tank where the heat is then removed by our safety related residual heat removal LHSI systems.

So essentially, in accordance with GDC 38, our heat removal system is the RHR LHSI system. We will talk about that in detail today.

We also have -- For hydrogen control, we have passive autocatalytic recombiners distributed throughout containment. There's 47 of those, I believe, and we use those for hydrogen control. They are predominantly for severe accident. So we won't be talking about the severe accident mitigation features today. We are going to be talking about the containment response to design basis events.

That is a good segue. We do have severe accident mitigation -- Yes?

MEMBER SHACK: On that severe accident system that you have, that is really a separate cooling system. Right?

MR. PARECE: Yes.

MEMBER SHACK: There are safety related ones for the design basis accidents, and then a separate -- completely separate system for the severe accident one?

MR. PARECE: That is exactly correct for the severe accident. So the severe accident system -- The containment heat removal system is what it is called in Europe. We call it the severe accident heat removal system, because it is just for severe accident. So eliminate confusion.

That system has a dedicated component cooling water chain and a central service water chain that dumps the heat to the ultimate heat sink, and that system then is used to mitigate the severe accident, and it is used to cool the concrete below the spreading area and used as a containment spray to reduce the building pressure long term, well after 12 hours.

So we could talk a whole day on severe accident, but the main features that we have: We have features to depressurize the unit to low pressure, to prevent a high pressure melt-through.

We've got the passive autocatalytic

recombiners to reduce the hydrogen concentration to prevent deflagrations, and then we have a spreading area adjacent to the in-containment refueling water storage tank where we collect any melt from the postulated core melt and melt-through on the vessel, and then it distributes in a spreading area where it is passively cooled for at least 12 hours due to gravity feed from that in-containment refueling water storage tank.

In a nutshell I have given you many of the features for severe accident, but again today we are not talking about severe accident very much.

CHAIRMAN POWERS: Is the passive hydrogen system safety related?

MR. PARECE: No. I don't believe it is. I'm trying to remember. The guy that would know that is not here today, but I don't believe it is safety related? I don't think we need the passive autocatalytic recombiners to keep the containment inerted during a design basis event, a loss of coolant event. But we do need them to reduce the hydrogen concentration during a severe accident. So they are predominantly to mitigate severe accident.

CHAIRMAN POWERS: I suspect we will go about a day over the hydrogen recombiners. Indulge us a little bit on that one.

MR. PARECE: On Slide 12 we talked about the two-room containment, and the primary way that we transition to a single-room containment is again, as I said, by rupture and convection foils that open on top of the steam generator compartment, as you can see in the picture, up on top of the steam generator compartment, and by dampers that open down near the IRWST elevation. That connects the service space to the equipment space.

The convection foils: We have two sets of foils. They sit in a frame, as you can see in the picture. There is a frame, and the rupture foils on a delta P, as they pressurize, they will open, and they don't burst as you would think of a diaphragm bursting. There is a stress riser on them, so that as the pressure goes up, they essentially tear and open up.

They can open bi-directional. so if the delta P across the compartment is positive or negative, they will go whichever way they have to go. Those rupture immediately.

We also have, in addition to those, a number of foils in a frame that, if the temperature exceeds approximately 180 degrees Fahrenheit, they will open. So if you have a small break that pressurizes very slowly and possibly keeps the delta pressure across the foil low enough that it doesn't open, eventually the temperature will cause these to open. Next slide.

Slide 13 shows one of these installed.

MEMBER STETKAR: When you say temperatures open it, is that an active type opening?

MR. PARECE: No, it is a passive --

MEMBER STETKAR: Fusible link type?

MR. PARECE: Fusible link, and it basically melts.

MEMBER STETKAR: Thank you.

MR. PARECE: In fact, if we look at Slide 14, that shows what happens.

So what is interesting about the convection foils is that they are actually made up of rupture foil. So if the pressure goes up, they will rupture.

If we go back one slide, on Slide 13 you can see that those are rupture foils in the middle of that frame. So if the pressure goes up, they will rupture, but if the pressure goes up very slowly on temperature, then they will open.

A small break then also pressurizes more slowly, puts energy in containment more slowly, and the requirement isn't to mitigate the building pressure immediately. Their environment is just to provide a long term circulation path.

MEMBER MAYNARD: How critical is the integrity of these for normal operations? If there is a tear or something during normal ops, what impact does that have on normal operations?

MR. PARECE: Well, essentially -- That is a very good question. You would have to, during the outage, inspect these visually and make sure that they are all intact. The biggest issue is -- The answer to your question is "depends."

The biggest issue, if keeping them closed, is warm air from the equipment space getting into the service space and then affecting how well you can cool the service space, but that is a primary issue. So if you have a rupture foil that is open, then you probably have to replace it for operational concerns.

From a safety concern, obviously, open

would be the way to be. So it is not like a tech spec issue where, if one is open, you couldn't start up the unit, but if you get convection currents through there that is causing heat to go in your service space, you are going to tax your HVAC system and your ability to keep the temperature in a habitable zone.

MEMBER STETKAR: Would it also affect accessibility from a dose perspective? Personnel -- because you do normally have operators inspecting at least the accessible areas.

MR. PARECE: Yes. The expectation would be, if you had one of two of these foils torn, it would not be a significant change in the dose field, but we do maintain the dose field relatively low, even in operation in that service space.

The design goal for the EPR is less than 2 mr per hour. So it might have some small effect, but it wouldn't be significant. The big thing is, from an operational point of view, if during an outage someone damages one, you would want to replace it. And they are in frames, and they are easily replaceable.

We have tested these foils. We have

pressure tests, and we have temperature tests. So we have a fairly extensive testing on them, and I brought a little show and tell with me.

Now this is one that has been tested, and we can pass this around to look at it, but we've put some tape on it. Be careful. It is sharp. So as Sandra said, we don't want any OSHA reportable events here today, but you can see. This goes into the frame that you saw, and this is one of the rupture foils that we tested.

It is fairly sturdy stainless steel, but I don't remember the thickness. Anyway, the rupture pressure on these is about 0.7 psi plus or minus about 30 percent. We will pass that around.

There are 120 convection foil units, and there's -- got to be 28 times 4 is 112 rupture foils. So there's 30 units per loop and 28 units per loop respectively between convection and rupture foils. I will show you later how they are arranged when we go to the proprietary session. We can do more details on the data.

Next slide. So the way these foils perform then is for large breaks, all the foils open, and for small breaks, there will be a mixture; and as you go smaller in size, you will eventually transition to convection foils. They just provide the long term recirculation.

As you can tell from the example we are passing around, no debris is generated from the rupture of those foils, because it is all self-contained and attached. It stays attached, once it ruptures.

They are safety related. They are designed to meet seismic requirements, because we need them to operate during the design basis LOCA. Back up a couple of slides. We've talked a lot about foils.

The hydrogen mixing dampers: Here is a picture of one you can see at the bottom of Slide 12. There's eight of those spaced around containment, and they open on a differential pressure between the service space and the equipment space around a half a psi, but they will also open on a global pressure.

So if the containment pressure -- Right now that set point will be set by the safety analysis, but right now it is about 5 psi. At 5 psi gauge, they will also open. So these are spring loaded, and they are held shut.

The I&C system, when it detects -- I should say our digital protection system, when it detects a high building pressure, then it will send a signal to the dampers to open, and then they fail open on loss of power.

Next slide.

MEMBER SHACK: They will blow open on the global pressure?

MR. PARECE: Yes, on the global pressure, when you hit 5 pounds, they will open. But if you get a delta P in the equipment spaces, they will open. So again, if you have a fairly large break, anything above 4 or 5 inch break, you are going to get a delta P, and they are going to open. If you have something that is really small and the convection foils open first and you start venting vapor to containment, you will slowly start pressurizing the containment, and eventually you will hit the pressure set point for containment, and the dampers will open.

CHAIRMAN POWERS: On your previous slide, on Slide 15, it says MAAP4 analyses addressing CFR 50.44 show good mixing.

MR. PARECE: Yes. That is true, but what we are going to talk about today is we are going to talk about a number of analyses that have been performed. I wasn't going to talk about the MAAP4 analyses, because those were for severe accident, and in that case you are looking at the performance, but you are looking at the hydrogen mixing, in particular.

CHAIRMAN POWERS: Where in the MAAP4 has -- treats the momentum equation as a lump node code. So you had to assume good mixing to get the answer to good mixing.

MR. PARECE: Well, that is a possibility. I didn't do the severe accident analysis, but what we are going to talk about today is we have other codes between GOTHIC, and we are going to talk a little bit about other qualifications like GASFLOW, which is a Los Alamos code, and we've got some global analysis that we've performed.

So we are going to talk specifically about mixing and what we are predicting with these. So we are going to talk in more detail.

CHAIRMAN POWERS: I hope that there is something to substantiate that?

MR. PARECE: Yes, there is.

MEMBER RAY: You talked about

qualification testing. Have you come to any conclusion about surveillance testing, either the hydrogen or the dampers or the foils?

MR. PARECE: Well, yes. We do extensive testing to validate the foils. So we don't expect to do a surveillance test per se on the foil itself. We already have qualification tests on that.

What you will have is you will have to do in a visual to ensure that they are in place. On the dampers, you can test those easily by sending a signal from the I&C system, and then watching them open.

MEMBER RAY: And you would expect to do that?

MR. PARECE; Yes, I would expect to do that, because they are safety related. So we would test those, and it is a relatively easy test to do for the dampers during refueling outages.

So we have talked about the convection foils and the dampers and how they will open to connect the containment into two parts. What we are going to talk about then is the overall containment concept and the design concept and how we mitigate breaks, postulated ruptures.

So if we have a postulated rupture inside the equipment space which will then cause it to pressurize, then the foils will rupture, and the dampers will open, and that allows the energy to get out of the equipment space up into the service spaces and up into the equipment -- into the containment dome.

The steam will begin to condense on all of our heat structures, our containment concrete and our containment liner, and that condensation then will flow by gravity back down toward the bottom of the containment where it will drain into the in-containment refueling water storage tank.

So now we have warm, saturated water moving into that tank. In the short term, the pressure peak we get from the blowdown is mitigated purely from the physics, the size of containment -- you know, the volume of it and the mass and energy that is in the building. In other words, in the first 30 seconds during blowdown, you can vary the convection.

You can vary the condensation by large amounts, and there is very little impact on the peak pressure. The peak pressure is driven by ideal gas law and the energy that you are distributing. So the containment is sized to ensure the blowdown peak is well below the design.

CHAIRMAN POWERS: I would hope it is not dependent on the ideal gas law. The gases are minimum idea.

MR. PARECE: Well, all right. But I'm trying to simplify the point of view that air is our single biggest contributor to the pressure. Heating that air causes the pressure to rise significantly, and then you have the partial pressure contribution of the water vapor from the reactor coolant system. So those two partial pressures give you your peak.

In the longer term, we have actuation of our emergency core cooling system. So we have medium head safety injection pumps that take suction from the IRWST and inject into the vessel. We have low head safety injection pumps that do the same, but before they inject, they pass through our RHR heat exchanger and cool the water.

So that is our main place to take energy out of the building. We take the water out of the in-containment refueling water storage tank, and we run it through a heat exchanger, which cools it, and then we reinject it into the vessel.

So as we do that, we reinject into the vessel, and we set up a circulation where steam from the reactor coolant system goes into the building, condenses on the walls, goes into the IRWST where the RHR system takes the heat out.

In the longer term then, at a certain point in the transient for boron concentration control and for steam suppression, we open valves on the low head safety injection that are normally closed and allow flow to go to the hot legs.

So we get each of these systems. There's four by 100 systems. So we have four LHSI systems. Each run around 1900 to 2000 gallons per minute each. So when we turn those on, the majority of that flow is rerouted to the hot legs.

So at this point, we have subcooled water going into the hot legs and going into the vessel, and essentially causing steam to condense, and we get mixing in the core which causes a reduction in the steaming rate of boiling in the core, and we put warm water out the break, and we eventually suppress the steaming.

So in the long term, we get to a point

where we are suppressing the steaming -- or stop the steaming in the hot leg, in the core exit, which reduces the steam going around the steam generators through the broken loop to the break site.

So once that steam contribution to the containment stops, then we have condensation, and we continue to depressurize. The long term depressurization of the building is assured by the condensation of the vapor on the heat structures.

So over the long term, that is how in the design basis phase we transfer the energy from the core to the IRWST.

The other thing I would note, on the picture you have on 16, my colleague, Louis Charles, has reminded me that that line that you see from the LHSI that goes to the IRWST actually occurs inside containment, not outside containment. But the point I wanted to make with that little cartoon is that we also send some of the cooled water back to the IRWST. So we are cooling the IRWST as well as cooling the core.

Now this is what we have kind of talked about. A key to making the containment approach work, as you can imagine, if you have a cold leg break in particular, it will eventually get to a point where you have steam being produced in the core and, as long as you have steam being produced in the core, that steam will condense on the emergency cooling that is available, but predominantly steam will go through the broken loop through the steam generator and over to the break, and you will have an almost constant steam source to the containment, and without a fan cooler or a spray system, your containment pressure response is based on the ratio of condensation to steam production.

As you can imagine, our heat structures will heat up over time, and the efficiency can reduce over time. To suppress the steaming then, the hot leg injection -- and on page 17, this just shows a cartoon of a large amount of ECCS flow coming into the hot leg and into the upper plenum, and then cold water will mix with some of the warm water in the plenum, but a large portion of it will go down the core, one part of the core, where if you have boiling especially but heating on the other part of the core, some of it will mix and migrate to that side of the core. Then warmed water will go up the downcomer and out the break.

MEMBER STETKAR: Marty, you said the switchover to hot leg injection is manual?

MR. PARECE: Yes.

MEMBER STETKAR: Is that also in the European version?

MR. PARECE: Yes.

MEMBER STETKAR: It is?

MR. PARECE: It is. It surprises you that the Germans like automation?

MEMBER STETKAR: It does.

MR. PARECE: Yes. We have a design rule on EPR that says no design basis accident should require a manual operation before 30 minutes. So if it is before 30 minutes, it is automated. If it is after 30 minutes, it tends to be manual, and you would follow your emergency operating procedures. We will discuss later, but those switchover times, based on meeting the acceptance criteria, can be anytime after 30 minutes and probably anytime before 90.

MEMBER SHACK: But globally they will switch at 90 minutes, and in the U.S. we'll switch at 60?

MR. PARECE: Well, it is interesting you picked up on that. Yes. Right now, we are discussing with our European colleagues about when would be the time to credit. It is really -- The time that you pick has to be validated by training the operators in the simulators, and so they can get through the EOP steps in that time. We are pretty confident they can do it in 60 minutes. I am very confident they can do it in 90.

Today we are talking about the evaluation model and the safety analysis and the design basis accident and the safety related equipment, but obviously, if an event were to happen on a real plant, the operator also has other equipment, and he does have the non-safety severe accident heat removal system that he could use but, obviously, we don't credit that. Next slide.

MEMBER SHACK: But his emergency operating procedures tell him he has this system?

MR. PARECE: Certainly. His EOPs -- Certainly. EOPs take credit for everything you've got in the plant.

CHAIRMAN POWERS: We will have more on an adequate exploration of human errors of commission for this plant.

MR. PARECE: So on the next slides what

I want to do is set us up for the rest of the presentation. So as we talk about circulation patterns and multi-dimensional models and what-not, that in your mind's eye you've got a good idea of where stuff is and where everything is going.

So on Slide 18, this shows a section view of the containment, which many of you have seen many times. We've cut it through the steam generator compartments.

This bioshield wall here is the separation of the equipment space from the service space, and you can see the reactor vessel is in its own compartment, and we have cut through the refueling canal here.

As we talked about, you can just seen the holes in the heavy floor, and down here you can see the strainers. This is the general layout for the containment from an elevation point of view. Next slide.

So if we go all the way down in the basement -- If we go down in the basement, you can see here, this is the reactor pit where the reactor vessel is placed, and you can't quite see it, but there is a transfer tube. This is the spreading area, which is a severe accident feature for the corium to spread in case of a severe accident where it can be cooled.

This area is our in-containment refueling water storage tank. I will point from both sides. We will have to swivel our heads. Here you can see the footprint here and there and there and there and there. Those are the footprints for the baskets I talked about that go under the flow holes. So that is the footprint for the baskets.

These four rectangles are the footprint for the strainers for the ECCS. So each one of those would be connected to its own line and to its own division. Each division has a separate intake and strainer, of the four divisions. This is our containment wall, and you can see this is our shield wall. Next slide.

So now we have moved up, and you can see the heavy floor. This is where the reactor coolant components will sit. We are cutting through the reactor vessel. You can see the lower head, and here are four drain holes that have those baskets. These are the four drain holes that have those baskets -- racks on top. We call the trash racks on top.

So we have a trash rack on top, and then right below that are those baskets that we showed the footprint for. In a moment you are going to see, this is where our pressurizer relief tank is going to go.

MEMBER SHACK: How deep is the basket?

MR. PARECE: Oh, you got me. I knew that number. I don't remember. Next slide.

So now you can see, we have come up a bit, and now you can see the subcompartments, and there's a few things I want to point out in this.

Here our pump compartments are these areas on the quadrant. That is where the pumps are, and here are steam generator compartments. You can see a wall has popped up. Now at the bottom of this wall, it is open, but right here this wall has popped up. What I want to point out is that there is a concrete wall between every hot leg and cold leg, not just between the loops, but between the hot leg and cold leg on a single loop.

