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

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARD (ACRS)

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SUBCOMMITTEE ON THE US EPR DCD

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WEDNESDAY

MARCH 3, 2010

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ROCKVILLE, MARYLAND

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

SUBCOMMITTEE MEMBERS PRESENT:

DANA A. POWERS, Chairman

MICHAEL T. RYAN

WILLIAM J. SHACK

JOHN W. STETKAR

NRC STAFF PRESENT:

DEREK WIDMAYER, Cognizant Staff Engineer
GETACHEW TESFAYE
JASON CARNEAL
FRED FORSATY
RALPH LANDRY
LAMBROSE LOIS
JOHN BUDZYNSKI
SHANLAI LU
ROBERT DAVIS
JOHN HONCHARIK
TARUN ROY
JOHN WU
JOE COLACCINO
TIMOTHY STEINGASS
LI CHANG-YANG
JOEL JENKINS
NEIL RAY
STEVE DOWNEY
GREGORY MAKAR

ALSO PRESENT:

SANDRA SLOAN
JEFF TUCKER
MARTY PARECE
JONATHAN WITTER

ALSO PRESENT (Con't.):

GARY WILLIAMS

BRIAN HAIBACH

MARK ROYAL

SARAH DAVIDSAVER

CARL BEYER

JOSE MARCH-LUEBA

DENNIS NEWTON

DALE MATTHEWS

JIM BANKE

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

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

(8:29 a.m.)

CHAIR POWERS: The meeting will now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards U.S. EPR Subcommittee. I am Dana Powers, Chairman of the Subcommittee. ACRS members in attendance are Bill Shack, John Stetkar, Mike Ryan, and I am told that the esteemed Professor Apostolakis academician and presumptive commissioner may join us. We will, of course, have a small ceremony should that happen.

The purpose of the meeting is to continue our review of the SER with open items for design certification documents submitted by AREVA NP for the U.S. EPR design. Today we are going to hear presentations and discuss Chapter 4, The Reactor, a relatively inconsequential piece of the overall system, I am sure, and Chapter 5, the Reactor Coolant System and Connected Systems.

The subcommittee will hear presentations by and hold discussions with representatives of AREVA NP, the NRC staff, and other interested persons regarding these matters. The subcommittee will gather relative information today and plans to take the results of the review of these chapters, along with other chapters reviewed by the subcommittee to the full committee when it meets April 8th through 10th 2010.

And as I said, I have not exactly decided what we are going to do there but unfortunately the rules of the game will probably require presentations. Actually, I am going to see if we can get a wavier on that because I am not sure it is useful. We will see. But I doubt it. They are usually fairly draconian on that subject. But we will try to give you some guidance on what will amount to fairly summary presentations.

MS. SLOAN: Very good.

CHAIR POWERS: Okay. The rules of participation in today's meeting have been announced as a part of this notice of this meeting previously published in the Federal Register. We have received no written comments or requests for time to make oral statement from members of the public regarding today's meeting. Should anyone want to make an oral statement or provide me a written comment, they are welcome to try to do so. If they get my attention, we will give them time to make such statements.

A transcript of the meeting is being kept and will be made available as stated in the Federal

Register notice. Therefore, we request the participants in today's meeting to use microphones located throughout the meeting room in addressing the committee. 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 in the back of the meeting room.

And as we had done in the past, people who are speaking for the first time in front of the committee, I ask that you give us a little background about yourself, you, know where you went to school, what you do, what you think you know, and generally why you think you are qualified to speak in front of such an august body as this subcommittee and just so we kind of know who you are and things like that.

The subcommittee meetings are fairly relaxed and whatnot. Not the formal strictures of the ACRS. They are tough. We are easy.

A telephone bridge line has been established for the meeting room today and I understand we have participants from AREVA NP on the line. We would request the participants on the bridge line identify themselves when they speak and keep your telephone on mute during the times when you are just listening.

Do any of the members have opening statement they would like to make with the regard to copies of discussion today or any other aspect of the EPR? I don't actually see any people faunching at the bit to present themselves.

So, I think we will get started and I will call on Getachew Tesfaye, the NRO EPR project manager for some introductory remarks.

MR. TESFAYE: Thank you, Dr. Powers. Good morning everyone. My name is Getachew Tesfaye. I am the NRO Project Manager for AREVA's U.S. EPR Design Certification Project. This morning we will continue our first FSAR presentation of the staff's Safety Evaluation Report with open items.

To date, we have presented Chapter 8, Electric Power, Chapter 2, Site Characteristics on November third, and Chapter 10, Steam Power Conversion System, and Chapter 12, Radiation Protection on November 19 of 2009.

On February 18 and 19 of this year, we presented Chapter 17, Quality Assurance, and portions of Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation. We are currently working on a plan to finish presentation of Chapter 19 the week of April 26.