So there is a concrete wall around every reactor coolant pipe which limits how far your zone of influence would be on a break with regard to debris generation and the effects on our metallic insulation. We use reflective metal insulation.

Here you can see the pressurizer relief tank sitting there, and -- Next slide, please.

Now you can see the steam generators. We have cut through the steam generators, and you can see the steam generators in their cubicles, and the pumps. You can see the tops of the pump motors. You can see the four accumulators. They are in the service space. So here is one, two, three, four accumulators.

The ECCS system is very much like existing PWRs. Each division has an MHSI medium head safety injection, a low head safety injection, and an accumulator. They combine into a single line that goes into each cold leg. So they are connected -- Separate divisions are connected each to its own loop.

So there are the accumulators, and you can just see up here in the upper righthand quadrant, we have just cut through the bottom of the pressurizer. So this is where the pressurizer is, in this cubicle here.

Then the thing to note about -- This is the refueling canal, obviously. This is a storage location that we use during refueling. This area on the left is for instrument lines that we pull out of the core and other things we store there.

This right here, this narrower one, this is the transfer tube and where we bring the -- transfer the fuel between the fuel building and the vessel. So down at a lower elevation there is actually -- there is a transfer tube to the field building. So the field building is south on this arrangement. Next slide.

This elevation shows us the operating floor.

MEMBER STETKAR: Marty, as long as they are talking, do you do a full core offload refuel or do you just do a fuel shuffle?

MR. PARECE: The answer again is it depends. Right now, our outage is designed -- the fuel pool is designed, boration, heat removal. Everything is designed to do full core offload, and that takes about six assemblies per hour. It takes about 40 hours.

So we are designed for a full core offload. But given that -- and a full core offload helps you, because if you are doing steam generator inspection or pump seal work, then you just drain down. You don't have to do nozzle dams, whatever. But given that you would do samples of tubes and you would do pump seal work delay maybe every third outage, in those intervening two outages you would want to do a shuffle. So we can do a shuffle as well.

So this just shows the operating floor and the slabs that are over the refueling canal during normal operation. Next slide.

So you can see the top of the steam generator cubicles and also here is our equipment hatch in the lower righthand quadrant, and that equipment hatch allows you to take large equipment through the fuel building and out through a hatch in the wall, so it can go out of the building, out of the plant.

There is our refueling machine parked at the bottom. Next slide.

MEMBER MAYNARD: What level is your equipment hatch compared to ground level?

MR. PARECE: The bottom of the equipment hatch lines up with the operating deck, which is 19 1/2 -- plus 19 1/2 meters.

MEMBER MAYNARD: So above the ground. Okay.

MR. PARECE: Above, yes.

MEMBER SHACK: Can you get a steam generator out without breaking down concrete walls?

MR. PARECE: We can get everything but the reactor vessel out without deconstructing anything. That is not just for the containment building. We have a design requirement that all equipment, except there is some equipment in the turbine island -- but all equipment has to be able to be removed or replaced without deconstructing anything in the plant.

So we have pre-engineered hall routs for all equipment, and we already have pre-engineered lifting points for all the equipment and, if something is at a lower level, we have grates in the floor. So you can pull the grates out, and you can grab the heat exchanger, for example, and bring it up to the level and then get it out of the building. So there's hatches on the different safeguard buildings.

Now the design life of these components is 60 years, but we designed it anyway so that you can replace the major components.

The reactor vessel fits through that

equipment hatch and can be handled by the puller crane. The thing is, if you noticed how we built up from the bottom, there is concrete around the vessel. So we would have to take out some concrete to replace the vessel. Before we did something like that, we would look at other ways to extend the life of the vessel.

So on Slide 25, this gives you a better picture. These blue areas on top of the steam generator cubicles are where we lay out those rupture foils and convection foils, right in there. So you can see, they are also relatively open to the containment up there.

This little area right there you can just see, that is actually a storage place for the reactor vessel head during refueling operations, and it has a wall that you can put in place to prevent people from the shine from the head, but it also has a stand, so that you can do inspections on the top or the bottom or do any activity on the head you want off critical path while people are doing other operations. Next slide.

So I've just plummeted you back down to the bottom of the building, and this shows the location of the eight dampers down near the bottom. Next slide.

This kind of shows you what it looks like down there, kind of a three-quarter angle shot of the dampers. So you can see where they are, obviously, below the heavy floor but above the IRWST, the in-containment.

Let me step closer to the microphone. Next.

So we've got an overview of what the containment looks like and what makes up the equipment service spaces and where the dampers and the foils are. So let's talk a bit about the methodology that we are using to analyze the design basis accidents for the containment.

We submitted a Technical Report to the NRC, ANP-10299, and in that report there is a lot of mileage in that report on validation of the evaluation model, including the evaluation model development and assessment process (EMDAP).

So we describe the M&E, mass and energy release models that we are using and the GOTHIC models and approaches that we are using for the containment response, and also looking at uncertainty analysis, and we have discussed our scaling methodology.

If I am correct, we have a Revision 2 of this coming in December, which will have the results of the of the scaling analysis.

We are going to go into it in a moment, but we -- As part of the uncertainty -- We did an uncertainty analysis. We will talk about the PIRTs that we looked at for mass and energy release and for containment, and that PIRT -- As you know, doing that phenomenon importance ranking table is a thought exercise by experts based on what they know and the state of knowledge of the different phenomena.

That is what I would consider to be a top-down approach. We also, by doing a scaling analysis where we look at -- we do a non-dimensional analysis of the state equations that would affect the building, and from that we develop our non-dimensional groups as they relate to the parameter of importance -- say, in this case, building pressure at different times.

Then from that, those non-dimensional groups should give you insights to your PIRT. The importance of those non-dimensional groups should match up with the phenomena that you have selected in your PIRT. We are in the middle of that scaling analysis right now. It is not finished yet.

The other thing the scaling analysis can do for you is it can look at what I would call differences in aspect ratios of the test facilities that you are looking at. We are going to talk about some test facilities later, some containment test facilities that we benchmarked, and by looking at them, you will tell that they don't exactly match the aspect ratios, for example, of the EPR containment.

So from our uncertainty analysis, we would identify any of those non-dimensional parameters that seem to be or appear to be of higher importance in the facility versus the EPR containment based on its dimensional characteristics.

You have just about exhausted everything I know about how we are doing the uncertainty analysis. We will be talking about that.

CHAIRMAN POWERS: We have not exhausted my questions on the uncertainty analysis.

MR. PARECE; Well, and we are going to

talk about that later, because on the uncertainty analysis we also then -- of those PIRT and those important parameters -- so put the scaling part aside. From those important parameters, we then looked at quantifying what the uncertainty might be on those parameters, whether it is wall condensation, material properties or whatever, and then we took a statistical sampling, and we ran 49 cases with those variations to determine the sensitivity of the pressure to those, and we are going to talk about that a little bit later.

The requirements, of course, that we are looking at: GDC 50 requires that your containment be designed to handle the pressure and temperature conditions following a Loss of Coolant Accident, and GDC 38 requires the containment heat removal system to rapidly reduce containment pressure and temperature following a LOCA.

The way that is sub-defined is in the Standard Review Plan. The Standard Review Plan basically says that, if you show that your pressure at 24 hours is half the peak pressure, then you have demonstrated the adequacy of that cooling system.

So those are the dominant GDC.

So in our evaluation model development program, we looked at PIRT for the containment for the mass and energies. We used some existing PIRTs and demonstrated their applicability to the EPR, and then we had our experts look at any differences between the EPR or special functions or features of the EPR compared to what might have been looked at in the PIRT to make sure we identified any other issues. So it based on specific PIRTs that exist. Next slide.

We also did an assessment of the -- We looked at the data assessments for RELAP5, which is our mass and energy code for the early blowdown phase and reflood. GOTHIC -- we use GOTHIC for long term M&E. So we just -- GOTHIC does the M&E and the containment building internally in the long term.

We looked at previous test data and code assessments, and for the scaling we developed equations for scaling analysis. So our intention when our scaling analysis is complete is to demonstrate that the GOTHIC benchmarks in particular that we used to validate the EPR response are applicable to the EPR geometry.

So when we went through all of this in

the PIRT, what we determined was that, from the previous code assessments and from the PIRTs that we did, that RELAP5-BW is the code we are using. It is a version of RELAP5, MOD2, and it was originally developed for Appendix K type applications, and GOTHIC which is pretty standard industry containment code now -- that they predicted the medium or high ranked PIRT phenomena that were important, except we had a couple of notable exceptions.

That process we talked about with hot leg injection is a multi-dimensional process, and RELAP5 is a 1-D code. So RELAP5 couldn't model that.

The other issue was interfacial heat transfer between the in-containment refueling water storage tank and the atmosphere. If you remember the geometry, it is pretty complex down there. We have a roof over the top with holes in it, and the water level is down below. How wavy is the water level?

We've got water running across the floor and dumping into the holes, and it is relatively hot, because that water is coming from the reactor coolant system through the break, and so the point was you have a lot of uncertainties about any heat transfer mechanism.

So our approach on these two things is to use conservative biases or analytical treatments. In particular for the IRWST, you can just turn the heat transfer off between the interface heat transfer. So we did.

So we will talk about that later. Some of these treatments we will talk about in the proprietary session.

So in addition to all the various code benchmarks that are out there for these different codes, we did run some specific items from our interaction with the NRC and from our own approach, for our own edification.

So for the RELAP assessments against FLECHT-SEASET data show that we have good carry out from the core, a good movement of the quench front during the refill or refill of the core.

Why that is important is that moisture goes out through the broken loop and into the steam generator where it gets vaporized due to the secondary to primary heat transfer. So that is a source of energy into the building. So we validated that, and the heat transfer from secondary to primary from the FLECHT-SEASET data was also well predicted, and that is the second piece of what I talked about.

So those were some additional benchmarks that we performed, and then for the hot leg mixing and condensation efficiency, we looked at a number of tests, including cylindrical core test facility, the slab core test facility, the upper plenum test facility.

So we looked at how those facilities worked, and we did some benchmarks to those tests, and then we also looked at some CFD codes and what they would predict for mixing, because again you need a link -- In many cases, you need a link from the test facility to the EPR geometry and flow rates.

So we validated that hot leg injection eventually terminates the steaming to the containment, and we developed a conservative model which we will discuss some details in the proprietary session.

Then we assessed GOTHIC multi-node and single-node models against a number of tests, but we were very interested in two particular tests in particular, HDR, Germany acronym, HeissDamph

Reaktor; and BFMC is Battelle-Frankfurt Model Containment.

These were of particular interest to us. GOTHIC has been evaluated against a number of test facilities for a number of -- all the models and phenomena, but these were important to us, because they had some tests that were no sprays or fan coolers, which is like our design basis, dry containments, meaning that they didn't have suppression pools, and then multi-compartments.

So that was important to us, because our energy flows from one compartment to the bigger compartments, and these gave us good benchmark opportunities.

We looked at -- Many of these tests are short term, but we had a few longer term tests that were important to us.

We also looked at not just pressure and temperature, but we looked at -- We have some tests where we benchmarked the hydrogen concentration predictions and other phenomena, but today we are going to talk mostly about pressure and temperature.

Then the other thing that is in our technical report is an analysis with GASFLOW, which as I said, is a Los Alamos code for modeling gas transport and combustion of various gases and vapors. We looked at a model of the EPR and looked at response of the LOCA and the building to that LOCA over a long time period.

The uncertainty analysis follows the Code Scaling and Applicability and Uncertainty approach. In that methodology, you look at the range of values and look at sensitivities of your parameter of interest, which in this case was predominantly pressure, to a range of values that bound the value of the parameter.

We confirmed the important parameters were identified in the PIRT. So the PIRT tells you what should be important. Then you do a sensitivity analysis and look at a large number of cases. As I said, we did a sample. We looked at 59 combinations.

When you do the analysis, it should validate the PIRT. In other words, if -- You shouldn't get any surprises. If you did your PIRT and you are all smart guys and you know the phenomena and you know the processes, then when you do the uncertainty analysis, it should show you that the parameters that said we were medium and high are, in fact, showing up as the ones that affect the sensitivity.

If you get something -- you know, some importance that you didn't expect, then you have to go back and revalidate your approach. So that -- I like the way those two things work together, because one is a thought exercise, and then it is backed up through analytical results of the uncertainties, and you try and bound the uncertainties.

The modeling approach that we took, of course, meets the regulatory requirements of Standard Review Plan, and I will just reiterate again for later. The codes we are using for mass and energy release in the short term, we are using RELAP5, and in the long term we are using GOTHIC. We will discuss -- The GOTHIC model is a multi-node GOTHIC model that represents the different compartments of the EPR containment.

CHAIRMAN POWERS: You have spoken extensively about validation and handling of mass and energy going into the containment line break, certainly a key part of the regulations.

You have not said anything about the containment vis a vis 10 CFR Part 100 and the source terminations in the containment. AT what point will we discuss those?

MR. PARECE: I was not going to discuss those today, but we can. You are being chastised for not being close to the microphone.

CHAIRMAN POWERS: I get even right after the session is over. He has to write the meeting report. He may find it is difficult to get it approved.

MR. PARECE: I was not prepared to talk about the radiological part today, because --

CHAIRMAN POWERS: That's fine. Just when we discuss that.

MR. PARECE: Well, I guess --

CHAIRMAN POWERS: The containment is there for a purpose.

MR. PARECE: That's correct.

CHAIRMAN POWERS: And it is radiological in nature.

MR. PARECE: So we are okay on time. So I'll give you a preview of that. We are using alternate source term methodology. So the source term into the building comes entirely from that, and then --

CHAIRMAN POWERS: I am unaware of an

ultimate methodology that is appropriate for this plant.

MR. PARECE: No, we believe the ultimate source term approach is appropriate, that there have been approvals for other units. Westinghouse doesn't have a safety related system either, and they are using alternate source term.

So that can be presented to you at another time, but we use the alternate source term approach, and we made one adjustment to the design in the U.S., and that is to buffer the in-containment refueling storage tank solution post-accident. We buffer, so that we keep the pH of the liquid in the IRWST.

So any iodine that is entrained or captured in the water that condenses and goes into the IRWST, we buffer so that we can limit the amount of iodine that goes back into the containment atmosphere.

From that, we use leak rate assumptions, and again all of our leakage, no matter where, is filtered by safety related systems. So we did our dose calculations for both the control room and for off-site dose to the public using that approach.

I will say, we made some adjustments to intakes to the control room to move them away from -- those receptors away from possible sources of emission during an event. In other words, the stack. We treat the stack as a ground level release, even though it is elevated, but we move the intakes away from the stack, and we made some adjustments on dampers and what-not.

The source term in the United States using rules according to alternate source term is generally 4,000 times greater than what is used in the IAEA approach in Europe. So we had to make some adjustments to accommodate that.

MS. SLOAN: Well, Marty, I would also add that we are on the schedule for next July to talk about Chapter 15, and Chapter 15 does address radiological dose consequences. So you asked when. That is what is currently on the schedule, unless you wanted a discussion earlier than that of that particular topic.

CHAIRMAN POWERS: What I am concerned about is that what is conservative for thermal hydraulics may not be conservative for source terms. In fact, it can be absolutely inverse to each other. So I get nervous when we start talking that this is conservative, because it is conservative in a context.

The context here, of course, is thermal hydraulics and pressurization and heat loads. It may not be conservative for source term considerations.

It is unfortunate that things are separated as much temporally as they are. But we will live with it. But I understand, we will come back to what is claimed to be conservative here when we get to the source term issue.

MR. PARECE: And I would expect that, and a source term is done specifically according to the rules of that methodology. So they have long been disconnected.

So we take that as a note, and you will get your chance to review that and discuss that in that other meeting.

So this slide here is just kind of an overview of our prediction of a sample cold leg pump suction break and the prediction by our evaluation model of that break, compared with the best estimate plus uncertainty analysis we did for the CSAU methodology.

So this shows all 59 cases, if various

parameters, and whenever possible we did double-sided sampling. So if there is a parameter that has plus or minus 10 percent, then that was the sampling range. Some parameters, we might not be able to do double-sided, but we would do then either a conservative approach for that one parameter and just leave it that way or do a single-sided sample, either of nominal or the worst case.

This shows the 59 cases all plotted together, and from this we gleaned certain information from this. And in the nature of the best estimate analysis, as we will discuss later, this analysis includes best estimate M&E model, mass and energy model, versus a conservative M&E model and has best estimate to K heat in it, and then a number of sample parameters.

This just generally shows the margin inherent in the approach for the evaluation model.

MEMBER STACK: One problem I had with this, it was sort of an apple and orange comparison, because you did a multi-node model for the evaluation model, and you did a single-volume calculation for the best estimate.

Have you ever done a single-volume

calculation with the input variables for the evaluation model?

MR. PARECE: Yes, we have. What you've seen here is a delay in the approaches. As we will discuss later, our original approach was to use a single-node model and justify that, because it becomes a one-room containment, that a single-node model is appropriate.

Over time and working with the NRC, what we determined was that the questions arose on whether in the long term, as you get out to 20 and 24 hours, whether the single-node model was still adequate, because with a single-node model, by definition, you are assuming perfectly good natural convection and all the surfaces see the vapor.

So in that process, we switched to a multi-node model, and we do agree that the multi-node model gave better accuracy. So in the E&M we switched to the multi-node model. But based on the results that we had and the mixing results we will talk about later, we didn't feel compelled to redo all the uncertainty calculations with a multi-node model, because of the results --

MEMBER SHACK: Just do the evaluation

model with a single-node and -- You know, if you convince me that it doesn't change at all that much, then I will believe this comparison, but at the moment --

MR. PARECE: In general, the single-node model -- A single-node model over-predicts the blowdown peak a little bit, but in the long term it will under-predict the long term pressure by a little bit. And we will show a benchmark later.