Today we will present Chapter 4, Reactor and Chapter 5, Reactor Coolant System and Connected Systems.

On April 6 we will present Chapter 11 Radioactive Waste Management and Chapter 16, Technical Specifications.

And that ends my introductory remark, Dr. Powers.

CHAIR POWERS: Okay. What date did you say you were going to try to --

MR. TESFAYE: The week of April 26 is when we were planning to finish up Chapter 19.

CHAIR POWERS: April 26?

MR. TESFAYE: Yes, we are just working on a plan.

MR. WIDMAYER: Yes, we are working on it.

CHAIR POWERS: Yes.

MR. WIDMAYER: That is actually for the -- I don't know that we want to complete the presentations on April 26th. That is just for a meeting at AREVA.

MR. TESFAYE: My understanding was to finish up the Severe Accident portion of Chapter 19 that we began.

CHAIR POWERS: Let's you and I huddle a little bit on this on the break.

MR. TESFAYE: Okay.

CHAIR POWERS: We have got scheduling difficulties.

Okay, Sandra, I think you are up.

MS. SLOAN: All right. Thank you, Dr. Powers.

Consistent with what we have done with the other chapters, our objective here today is to provide a summary level presentation of the organization and material in Chapter 4 of the U.S. EPR FSAR. The focus of the presentation, again consistent with other presentations that we have given is on those features or analytical methods or modes of operation which may be unique to US EPR.

Our two presenters today are from our fuels organization, Jeff Tucker, who will provide more information about this background and his credentials when he starts and Dr. Jonathan Witter, who will also provide part of the presentation.

I would just call your attention to the fact that there is a nomenclature list at the end of the presentation. It is not quite as bad as when we talked about PRA and the acronyms list was multiple pages but just in case, the very last page of the slide handouts is the decoder ring for this.

So I think without any further delay, I will go ahead and let Jeff Tucker start with the technical portion of the presentation.

MR. TUCKER: All right. Good morning. My name is Jeff Tucker. I am from AREVA. Thank you for the opportunity to present U.S. EPR Tier 2 FSAR Chapter 4 today.

A little bit about myself. Currently, I am the project lead for the AREVA fuel activities in support of the U.S. EPR Design Certification. We have a group of engineering staff that works on fuel act. I kind of coordinate with the new plants organization.

I have been with AREVA and predecessor companies for approximately 30 years. During that time I have worked at various aspects of nuclear fuel from fuel design, mechanical analysis, prototype testing, fuel services, and post-irradiation exams of lead assemblies, along with several project roles. I have been involved with design and licensing of several of the AREVA designs over the years, the AREVA 15 by 15 or 17 by 17 designs in particular.

Education. I have a Bachelor of Science degree in mechanical engineering from Virginia Tech.

And that is me.

CHAIR POWERS: Thank you. I think you and our man Sam Armijo could probably discuss for days on this the areas of expertise. But we are going to -- we didn't let Sam show up here. We knew that would happen. So, let's go ahead.

MR. TUCKER: This is just an outline of what we have to present today. I will open with some summary description from Section 4.1. I will also go over in some detail the description of the fuel design system. Then my colleague Dr. Witter will pick up and talk about Sections 4.3 through 4.6.

So the first slide here we would like to present just an overview of the things that our design features and processes that are the same as previous designs very quickly. It is a standard 17 by 17 lattice like our 12 foot designs. We use the High Thermal Performance, HTP intermediate spacer grids in the active fuel zone. We use a nickel Alloy 718 grids at the top and bottom ends of the assembly. We use M5 alloy as our cladding material. Also we use that for the guide tubes and the spacer grids in the active zone.

We use uranium dioxide and uranium gadolinia fuel pellets. We use previously approved codes and methods which we submitted in the topic report 10263, the codes and methods. We used things like our neutronics codes, our physics codes, our mechanical codes. So we are using typical proven. And we use standard fuel management procedures for strategies for power distribution burnup control, such as uranium enrichment, number of fuel assemblies, cycle loading patterns, split BATs enrichments and use of gadolinia radial zone loads at various percents, the use of axial blankets on fuels. All these type of things we have used on our current fuel designs.

The next slide we would like to mention some of the things that are new and different, as Sandra mentioned, about the EPR design. The 14 foot length fuel is not totally new to U.S. but it is new to AREVA here in the U.S. We have extensive experience in Europe with 14 foot fuel. And in the U.S., we also have the 14 foot core that we don't supply but at South Texas and the Palo Verde fuel is a large core, 13 and a half foot core.

So the 14 foot, again, is new for this application but again in France we have quite extensive experience.