MEMBER SHACK: The other thing that struck me as peculiar is that you did the switch at somewhere between 1,000 and 1200 seconds. I am not quite sure just where it was done in these particular calculations from RELAP to GOTHIC.

MR. PARECE: We will discuss those details in the next session. So write that down, and save that question. We are going to cover that.

MEMBER SHACK: Well, this diagram only appeared in this presentation.

MR. PARECE: This one? I'm hoping I have this diagram in the next one, too.

MEMBER SHACK: Got it in the next one, too? Okay. I missed it when I flipped through.

MR. PARECE: Okay. Good. So I believe this is the end of the open session. Right? This is the last slide.

MEMBER RAY: Are the initial conditions for these analyses declared to be what I will call your nominal two volume temperature distribution, two rooms in the containment? In other words, is that a limiting condition for operation or can you operate the plant with loss, let's say, of cooling of the plant?

MR. PARECE: The answer to that question is right now the analyses are performed with the limits for those rooms. For example, the service space limit is 86 degrees Fahrenheit for habitability, and the equipment space for concrete production is 140, but we try and control it to 131. That is a metric -- you know, Celsius to Fahrenheit.

MEMBER RAY: So habitability is, in fact, a limiting condition for operation?

MR. PARECE: But -- But we will do some assessments if there is a utility -- and there is likely to be -- that they don't want instantaneous access all the time, that in fact they would like to run the unit and, if they think they need access, they will turn on the HVAC and purge everything. If that is the case, we will have to run those assessments to see if there is any significant effect of changing the normal operating service space to, say, 90 or 95 degrees.

MEMBER RAY: Okay, but for design certification purposes at this point in time, I assume that, even though the cooling in that space isn't safety related, it does constitute a limiting condition for operation. Correct?

MR. PARECE: Right. So if during operation the temperature started to rise because there is some problem with the HVAC system, they would have to go into those service spaces and see what is going on with the fans or the chillers. But the good thing is the compressors and all that for the operational chilled water system are outside the building.

MEMBER RAY: Yes, sure. No, I understand. I just want to be clear in my mind that these do constitute limiting conditions to operations.

MR. PARECE: Right. And then there is one advantage you get from having a shield building around the containment, and that is that the radiation heating that you get, especially in the summertime at plants in the U.S., doesn't exist. The sunshine and the outside air temperatures are on the shield building, not on the containment. So we've got that going for us.

CHAIRMAN POWERS: Are there any other questions from Members on this presentation? We will be plunging further into the details in the next session.

In that case, we will take a break until 25 after the hour.

(Whereupon, the foregoing matter proceeded to Closed Session at 10:32 a.m.)

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

Time: 1:14 p.m.

CHAIRMAN POWERS: It is time to come back into session. I will remind the members of the Subcommittee that Derek has to give a summary to the full ACRS. So at the end of the meeting, I will ask you for input to Derek's summary that he is giving to the ACRS.

I may have overlooked mentioning that to him.

We are ready to go back into an open session now, overview of US EPR Analysis Methodologies, and Mr. Salm will lead us through this.

MR. SALM: Yes, thank you. I am Bob Salm. I joined B&W in 1973, working in the LOCA Analysis Group. In the early Seventies -- well, middle Seventies, I moved to Germany and was involved in the design, licensing and start-up of Meulheim-Kaerlich Power Plan in Koblenz, and that was a B&W 205 fuel assembly plant like Bellefonte.

CHAIRMAN POWERS: Dr. Bonaca knows that plant, too.

MR. SALM: Yes. And it ran fabulously for a year. I came back to the U.S., was involved in space nuclear and a variety of DOE type nuclear projects for BWXT, retired from BWXT in 2002, worked as a contractor, joined AREVA in 2006.

I am Manager of the New Plants Process Engineering Organization, which contains thermal hydraulic analysis, LOCA analysis, non-LOCA, radiological, PRA and severe accident. So it does most of the analytical -- process analytical work.

I am here to talk about the safety analysis methodologies, and I will start off by presenting the features of the US EPR that are relevant to safety analysis, then talk a little bit about the AREVA methodologies, their applicability to the EPR, and then in a closed session we have picked out three specific methodologies to talk about in more detail , rod ejection, realistic large break LOCA methodology and small break LOCA methodology.

Jonathan Witter is an advisory engineer in the fuels analysis organization, and he is our expert on the fuel methodologies, and he will be talking about the rod ejection methodology. Next slide.

Just as a point of orientation, this is a four-loop PWR, very similar to the Westinghouse four-loop plants. The volumes of the primary system have been scaled up in proportion to the power. The flow areas have been adjusted to have the same velocities, the pressurizer a little bit larger, the secondary side is a little bit larger to give the operator a little more time to respond. But basically, it is a scaled-up four-loop PWR.

A couple of things to notice on this. I will try Marty's laser. We've got the hot legs are grouped together, and the cold legs are grouped together. This provides a more compact arrangement for the components, and on the hot legs you have the nozzles where the RHR residual heat removal system let-down lines are located. Those are the same nozzles where we inject hot leg injection to suppress steaming.

In the cold legs we have the accumulator line nozzles that also are used to inject the low head safety injection and medium head safety injection. Next slide, please.

This is a side view of the plant showing the elevations, and in particular, what I wanted to point out is the relationship between the top of the cold leg and the top of the core and the loop seal, and the top -- the cold leg cross-over pipe is 30 millimeters below the top of the core.

So it is a shallow loop seal which enables steam to be vented more easily. And as Marty pointed out, the steam generator tubes are a little bit larger, and things are done to reduce the pressure drop around the loop, which also promotes the venting of the steam.

Okay. As Marty has already told you, there are four trains of accumulators, medium head safety injection, low head safety injection which also functions as a residual heat removal system, has a heat exchanger in it, and four trains of emergency feedwater, and four trains of main steam relief.

I will tell you a little bit more about the main stream relief train. It is a safety grade system. It is comprised of two valves in series. One is a control valve, and the other is an isolation valve. When the plant is operating, the control valve is open, and the isolation valve is closed.

The isolation valve is opened on SI signal. It also opened to respond to like a turbine trip where you pressurize the secondary side. It has a capacity of 50 percent of the steam flow, and it has a nominal setpoint of 1385 psia.

There are two main steam safety valves, each with 25 percent capacity, and those have setpoint of 1475 psia. So they are quite a bit higher, and for virtually all events the main steam relief train is able to mitigate the event without challenging the safety valves.

When you get an SI signal, setpoints of the control valve for the main steam relief train are ramped down in pressure at a predetermined rate corresponding to 180 degrees per hour from the 1385 psi down to 870 psi, which takes about 20 minutes. This is preprogrammed and has nothing to do with the actual response of the primary system or secondary system. It just a change in the setpoints.

After the partial cooldown is complete at 180 degrees per hour, the operator is able to initiate a 90 degree per hour cooldown, and that is generally assumed in our analyses. As Marty has told you earlier, part of the design of the plant is that no operator actions are credited in the first 30 minutes of the event.

There is another mode for the main steam relief train. If there would be a tube rupture, a setpoint of the main steam relief train can be raised to 1436 psi, which is high enough so it is above the cutoff head of the medium head safety injection.

This way, the medium head safety injection can't open the valves and cause a discharge outside of the containment.

MEMBER STETKAR: Does that happen only on the ruptured loop, or do you raise them all four?

MR. SALM: Only in the affected one.

MEMBER STETKAR: You don't want to blow down the other three.

MR. SALM: Correct. And there is an automatic actuation. If you have completed the automatic partial cooldown and it detects a high level on one of the steam generators, it will reset that main steam relief train based on that high level.

The operator can also do it. There is radiation monitors in the blowdown line and in the main steam line. So if the operator detects activity, he, too, can raise the setpoint.

All right. I guess we are turning to our slide up here. The LHSI are cross-connected, two by two, between adjoining loops, and these cross-connects are opened when any train of LHSI is removed from service for preventive maintenance; and as Marty said, this is to ensure an even distribution of liquid around the downcomer, so it doesn't get entrained out a broken leg.

Let's see. The design also includes an automatic reactor coolant pump trip. This occurs on an SI signal and low DP across the pump. The reason the DP signal is there is to differentiate between a small break, maybe a tube rupture where you would want the reactor coolant pumps to continue to operate, and a larger LOCA where you get two-phase conditions and the pump starts to degrade. That is automatic.

The design has low DNBR and high when your power density trips in containment. Refilling water storage tank -- Marty talked about that, and besides being the source of water for the MHSI and LHSI, it also obviates the need for having a switchover to the sump at some point during an event.

CHAIRMAN POWERS: You are going to have to remind me. Sandra failed me -- all her fault. She promised her crib notes over here. It does not have her this LPD RT. Tell me that acronym. It's all Sandra's fault. Blame her.

MR. SALM: All right. I'm sorry.

CHAIRMAN POWERS: LPD RT is?

MR. SALM: Reactor trip.

CHAIRMAN POWERS: RT is the reactor trip. LPD is?

MR. SALM: Low power density.

CHAIRMAN POWERS: Low power density.

MR. SALM: Linear power density. I'm sorry.

CHAIRMAN POWERS: Ah, we don't even know what this is. No wonder Sandra didn't include it. Too much argument.

MR. SALM: No, I'm in the safety analysis. This is fuel. They don't let us use it.

The design also has an extra borating system. This is a system that injects very high concentrated boron.

CHAIRMAN POWERS: Can you tell me -- I was going to ask. How high is the boron concentration in this?

MR. SALM: I don't know. Does anybody here know? Marty, do you know?

CHAIRMAN POWERS: There is a limit. It is going to saturate sooner or later.

MEMBER STETKAR: Well, the key is how big is the pump?

MR. PARECE: This is Marty Parece. The extra borating system contains a boron concentration in the tanks of 7,700 parts per million, and that is enriched B-10. So it is equivalent of natural boron of 12,000 ppm, and the pumps are positive displacement pumps, and each pump is about 44 gallons per minute. But it does allow you to put in approximately one percent DK over K in about 20 minutes.

MEMBER STETKAR: But it is not an ATWS mitigation. It is a positive hold-down type.

MR. PARECE: Right. So without stealing Bob's thunder, what it allows you to do is allows you to add boron to the plant. So you can reach cold shutdown using a safety grade system from the control room, even with a loss of off-site power and a single failure.

MR. SALM: Thank you, Marty. Okay, a

couple of other design features I just want to mention: The steam generators have an axial economizer in them. What this is, there is a separator plate that comes about halfway up the tube bundle that separates the hot leg side from the cold leg side, and on the cold leg side there is a double wrapper that functions as a downcomer that channels the main feedwater to one side of the bundle.

It is open on the top. There is recirculation, but there is only about 10 percent recirculation on that side and about 90 percent recirculation on the hot side. This allows you to bring your cold leg temperatures down.

The design has a heavy reflector in the vessel. It has a 14 foot core, active core, and -- let's see, what else? I think I have talked about everything. Next slide. Any questions about any of those systems?

MEMBER STETKAR: Yes, only because I haven't read enough yet. In the IRWST suction for the MHSI and LHSI pumps, there is what looks like a three-way MOV. Is it really a three-way MOV? Both pumps take suction at the same time?

MR. SALM: I don't know. Marty, would

you know?

MEMBER STETKAR: One valve only blocks the IRWST suction to both pumps. Right?

MEMBER RAY: I have been trying to think through the four times 100 principle that Marty explained, long term out of service for one and then normal tech specs on what would then be the remaining three. That wouldn't apply, or does it, to the main steam relief train, because I am just trying to think about it as each leg serves just one steam generator.

So any out-of-service time on a main steam train is going to remove 25 percent of your blowdown capability on the secondary side. Does it work the same way in terms of requirements for operability of that train, each train? Involve a safety analysis assumption -- is that right?

MR. PARECE: To answer that question, the main steam relief trains don't normally need maintenance on line. So the main provision for the main steam relief train is that they are operable during operation.

I believe that in our analysis we have assessed the effect of having either main steam relief trains available for over-pressure protection or safety valves available for over-pressure protection, and not both at the same time. However, generally, you would have to have those operable.

So when you take a division out of service, we do take all the big movers out of service and the emergency diesel and part of the HVAC systems out of service, but those main steam relief trains remain operable, because they are on a powered bus, and if we lose off-site power, they are battery backed for two hours.

So you have the use of them, even in a loss of off-site-- a total loss of power to that division. You have the use of those relief trains for at least two hours.

MEMBER RAY: Okay, but still the answer remains, you are relying on all four, subject to some inoperability, but limited in time.

MR. PARECE: Correct.

MR. SALM There are scenarios where we assume that the failures of the MSRT either to open or to close.

MEMBER RAY: You have to take one train out of service to do surveillance testing, for example. I'm sure you must have some kind of online surveillance of the actuation system.

MR. PARECE: The surveillance testing for that system is done separately. It is powered by solenoids valves that are powered from different divisions that -- or pilot valves that equalize the pressure across the main valve.

So we do surveillance on each one of those, and it is out of service for such a short time that it is governed by the AOT.

MEMBER RAY: Right. That is all I was saying.

MR. SALM: Any other questions? All right.

This next slide shows the pressure, primary system and secondary system pressure response for a spectrum of LOCA. On the lefthand side there's bars that show the degraded heads of the MHSI, the accumulators and the LHSI. So you can see where they are going to discharge.

We look at the secondary side pressure. That is the dark blue line. Pressure starts out 1120-1130 psi range. When there is reactor scram, typically we assume turbine trip, and it closes off the secondary side. We don't take credit for the main steam bypass.

So the secondary side pressure rises quickly to the normal setpoint for the MSRT. It stays there until you get an SI signal. When you get an SI signal, it starts the automatic depressurization at 180 degrees per hour until it gets down to 870 psi where it holds the pressure until the operator initiates a 90 degree per hour cooldown.

We look at the primary system response. If we start out with a double-ended guillotine break, it is large enough that it quickly depressurizes the primary system down to low pressure. No surprise there.

As the break gets smaller, the rate of depressurization decreases. If we look at the four-inch break, that is roughly the size of a break that is capable of depressurizing the primary system without the secondary side.

If you get above that -- or excuse me, smaller than that break size, then the break is too small to depressurize the primary system. It relies on the secondary side to provide some amount of energy removal, and so the primary system gets pulled down by the secondary side as it is depressurized.

If you have a smaller break, such as a one-inch break, now the capacity of the MHSI is sufficient to hold the primary system pressure up, even though the temperature of the primary system is being brought down with the secondary side. Questions? Next, please.

Okay. Talk a little bit about the methodologies, starting out with the non-LOCA methodology.

The AREVA methodology is defined in EMF-2310. That was approved in 2004. The methodology used for the EPR is very similar to that except that the COPERNIC code has replaced the RODEX2 code. COPERNIC is a newer code.

We use S-RELAP5, which is a derivative of our RELAP5 MOD2.5, for the primary system response. We use LYNXT, which is a derivative of COBRA, for doing the core analysis DNB, and for the EPR we use the in-core trip, and there has been a special methodology developed for that. That is currently being reviewed by the NRC.

The main steam line break is a special methodology, and I will talk a little bit more about that later on.

Then rod ejection, Jonathan is going to

talk about later on in the closed session. It is a revised methodology. It was revised to implement the guidance of SRP 4.2. It couples the neutronics and hydraulic to predict the core response and, again, there is a topical report that is being reviewed by the NRC right now on that, and Jonathan will talk more about that. Next page.

As you heard already, we use a realistic statistical methodology for evaluation large break LOCA. This is the Realistic LOCA methodology. that uses RODEX3A to do the fuel analysis, S-RELAP5 to do the plant analysis. It has ICECON, which is a derivative of CONTEMPT, to do the concurrent containment analysis. They are explicitly coupled, and that methodology is described in EMF-2103, which was approved in 2003.

Small break LOCA methodology is a deterministic Appendix K methodology. It uses RODEX2. Portions of the RODEX2 code were incorporated in S-RELAP5, so it could do the hot pin calculation during the plant analysis, and that is described in EMF-2328, which was approved in 2001. Next slide.

Fuel analysis methodologies -- this is

Jonathan's area: That is described in EMF-96029, which was approved in 1997. It uses MICBURN/CASMO-PRISM, and I will let him answer any questions about that. And NEMO-K for the kinetics uses COPERNIC for the fuel responses and LYNXT for the core hydraulics and DNB.

Any questions? I will talk more about that in the closed session.

All right. Next page. These methodologies have been used for a variety of operating plants. You can see the list. They are little changed for EPR.

MEMBER MAYNARD: A question: These have been approved for other applications. You are doing the work now to show the applicability for the US EPR.

MR. SALM: Correct.

MEMBER MAYNARD: Is that being done within the DCD review or are you submitting separate topical reports?

MR. SALM: In most cases, we have submitted separate topical reports, and some of them have already been approved. Some of them are still being reviewed, and it is really going concurrently with the FSAR review.

MEMBER MAYNARD: Okay. This is something for the Committee. If the topical reports are being reviewed by the staff, unless we ask for them, we may or may not see those reports before we are asked to make a decision on the EPR from the DCD reviews.

MS. SLOAN: Just to be specific, we submitted a code applicability topical report addressing non-LOCA and small break LOCA, and that has been approved and an SER has already been issued by the staff, and the RLBLOCA topical report for application to EPR is currently under review by the staff as part of the design certification.

MR. SALM: Next slide. This slides shows the reports. The first one, ANP-10263, shows the applicability for the non-LOCA events and the fuel codes. 10278 is the applicability of the Realistic LOCA methodology, and there is a supplement, ANP-10291, that provides more information on the small break methodology.

The first one has been approved. The second one is being reviewed right now, and the third one is actually just a technical report to support the FSAR.

These reports take each event, and in many cases each phase of the event, and describe it, identify the important phenomena, identify the important components and functionality during that phase of the event, and provide an assessment of why these methodologies are applicable to the EPR and why the methodologies can be used. So it breaks it down in detail, phase by phase.

In the closed session, we will see more of what is reviewed in these topical and technical reports. Next page.

So really in summary, the AREVA methodologies are mature. The rod ejection methodology was updated to address a change in the SRP, but these are mature methodologies that have been applied for years to operating plants.

The EPR design, while it has some special features, is basically a four-loop PWR. It is very similar in operating conditions. It has the same phenomena, and the events, the phenomena, remain in the range of applicability for the constituent of models that are already used in the codes.

So the methodologies are applicable to the EPR where --

CHAIRMAN POWERS: You say it has the same phenomenon methodology -- I mean it has the same phenomena as occurring in a four-loop PWR. How do you know that?

MR. SALM: Well, I mean in the topical reports we go down in each phase --

CHAIRMAN POWERS: That says the code thinks it has the same phenomena.

MR. SALM: We will talk more about that in the closed session, you know, be more specific about which phenomena and which components.

CHAIRMAN POWERS: But we don't really have any experimental verification of your statement.

MR. SALM: Well, we have the experiments that have been used to validate the methods for operating plants, and where there is unique phenomena like the hot leg injection, we have gone back to the tests and looked specifically for those phenomena and justified the methodologies. But certainly, most of what is going to occur for this plant would be exactly the same as for current plants.

CHAIRMAN POWERS: I am just trying to understand why you think that is true.

MR. SALM: Well, we've looked at the differences in the design. We look at the tests and use engineering judgment. In the closed session, we will talk a little bit more about the level of detail that we went to.

When we run the analyses, it produces the results that one would expect. There aren't any surprises.

CHAIRMAN POWERS: It's just the code thing. I mean, if something happens between 4400 megawatts thermal and 4500 megawatts thermal that changes some physics computer code, you will never know about it unless you put it in.

MR. SALM: Well, I mean, how do you mean that? that it doesn't have the degree of resolution to --

CHAIRMAN POWERS: If there is a new physical phenomenon that shows up. The fuel guys know about this, because between 40 gigawatt days per ton and 65 gigawatt days per ton, a rim shows up. The fuel codes never predicted that. They do now, but they didn't.

MR. WITTER: This is Jonathan Witter. I think one response to kind of gain some confidence is the fact that the plant is really not all that different. If you look at the linear heat rates of the fuel rods, look at the core, the power-up scaling of the plant aspects of the computer codes that are general in nature of their fundamental basis, and as Bob mentioned, looking back at the test data and making sure that it does make sense that the scaling aspects of the plant relative to --

CHAIRMAN POWERS: We said the same thing about the fuel codes when we went from 17 to 34, when we were at 34 to 40. We said it again from 40 to 60 -- ah, all the physics is in there, and we don't need to worry. And new physics appeared. Now we are smarter, and we say it about going from 60 to 75, and we are just as wrong there as we were back at 17.

MR. WITTER: Well, I guess I don't really know exactly how to respond.

CHAIRMAN POWERS: Well, there is no real response. Until you've done an experiment, there is really never a response to that question.

MR. SALM: All right. Well, that is all I have for the open session.

CHAIRMAN POWERS: Okay. We are going to switch over into closed session.

(Whereupon, the foregoing matter proceeded to closed session at 1:52 p.m.)

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U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

AREVA EPR Design Certification Application Review Status

Getachew Tesfaye
Project Manager

September 9, 2009

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	June 30, 2010
Phase 3 – Advisory Committee on Reactor Safeguards (ACRS) Review of SER with Open Items	September 28, 2010
Phase 4 - Advanced SER with No Open Items	April 2011
Phase 5 - ACRS Review of Advanced SER with No Open Items	July 2011
Phase 6 – Final SER with No Open Items	September 2011
Rulemaking	February 2012

Phase 3 Review Plan

November 2009	February 2010	May 2010
Group 1 **(2-days - 11/18 & 19)** AREVA Intro ***Chap 2*** – Site Characteristics (*6/9/09*) ***Chap 8*** – Electric Power (*6/16/09*) ***Chap 10*** – Steam and Power Conversion (*8/24/09*) ***Chap 12*** – Radiation Protection (*7/23/09*)	**Group 2A** **(2 days - 02/02 & 03)** ***Chap 4*** – Reactor (*11/9/09*) ***Chap 5*** – Reactor Coolant and Connected Systems (*10/26/09*) ***Chap 16*** – Tech Specs (*11/25/09*) ***Chap 17*** – Quality Assurance (*9/11/09*) **March 2010** **Group 2B** **(2 days)** ***Chap 11*** – Radwaste Management (*7/13/09*) ***Chap 19*** – Severe Accidents / PRA (*9/11/09*)	**Group 3 (2 days)** ***Chap 3*** – Design of Structures, Components and Equipment (*3/25/10*) ***Chap 7*** – I & C Systems (*3/25/10*) ***Chap 9*** – Auxiliary Systems (*11/9/09*) ***Chap 18*** – Human Factors (*8/13/09*)

Phase 3 Review Plan

July 2010	September 2010
Group 4 (3 days) ***Chap 1*** – General Plant Description (*6/30/10*) ***Chap 6*** – Engineered Safety Features (*5/12/10*) ***Chap 13*** – Conduct of Ops (*8/24/09*) ***Chap 14*** – Initial Test Program (*5/12/10*) ***Chap 15*** – Safety Analysis (*5/12/10*)	**Summary (1 day)** Summation of Open Items, Cross-cutting Issues and Re-visit Earlier Chapters

U.S. EPR Containment Design and Analysis and U.S. EPR Analysis Methodologies

U.S. EPR Subcommittee
Advisory Committee on Reactor Safeguards (ACRS)
9 September 2009

Presentation Goal

To provide background information on two key topic areas to support ACRS review of the Safety Evaluation Report for the U.S. EPR design certification application

Presentation Topics

- **U.S. EPR containment design and analysis**
  - **OPEN session**
    - Overview of containment features
    - Review of containment response to postulated pipe ruptures
    - Analytical methodology summary
  - **CLOSED session**
    - EMDAP
    - Evaluation model
    - Mass and energy release (RELAP5-BW)
    - Containment pressure and temperature (GOTHIC)
    - Benchmarks
    - Limiting large break loss-of-coolant accident results

Presentation Topics

- **U.S. EPR analysis methodologies**
  - **Focus on safety analysis methodologies**
  - **OPEN session**
    - Design overview
    - Selection of methodologies
    - Overview of non-LOCA methodology
    - Overview of Main Steam Line Break methodology
    - Overview of fuel analysis methodologies
  - **CLOSED session**
    - Control Rod Ejection
    - Large Break LOCA
    - Small Break LOCA

Acronymns

Acronym	Description
ACCU	
AMS	Aeroball Measurement System
BC	Boundary Conditions
BFMC	Battelle-Frankfurt Model Containment
BOC	Beginning of cycle
CCFL	Counter Current Flow Limit
CCTF	Cylindrical Core Test Facility
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CSAU	Code Scaling, Applicability, and Uncertainty
CVCS	Chemical and Volume Control System
DC	Design Certification
DCD	Design Control document
(M)DNB(R)	(Minimum) Departure for Nucleate Boiling (Ratio)
DTC	Doppler Temperature Coefficient of Reactivity
EBS	Extra Borating System
ECC(S)	Emergency Core Cooling (System)
EDG	Emergency Diesel Generator
EFW(S)	Emergency Feedwater (System)
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
EOC	End of Cycle
EPR	Evolutionary Power Reactor
ERW	Ejected Rod Worth
FOP	Fraction of Power
$F_{\Delta H}$	Maximum Relative Rod Power, Axially Integrated Enthalpy Rise
F_Q	Peak Relative pellet Power
F_Z	Maximum Relative Axial Power Shape Peaking Factor
HDR	HeissDampf Reaktor
HFP	Hot Full Power
HZP	Hot Zero Power
IRWST	In-Containment Refueling Water Storage Tank
ISP	
LCO	Limiting Conditions for Operation
LOOP	Loss of Offsite Power
MHSI	Medium Head Safety Injection
MSRT	Main Steam Relief Train
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient of Reactivity
NSSS	Nuclear Steam Supply System
PCM	Percent Milli-rho of Reactivity (10^{-5} $\Delta\rho/\rho$)
PCMI	Pellete Clad Mechanical Interaction
PCT	Peak Clad Temperature
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor

Acronyms (Continued)

Acronym	Description
RCP	Reactor Coolant Pump
RCSL	Reactor Control Surveillance Limitation
REA	Rod Ejection Accident
RHR(S)	Residual Heat Removal (System)
RIA	Reactivity Initiated Accident
SAFDL	Specified Acceptable Fuel Design Limit
SAHRS	Severe Accident Heat Removal System
SCTF	Slab Core Test Facility
SRP	Standard Review Plan
TFGR	Transient Fission Gas Release
T-H	Thermal Hydraulic
UPTF	Upper Plenum Test Facility

U.S. EPR Containment Design and Analysis

U.S. EPR Subcommittee
Advisory Committee on Reactor Safeguards (ACRS)
9 September 2009

Agenda

- **Containment Design Overview**
  - **Specific Features**
  - **Response to Postulated Pipe Ruptures**
  - **Layout**
- **Summary of Evaluation Methodology**

U.S. EPR^{TM} Containment Design and Analysis

Martin V. Parece
Vice President, Technology
Rockville, MD
September 9, 2009

Agenda

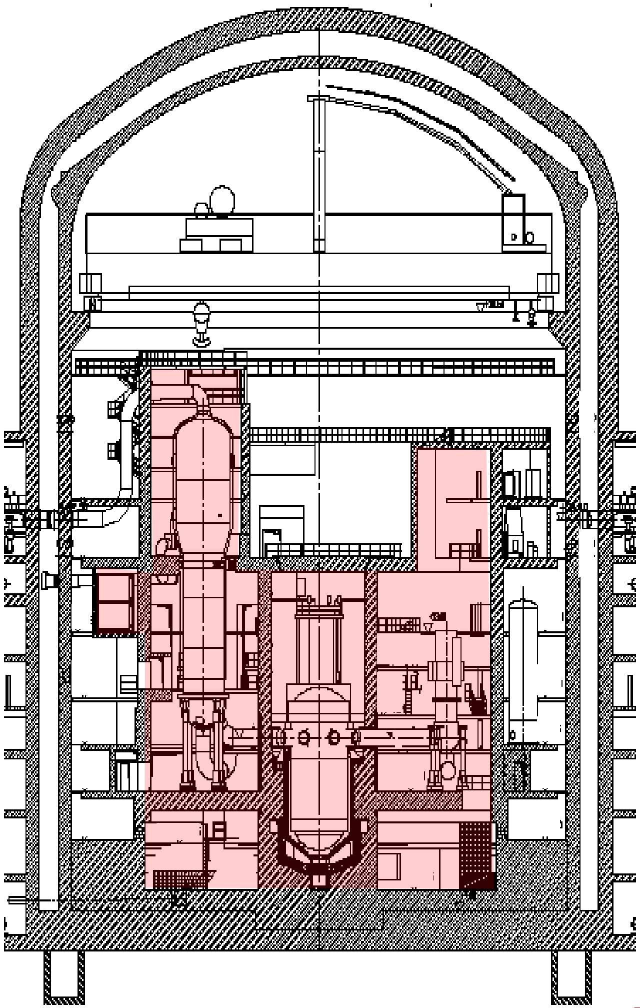
- Containment Design Overview
  - Specific Features
  - Response to Postulated Pipe Ruptures
  - Layout
- Summary of Evaluation Methodology

--------- Begin Proprietary Session ---------

- Mass & Energy Methodology
  - RELAP5/MOD2-B&W transition to GOTHIC
  - Hot leg injection
  - Benchmarks
- Containment Modeling & Analysis
  - GOTHIC model objective
  - Benchmarks
- Sample Problem

EPR Reactor Building

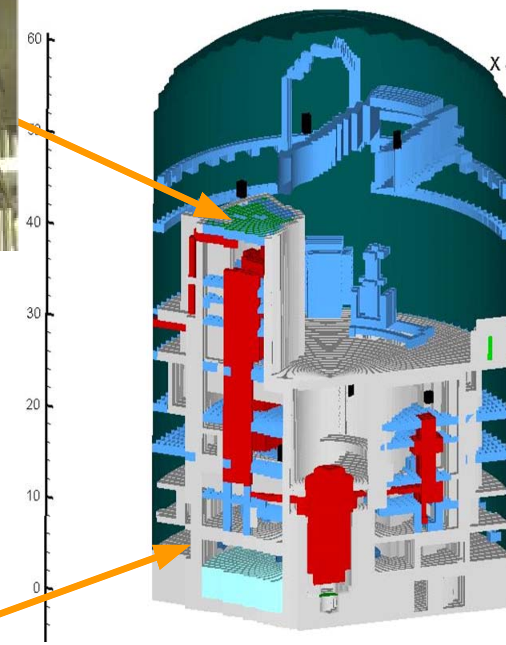
- Post-tensioned concrete containment with steel liner
- Shield Bldg wall reinforced concrete
- Containment Free Volume = 2.8 Mft^3
- Containment Inside Diameter = 153.5 ft.
- Containment Wall Thickness = 4.3 ft.
- Design pressure = 62 psig
- In-Containment Refueling Water Storage Tank (~500,000 gal)
- Design leak-rate at design pressure is less than 0.25 percent by volume
- Two-zone containment
- CONVECT system of rupture and convection foils and dampers connect zones during LOCA
- Passive hydrogen reduction system
- Severe accident mitigation features



CONVECT System – Main Components

- **Level +31.40 m: Atmosphere Release Section**
  - **Convection Foils in the Pressure Equalization Ceiling**
  - **Rupture Foils in the Pressure Equalization Ceiling**

CONVECT
Rupture & Convection Foils



- **Level -2.30 m: Atmosphere Inlet Section**
  - **Hydrogen Mixing Dampers**

CONVECT
Mixing Dampers

Convection Foil

Installed Convection Foil (Standby)

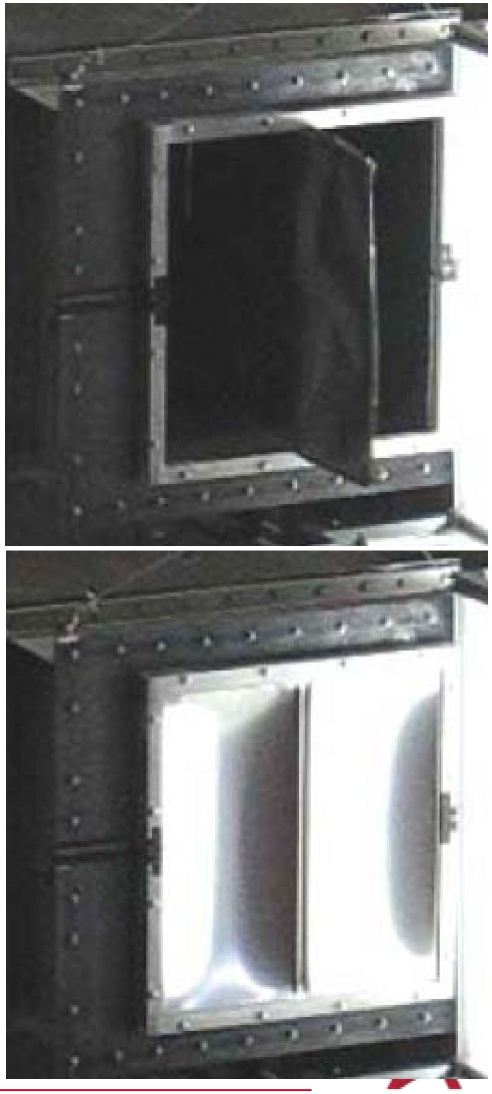
Convection Foil

▶ ***Opening Characteristics:***

- **Acts like a rupture foil for high pressure**
- **Elevated temperature opening**
- **No debris generation**
- **Bi-directional (on pressure)**

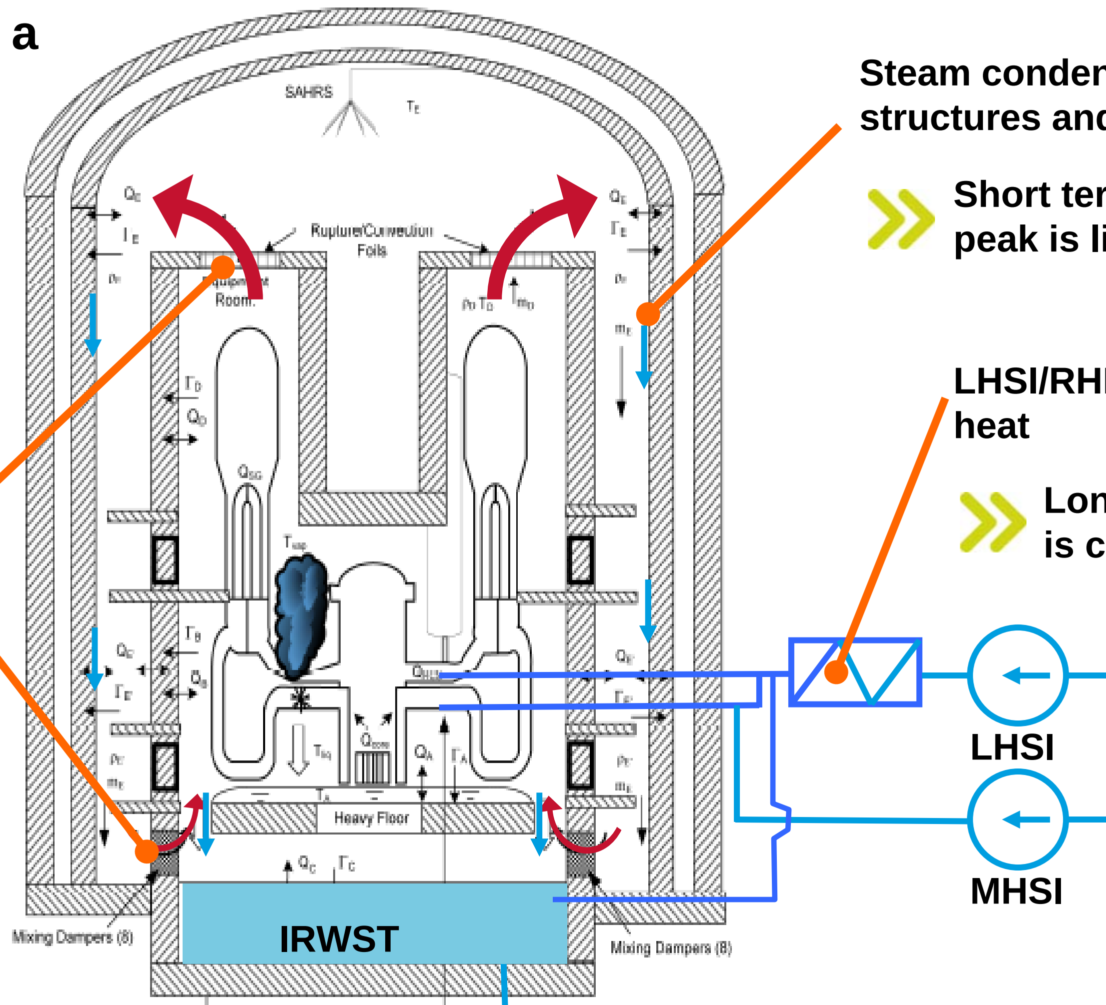
CONVECT System

- **Performance**
  - **Large breaks: all foils and dampers open**
  - **Smaller breaks: mixing relies essentially on convection foils. MAAP4 analyses addressing 10 CFR 50.44 show good mixing**
  - **No debris generated**
- **Component qualification and testing**
  - **Foils and Dampers provide redundancy and meet single failure criterion**
  - **Designed to meet Seismic I requirements**
  - **Part of EQ, Inspection, and Testing program**
  - **Qualification program to show proof of concept**



U.S. EPR Containment Design and Concept

- **Containment pressure mitigation after a LOCA or SLB**



Foils & Dampers opening allow steam convection

Steam condenses on the structures and returns to IRWST

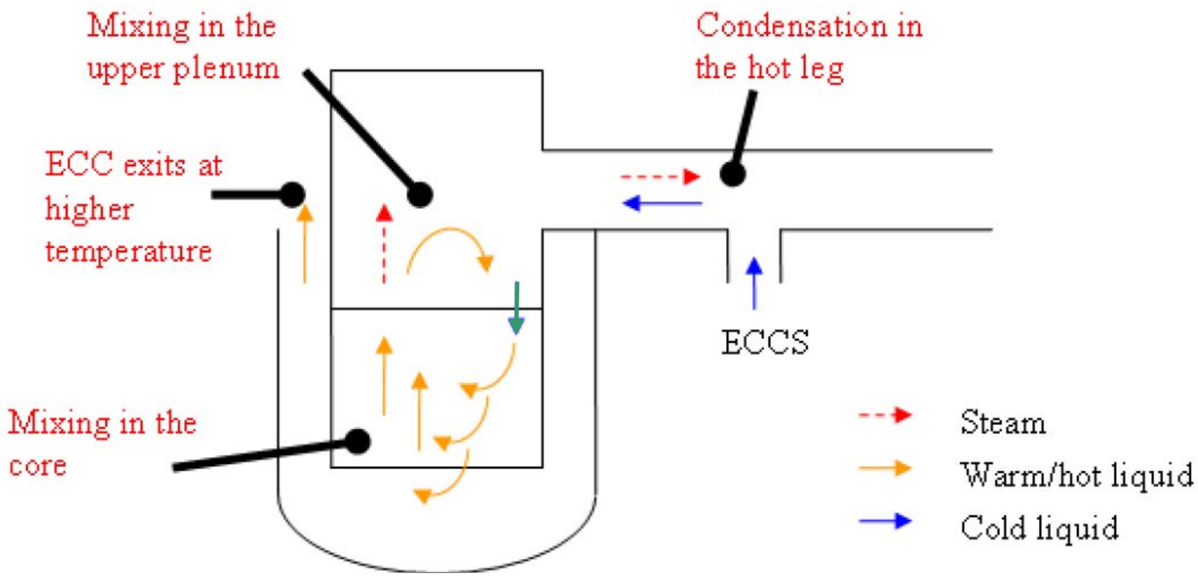
- **Short term pressure peak is limited**

LHSI/RHR HX removes heat

- **Long term pressure is controlled**

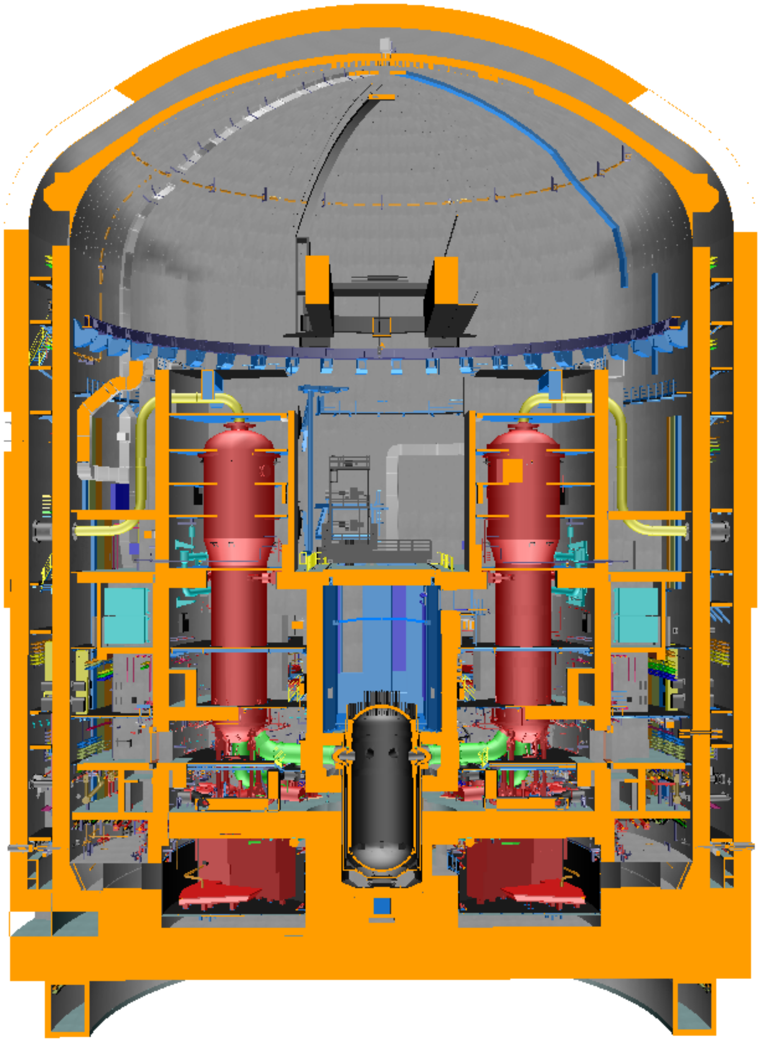
U.S. EPR Containment Design and Concept

- **Long Term suppression of steaming**
  - **Steam line break: Steam generator feed is manually isolated**
  - **LOCA: manual LHSI switch to hot leg injection at 60 minutes**

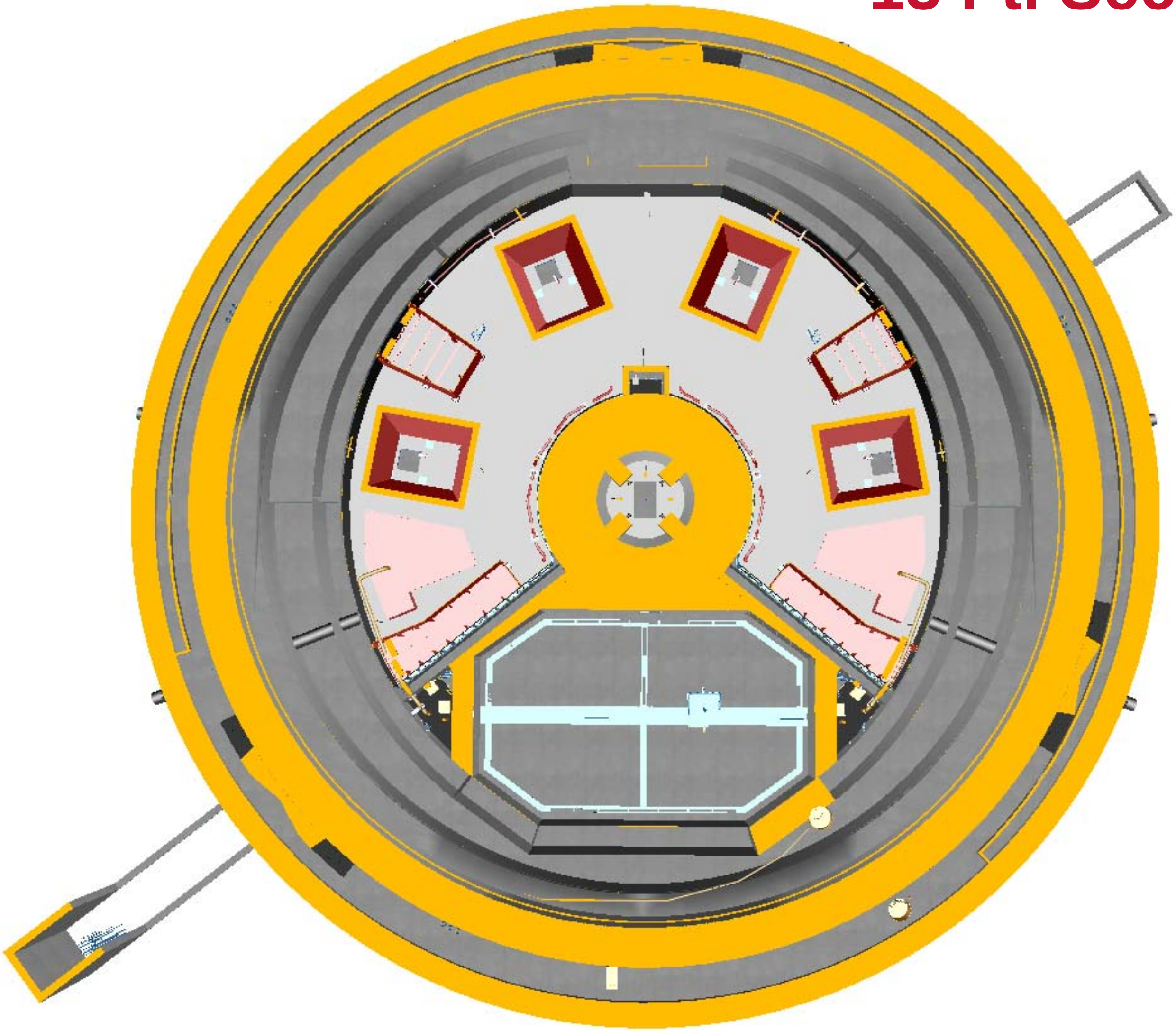


  - **As a back-up, non-safety related containment spray can also be used to condense the steam**

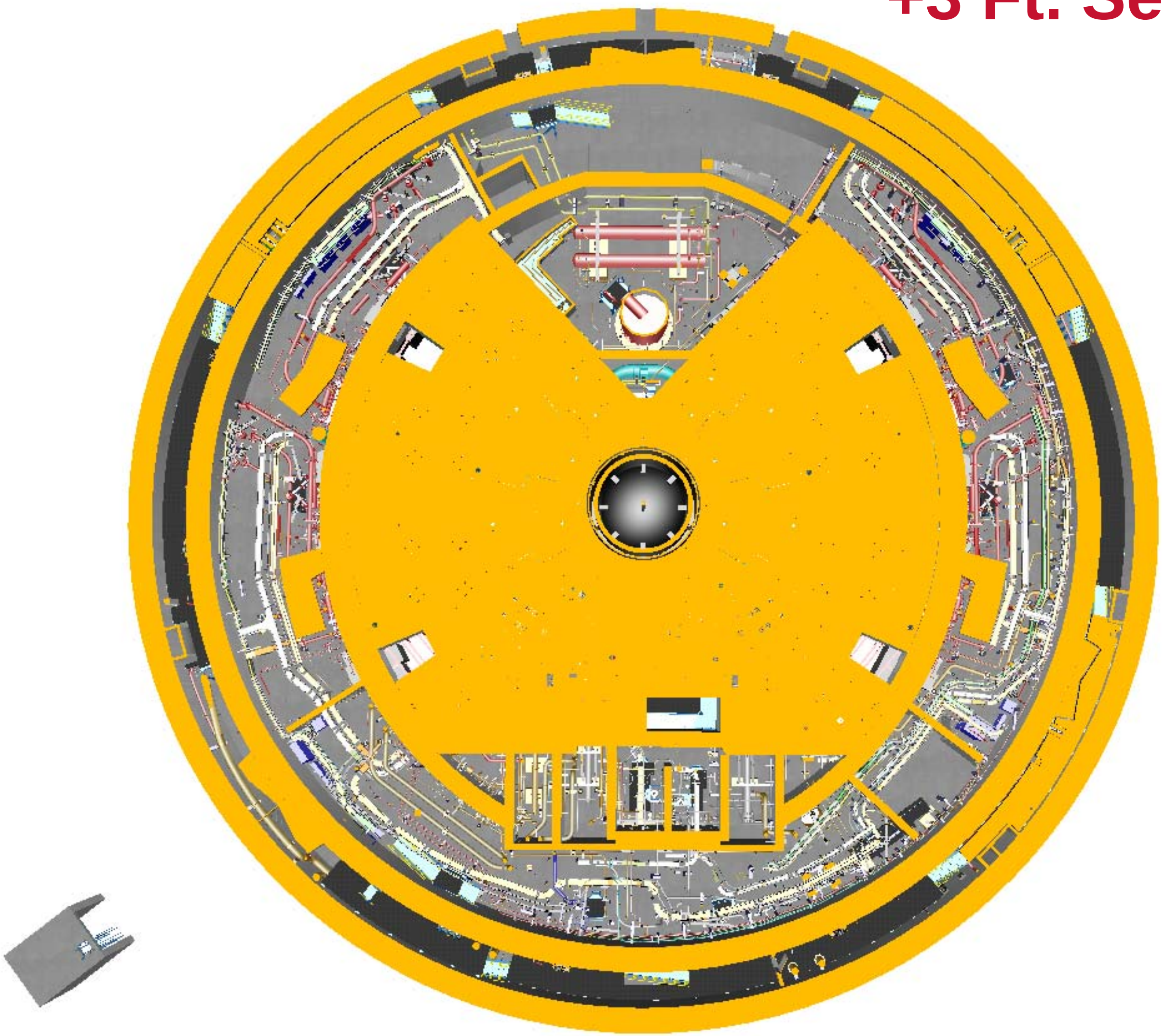
Reactor Building
Vertical Section



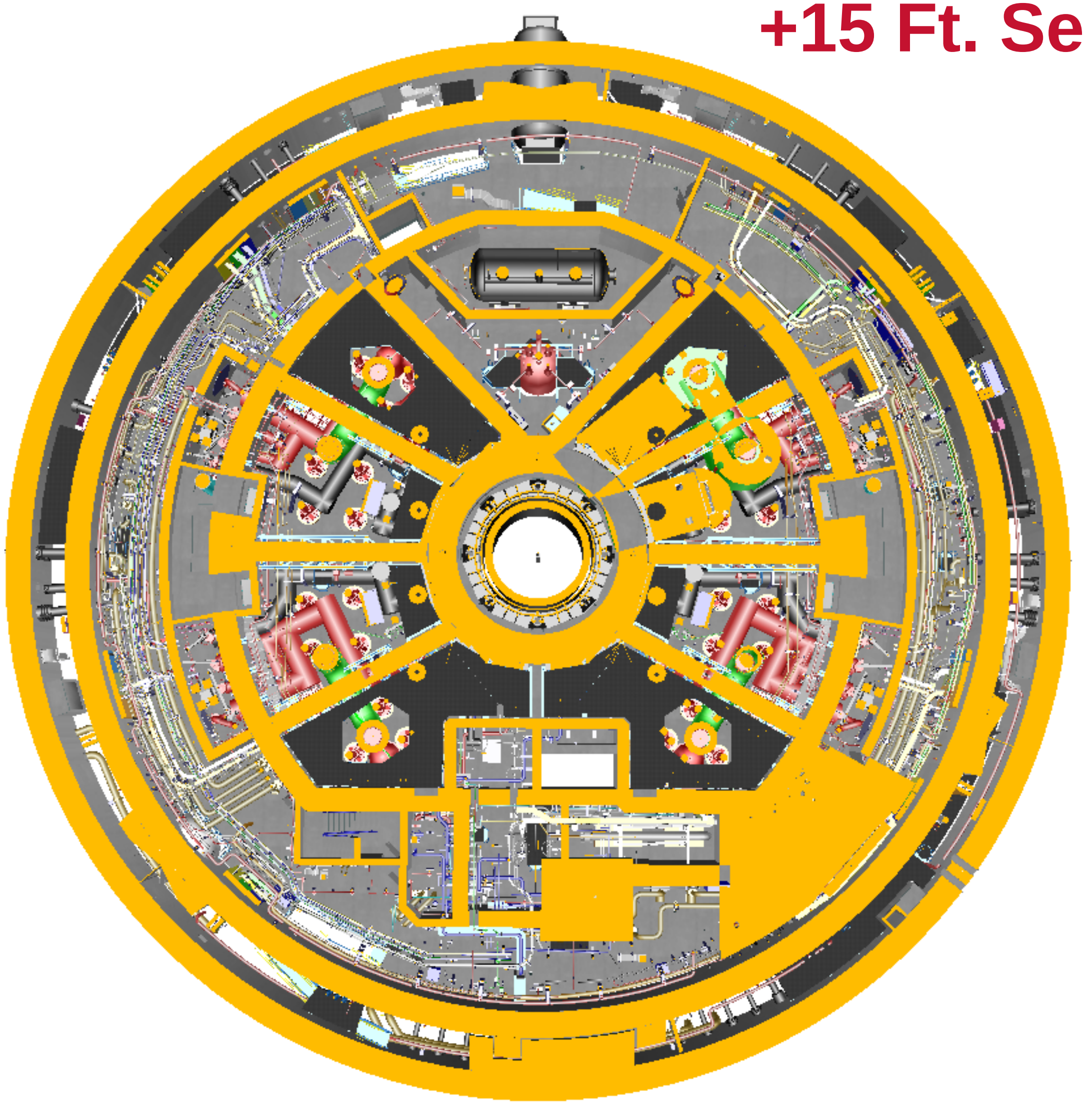
Reactor Building
-13 Ft. Section



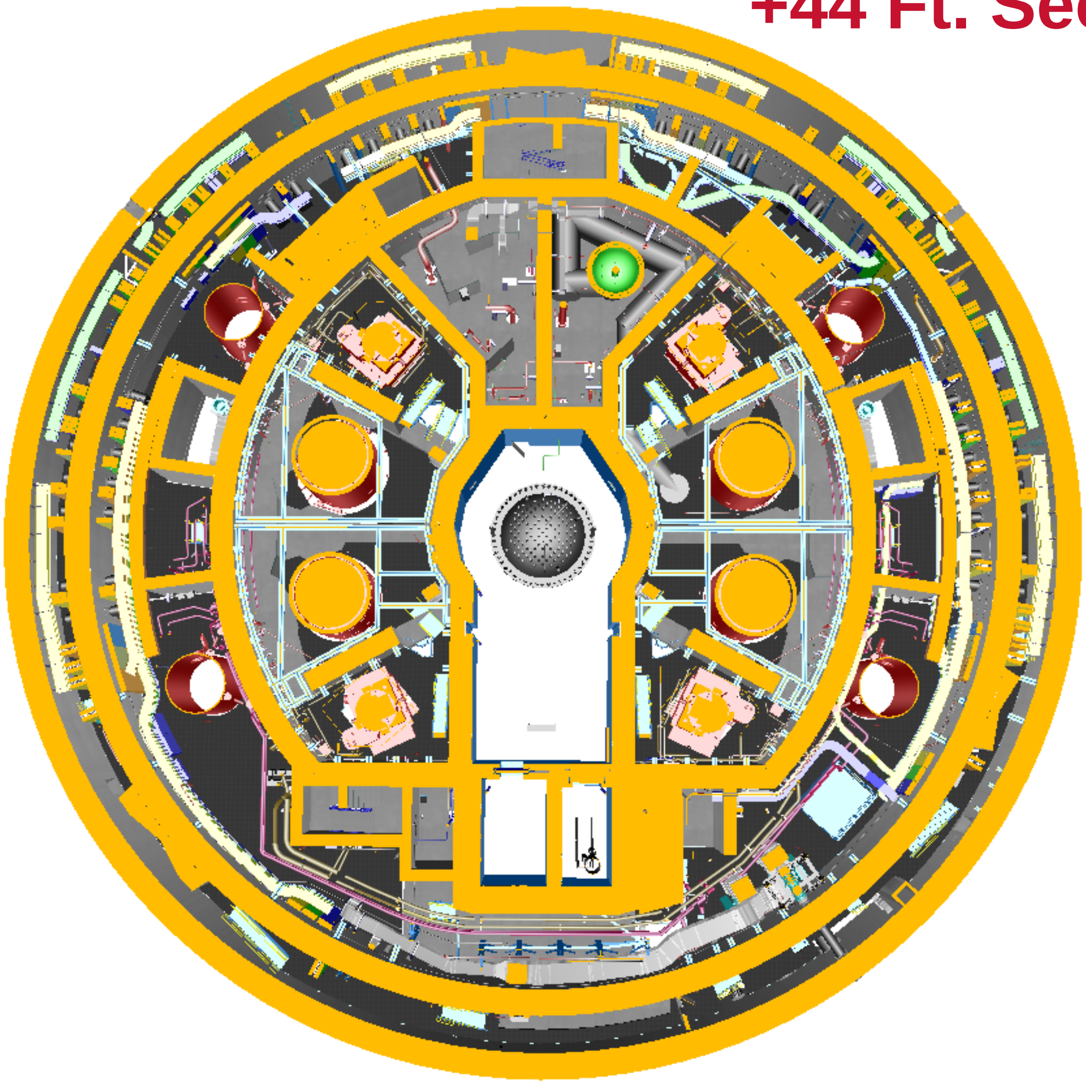
Reactor Building
+3 Ft. Section



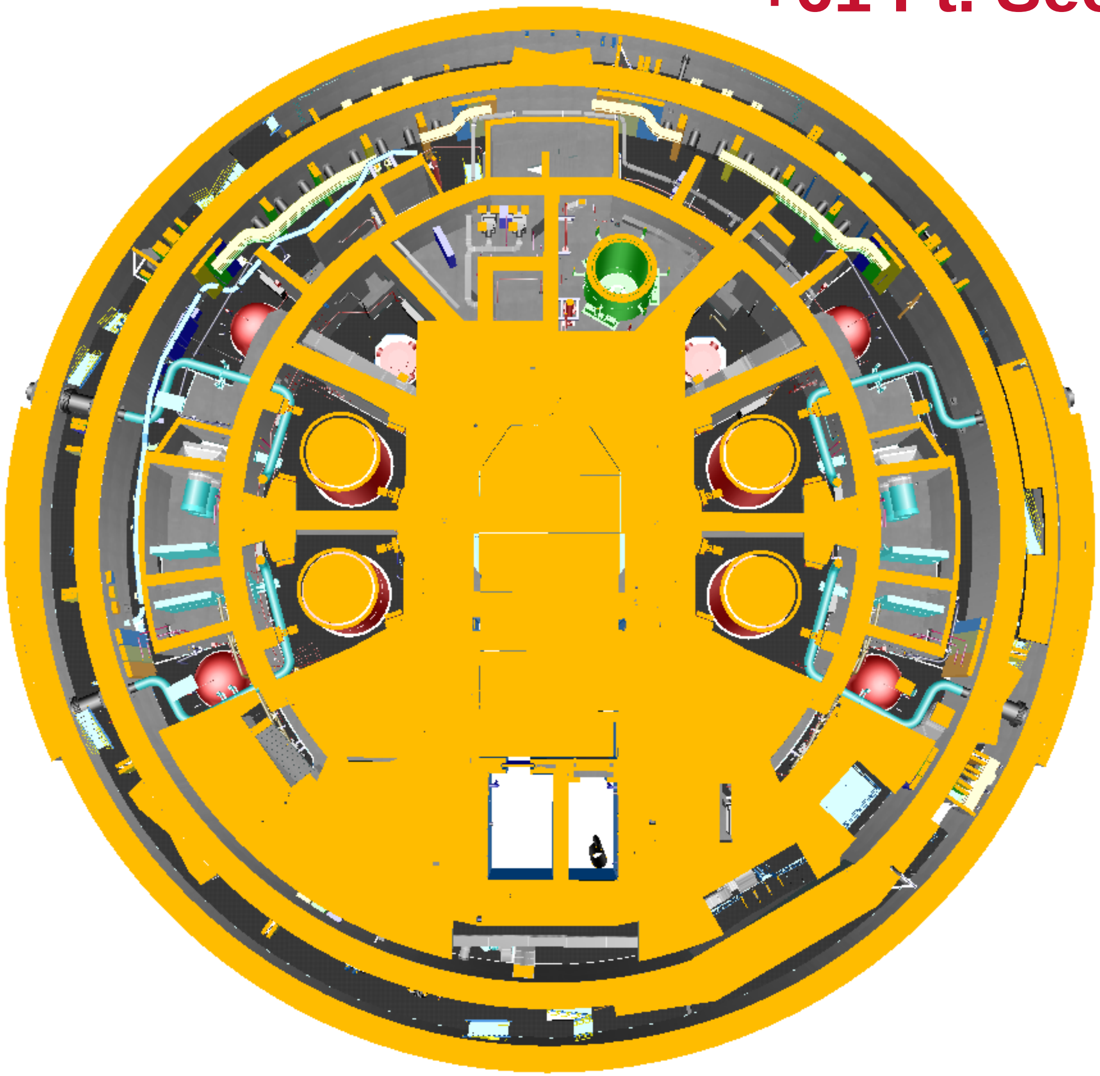
Reactor Building
+15 Ft. Section



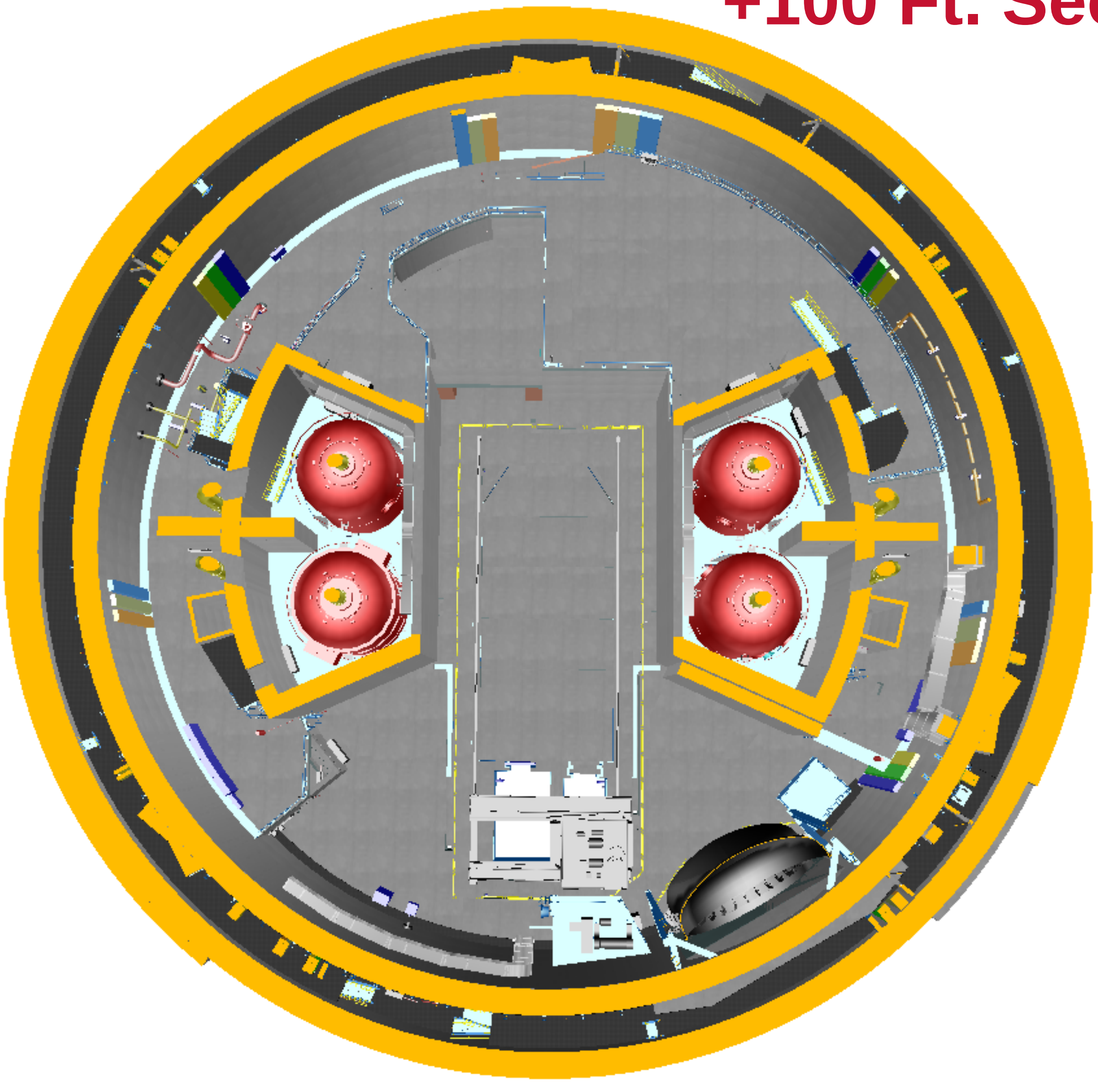
Reactor Building
+44 Ft. Section



Reactor Building
+61 Ft. Section

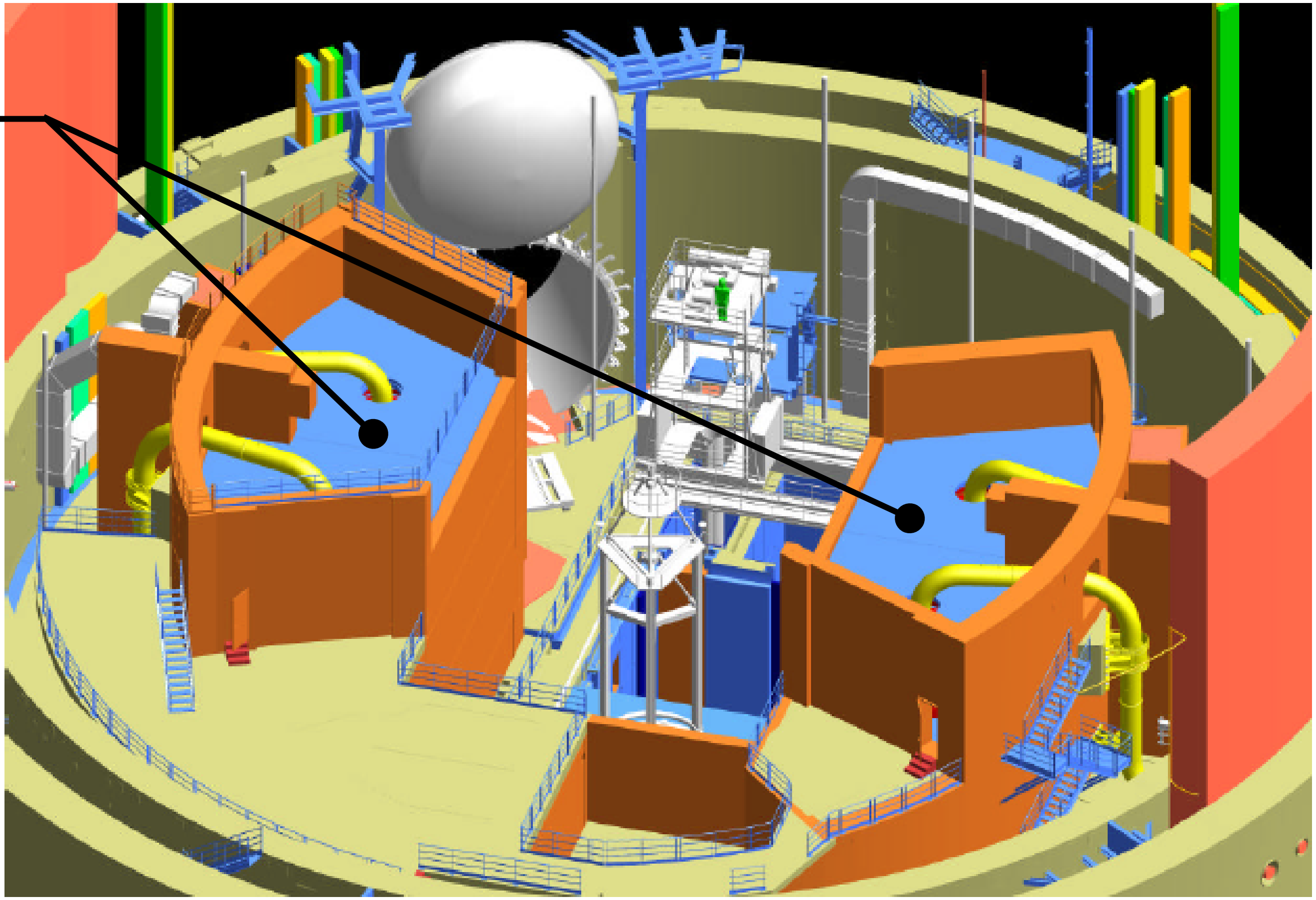


Reactor Building
+100 Ft. Section

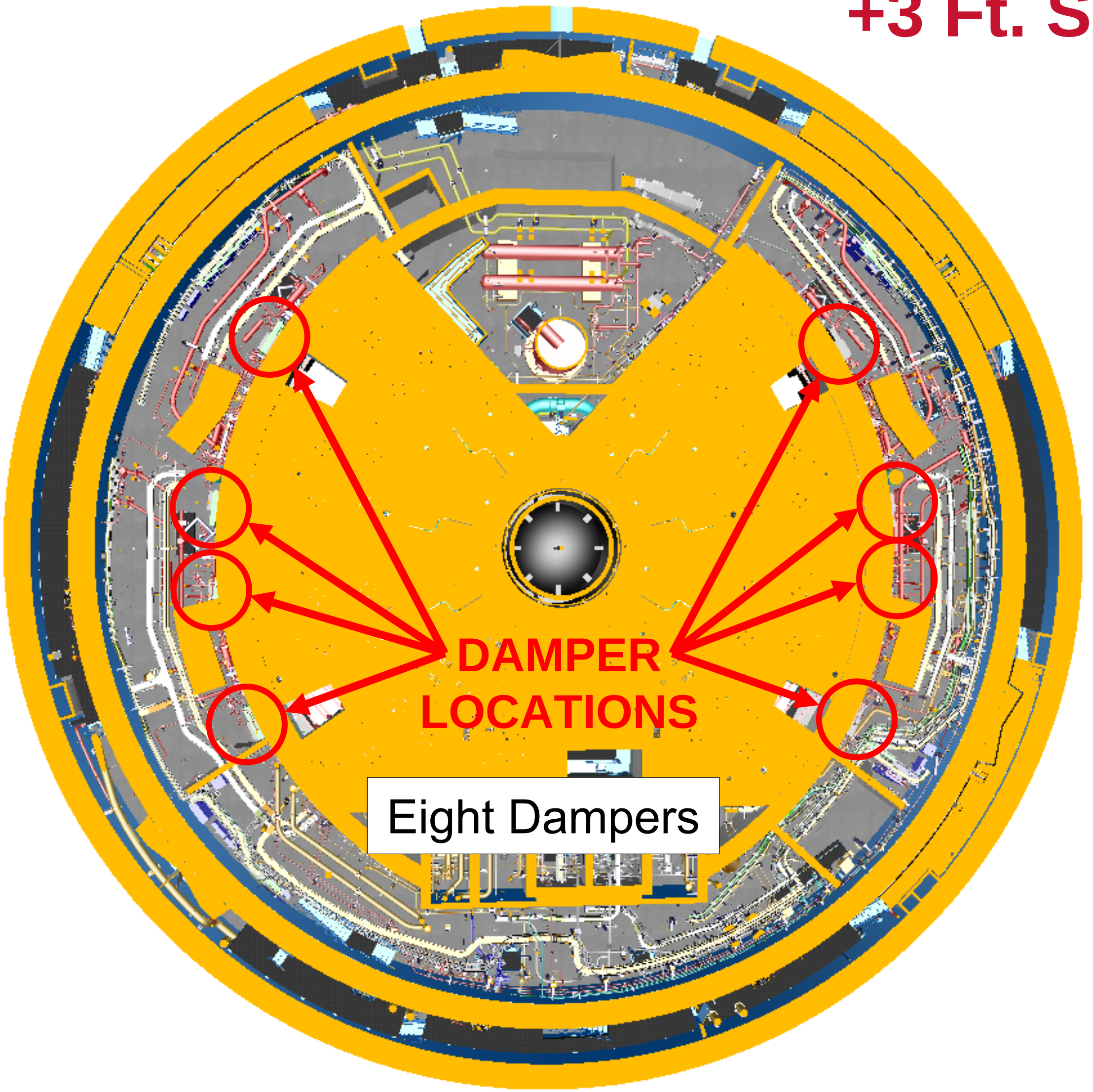


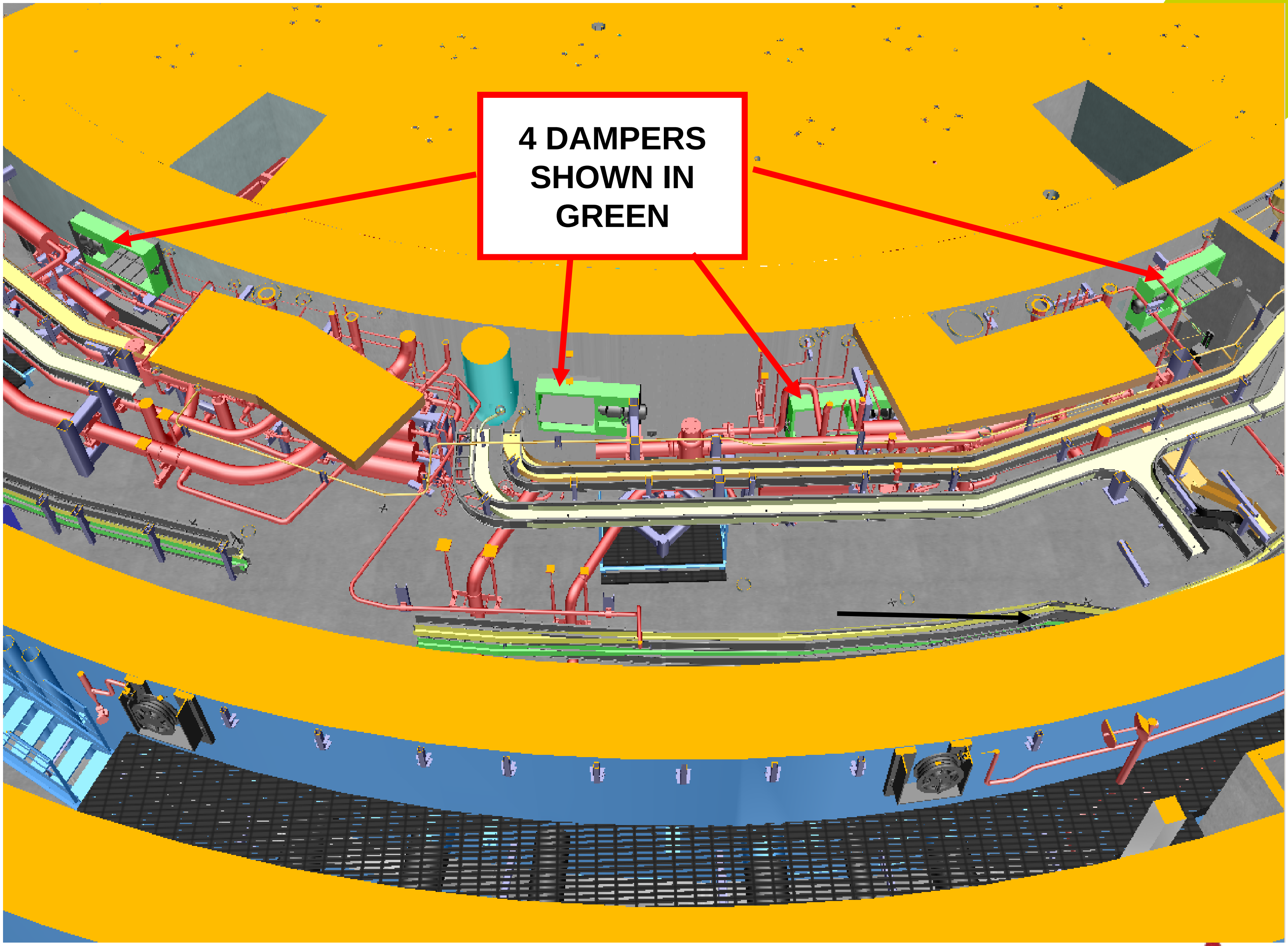
Inside Containment

Rupture and Convection Foils



Reactor Building
+3 Ft. Section




4 DAMPERS SHOWN IN GREEN

Containment Response Evaluation Methodology

- The methodology is described in Technical Report ANP-10299P
  - Applies RG 1.203 framework for Evaluation Model Development and Assessment Process (EMDAP)
  - Describes the M&E methodology (RELAP5-BW and GOTHIC)
  - Demonstrates applicability of GOTHIC methodology to U.S. EPR containment design
  - Describes the U.S. EPR containment design, including the CONVECT system, H_2 reduction system and severe accident heat removal system
  - Quantifies margin provided by conservatisms in EM

Containment Response Evaluation Methodology

- **Requirements**
  - **GDC 50 – Requires containment be designed to accommodate the pressure and temperature conditions following a LOCA**
  - **GDC 38 – Requires a containment heat removal system to rapidly reduce containment pressure and temperature following a LOCA**
    - SRP Section 6.2.1.1.A – Provides acceptance criterion that containment pressure be ≤ 50% of the peak pressure within 24 hours after accident
- **EMDAP Process and PIRT**
  - **Evaluation Model requirements are based on PIRT**
  - **U.S. EPR PIRT Identifies, ranks and assesses state of knowledge the important phenomena for**
    - Mass and Energy release evaluation
    - Containment pressure evaluation
  - **U.S. EPR PIRT is based on existing PIRT and U.S. EPR specific changes**

Containment Response Evaluation Methodology

- **Assessment Database and Scaling**
  - **Reviews previous test data and code assessments for RELAP5-BW and GOTHIC**
  - **Develops equations for scalability analysis to validate the adequacy of the benchmarks performed to validate GOTHIC**
- **Evaluation Model Adequacy**
  - **Assessment of RELAP5-BW and GOTHIC show that they predict medium- and high-ranked PIRT phenomena except:**
    - Multi-dimensional mixing in reactor vessel during post-reflood, hot leg injection phase
    - Interfacial heat transfer to IRWST liquid
  - **Methodology is adjusted to compensate for code limitations (conservative biases and analytical treatments)**

Containment Response Evaluation Methodology

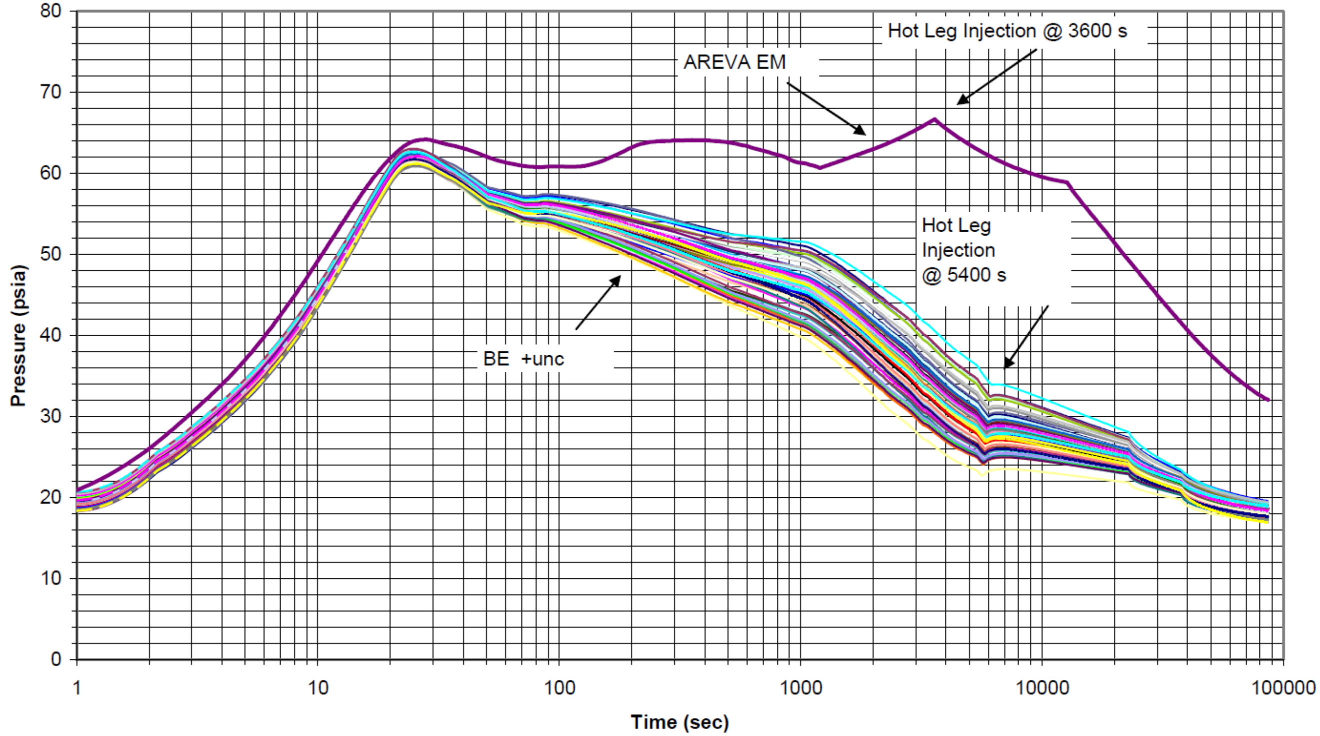
- **Validation and Sensitivity Analysis to demonstrate the applicability of RELAP5-BW and GOTHIC given the U.S. EPR specificities**
  - **RELAP5-BW assessments against FLECHT-SEASET data**
    - Core heat transfer and liquid carryout are well predicted
    - Heat transfer from secondary to primary is well predicted
  - **Hot leg injection mixing/condensation efficiency**
    - UPTF, CCTF, and SCTF test data support significant mixing in the core
    - Conservative mixing efficiency is validated for U.S. EPR configuration through CFD analysis with STAR-CD.
    - Hot leg injection suppresses long-term steaming to containment
  - **GOTHIC single- and multi-node models are assessed against HDR and BFMC integral-effects containment tests representative of the U.S. EPR design**
    - No sprays or fan coolers
    - Multi-compartment configuration (BFMC Biblis)
    - Short-term and long-term phenomena
  - **GASFLOW 3-D CFD code with U.S. EPR model shows good atmospheric mixing and convection in containment**

Containment Response Evaluation Methodology

- **Uncertainty analysis – follows Code Scaling, Applicability and Uncertainty (CSAU) methodology**
  - **Evaluates through sensitivity studies a range of values bounding the expected value of the parameter**
  - **Confirms the dominant phenomena identified in PIRT. Structure properties and condensation are the dominant phenomena for containment pressure**
- **Modeling and Regulatory Compliance**
  - **Methodology (includes codes, biases and treatments) is compliant with NUREG-0800 SRP and ANSI/ANS-56.4**
  - **Codes used:**
    - LOCA mass and energy release rates
      - Short-term – RELAP5-BW
      - Long-term – GOTHIC
    - GOTHIC with multi-node model predicts containment pressure and temperature response

Containment Response Evaluation Methodology

- Double-ended guillotine cold leg pump suction break sample case



The Evaluation Model is conservative

U.S. EPR Analysis Methodologies

Robert Salm
Manager,
New Plants Process Engineering

September 9, 2009

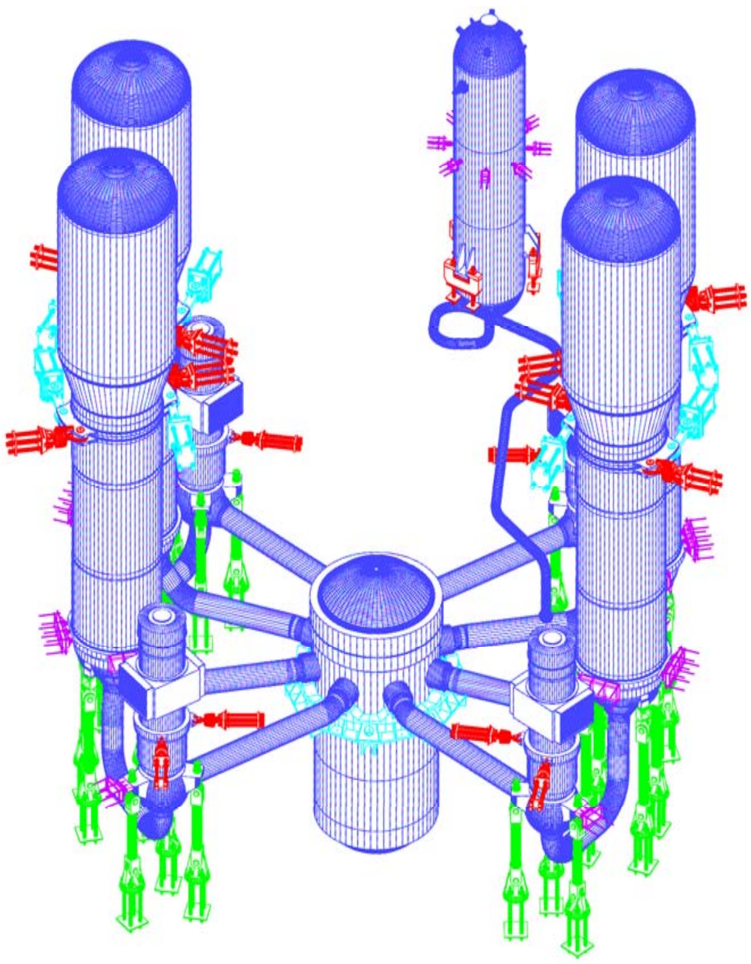
Presentation Topics

- Unique U.S. EPR design features important to safety analysis
- AREVA methodologies
  - Codes
  - Approved topical reports
  - Examples of application to operating plants
- Applicability to U.S. EPR
  - Supporting topical and technical reports
- Summary

Proprietary Session

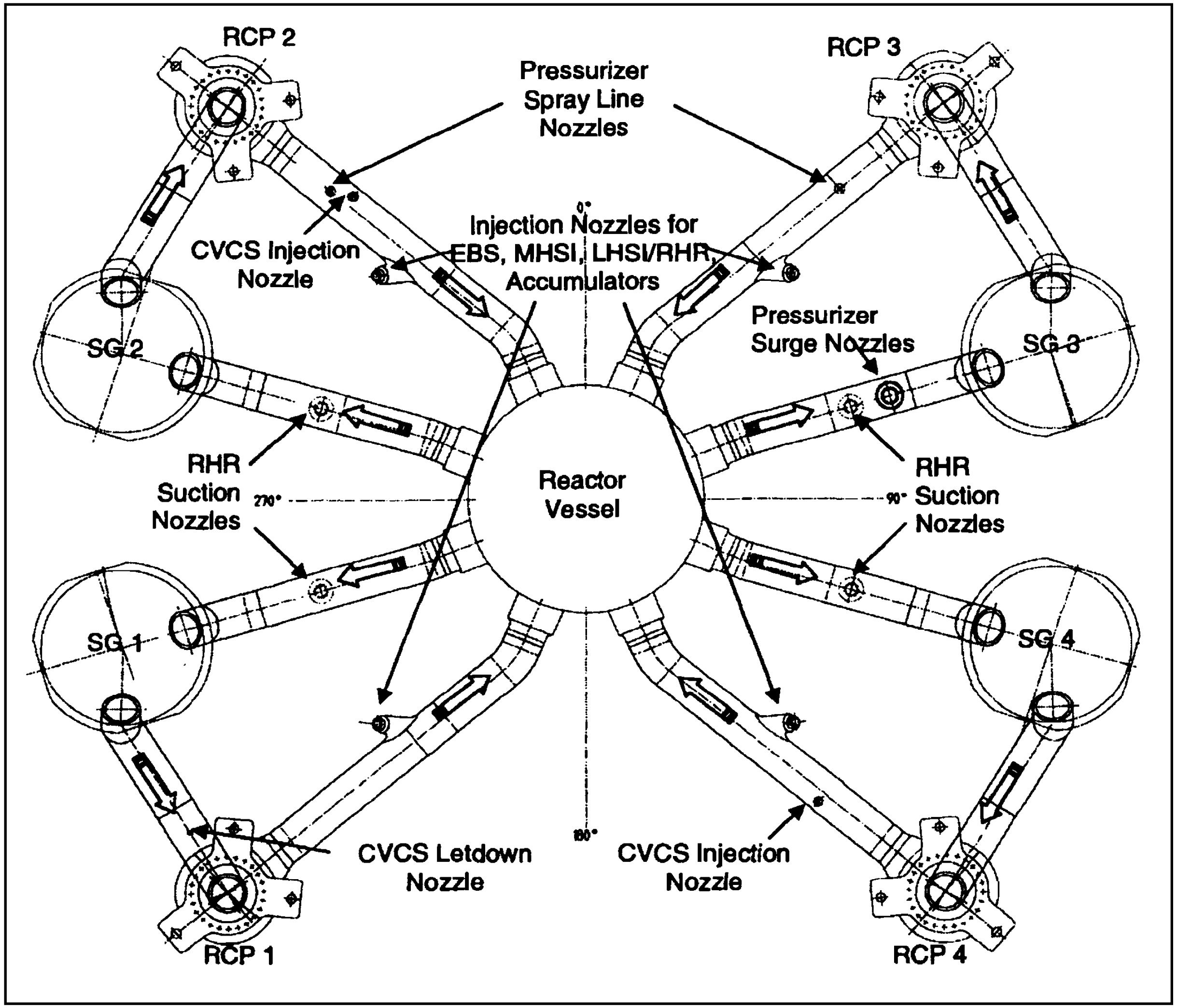
- Example methodologies
  - Rod Ejection
  - Realistic Large Break LOCA (RLBLOCA)
  - Small Break LOCA (SBLOCA)

Reactor Coolant System

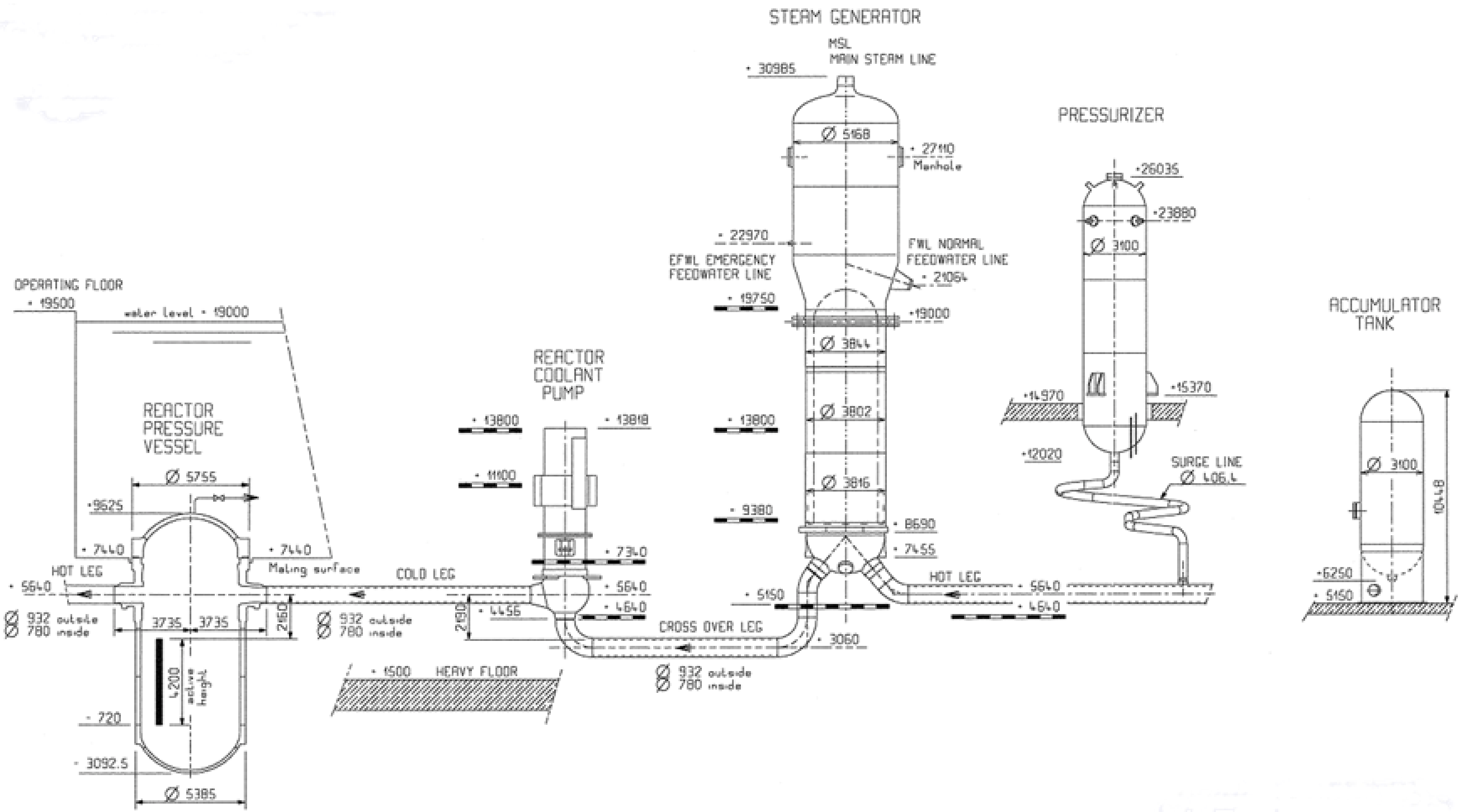


- Conventional 4-loop PWR design, proven by decades of design, licensing & operating experience

U.S. EPR RCS Layout



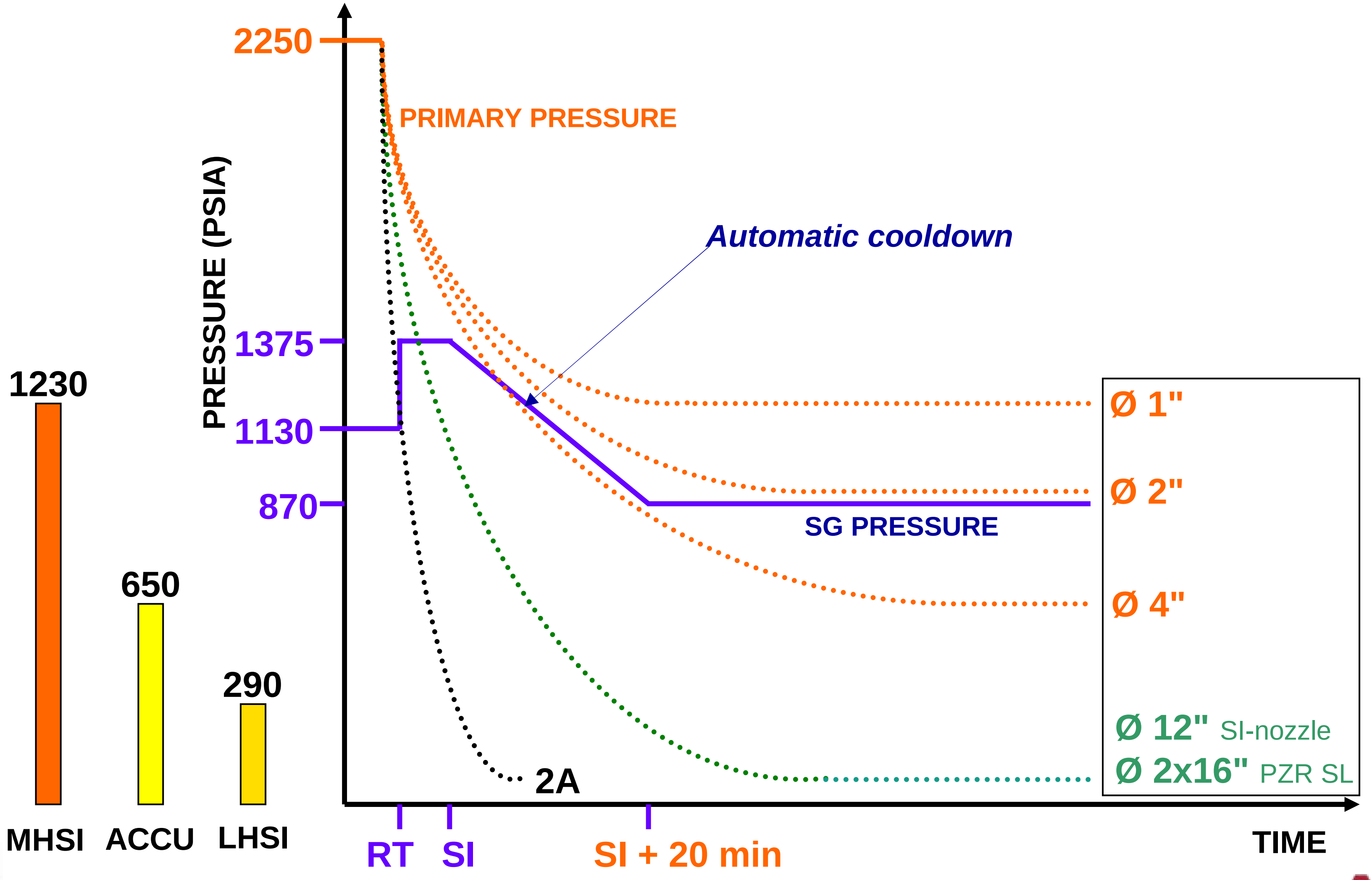
U.S. EPR RCS Elevation Drawing Showing Loop Seal



Unique U.S. EPR Safety Related Systems and Features

- **Four trains of Accumulators, Medium Head Safety Injection (MHSI), Low Head Safety Injection (LHSI)/Residual Heat Removal System (RHR), Emergency Feedwater (EFW), Main Steam Relief Train (MSRT)**
- **LHSI cross-connects opened when one LHSI train is removed for preventive maintenance**
- **Four-train, safety-related system (one per SG)**
- **Depressurizes SGs automatically on SI signal to reduce setpoint equivalent to 180°F/hr (T_{sat})**
- **Automatic partial cooldown of SGs on SI signal**
- **Automated trip of reactor coolant pumps on coincident SI actuation signal and low DP across the pumps**
- **Low DNBR and High LPD RT functions utilizing in-core measurements of local core power distributions at several locations within each core quadrant**
- **In-Containment Refueling Water Storage Tank (IRWST)**
  - **Source of ECC water**
  - **No switchover needed**
- **Extra Borating System (EBS)**

LOCA: RCS Pressure



Safety Analysis Methodologies

- **Non-LOCA**
  - COPERNIC replaced RODEX2 for determining initial fuel conditions
  - S-RELAP5 – RCS response
  - LYNXT replaced XCOBRA-IIIC for determining DNBR
  - PRISM used for neutronics portion of the calculation
  - "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," EMF-2310(P)(A), Revision 1, June 16, 2004
  - Applied Incore Trip Setpoint and Transient Methodologies, ANP-10287P (in NRC review)
  - Main Steam Line Break methodology is a special subset of Non-LOCA
- **Control rod ejection**
  - Implements acceptance criteria and guidance of March 2007 revision of SRP 4.2
  - Manually couples neutronic, plant system, and thermal hydraulic computer codes as necessary to predict performance and any fuel failures
  - "U.S. EPR Rod Ejection Accident Methodology Topical Report," ANP-10286P, November 2007 (in NRC review)

Safety Analysis Methodologies (Continued)

- **RLBLOCA**
  - **RODEX3A – computation of the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance**
  - **S-RELAP5 – system T/H calculations**
    - ICECON module used to determine containment backpressure
  - **"Realistic Large Break LOCA Methodology for Pressurized Water Reactors," EMF-2103(P)(A), Revision 0, April 2003**
- **SBLOCA**
  - **RODEX2 - initial fuel gap conductance**
  - **Portions of RODEX2 integrated into S-RELAP5 code for hot pin response**
  - **S-RELAP5 – RCS response**
  - **"PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," EMF-2328 (P)(A), Revision 0, March 2001**

Overview of Fuel Analysis Methodologies

- **Neutronics - Core Design and Neutronics Input to Safety**
  - **EMF-96-029(P)(A), "Reactor Analysis System for PWRs," Volumes 1 and 2, January 1997. (MICBURN/CASMO-3/PRISM)**
    - Benchmarking/validation calculations demonstrate applicability for use on U.S. EPR configurations
      - Adoption of thermal energy cutoff of 0.625 eV
      - Benchmarks to plants with aeroball measurement system
      - Characterizes and evaluates heavy reflector modeling methodology
  - **BAW-10221PA, Revision 0, "NEMO-K A Kinetics Solution in NEMO," October 1998.**
- **Thermo-Mechanical – Fuel/Fuel Rod Response**
  - **BAW-10231PA, Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004.**
- **Thermal Hydraulics – Core Hydraulics and DNB Analysis**
  - **BAW-10156A, Revision 1, "LYNXT Core Thermal-Hydraulic Program," August 1993.**

Examples of Operating Plants Licensed using AREVA Safety Analysis Methodologies

- **Non-LOCA**
  - Millstone
  - Robinson
  - Ft. Calhoun
  - St. Lucie
  - Palisades
- **SBLOCA**
  - Millstone
  - Robinson
  - Ft. Calhoun
  - St. Lucie
  - Harris
  - Palisades
- **RLBLOCA**
  - Robinson
  - Ft. Calhoun
  - Palisades
  - North Anna
  - Sequoyah

Applicability to U.S. EPR Design

- **The following reports demonstrate applicability to the U.S. EPR:**
  - **Non-LOCA, SBLOCA and Fuels - "Codes and Methods Applicability Report for U.S. EPR," ANP-10263P-A, Revision 0, AREVA NP Inc., August 2007**
  - **RLBLOCA - "U.S. EPR Realistic Large Break Loss of Coolant Accident," ANP-10278P, Revision 0, AREVA NP Inc., March 2007 – In NRC Review**
  - **SBLOCA - "Small-Break LOCA and Non-LOCA Sensitivity Studies and Methodology," ANP-10291(P), Revision 0, AREVA NP Inc., October 2007 – In NRC Review**
    - SG nodalization sensitivity analyses requested by the NRC during its review of ANP-10263P-A
    - Incorporates modifications of SBLOCA methodology for U.S. EPR and an updated sample problem
- **Applicability justified by event**
  - **Describe transient, when necessary, by phase**
  - **Identify important components/functionality**
  - **Identify important phenomena**
  - **Confirm phenomena same as for currently operating 4-loop PWRs**

Summary

- **AREVA safety analysis methodologies are mature**
  - **Rod ejection was revised to satisfy new SRP 4.2 requirements**
  - **Approved for application to numerous operating plants**
- **U.S. EPR design, a 4-loop PWR with RSGs, is similar to current 4-loop designs**
  - **Similar operating conditions**
  - **Same phenomena**
  - **Within range of applicability of constitutive models**
- **Methodologies are applicable to U.S. EPR design**
  - **Input models adapted to account for unique features**
  - **Produce expected results**