The RCCA control clusters use a full span silver-indium-cadmium that is annular. Silver-indium-cadmium is used to provide the rod width and poison and the annular is to match the rod weights with the trip times. So we have a similar trip time for the moving mass of the RCCA with our other reactors designs.

Another difference is there is no center instrument tube. We have 265 rods. The instrumentation uses a guide that moves to the center lattice location. Minor difference.

Incore instrumentation is different for the EPR. We used the Aeroball Measurement System for the calibration of the core monitoring neutronics and the computer codes. We used 12 cobalt self-powered incored detector strings at fixed axial positions. We used online monitoring of the actual DNB and Linear Heat Generation Rate. And my colleague, Dr. Witter, I think when he gets into the neutronics and TH sections, we will go into more detail on this.

Another new difference is the Incore Trip Setpoint and Transient Methodology defined in Topic Report 10287P that the staff, I believe, has approved. This does incore monitoring and provides input to the safety and monitoring systems.

And the last major difference that we touch on here is the stainless steel heavy reflector.

We are using a solid mass of pates, in lieu of the baffle plates and former plates of conventional reactors. And this neutron reflector serves to reflect neutrons back into the core for neutron economy, as well as it reduces the flux, neutron flux on the vessel for extended life of the vessel.

MEMBER SHACK: Do you happen to know what the gamma heating of that heavy plate is? What temperatures will it reach in the interior?

MS. SLOAN: Is there anyone here that can appropriately address that? I'm not sure we can address it now but we will take a note and we will follow up with you.

CHAIR POWERS: What is the stainless steel?

MR. TUCKER: Excuse me?

CHAIR POWERS: What is the stainless steel?

MR. PARECE: I think the reflector is -- this is Marty Parece. I am Vice President of Technology for AREVA. I believe the heavy reflector is 304 Stainless.

MEMBER SHACK: 304?

MR. PARECE: And I don't know the exact temperature. We will have to get that for you but the flow through, we have holes drilled in the heavy reflector to cool it for gamma heating.

MEMBER SHACK: Oh, you do have cooling?

MR. PARECE: Yes, we have holes drilled in it to keep it cool.

DR. WITTER: I will have a figure of the layout.

CHAIR POWERS: So what happens when this thing cracks, Bill?

MEMBER SHACK: Mostly it sits there, I think. I was sort of interested in void growth if it got a little too warm.

CHAIR POWERS: Please continue.

MR. TUCKER: Okay.

CHAIR POWERS: We are just having fun up here.

MR. TUCKER: The next slide shows just a comparison of the U.S. EPR Fuel Assembly to several of the current operating fuel assemblies we have in the AREVA fleet. Again, this is a comparison to our U.S. experience. We don't have the French one on here.

But what I would like to point out is that almost everything is the same; the matrix, the envelope size, the number and location of the guide thimbles, the spacer grid components, nozzle components. The new features again are the length, the 14-foot length, the extra fuel rod in the center lattice position, and the guide tube type. And this fuel assembly we are employing a MONOBLOC, which is a different, thicker guide tube, the different dashpot and I have got details on that later on in the explanation of the components.

So again, this is just to tell you what is new and what is different. Like I said, this slide does a comparison. So again, that is just some of the parameters, in case you are interested in the actual dimensions.

That is the overview part in Section 4.1.

CHAIR POWERS: Remind me what the delta T is from top to bottom of the fuel.

MR. TUCKER: Excuse me?

CHAIR POWERS: What is the delta T from top to bottom?

DR. WITTER: It is about 62 degrees at full power.

CHAIR POWERS: Sixty two degrees in --

DR. WITTER: Fahrenheit. Sorry.

CHAIR POWERS: I knew it had to be that.

MR. TUCKER: Okay, now I would like to move into 4.2, which is the fuel assembly, the fuel system design. And what I would like to do is just discuss some key fuel assembly features. A little deeper detail of some of the key fuel assembly components. Show the interfaces with the incore instrumentation, provide a little bit of the operating experience we have collected and discuss the design evaluations done.

Again, I would like to note that the details of the fuel assembly design for the Tier 2, FSAR Chapter 4 are contained in Topical Report 10285P, the U.S. EPR Fuel Assembly Mechanical Design Topical currently under review by the NRC staff.

MS. SLOAN: And Jeff, I wanted to add, too, there is a significant amount of the proprietary fuel information in that report. So there may be points where if we get into detailed questions we will indicate that that information is proprietary and we will work to figure out a way to get you a --

CHAIR POWERS: Please do so.

MS. SLOAN: We cross into that territory.

CHAIR POWERS: And we can easily do so. And we are perfectly capable of storing these until we can do that proprietary business.

MR. TUCKER: This is an overview of the fuel assembly. The component, of course, the top and bottom nozzles that are stainless steel. The Inconel hold-down springs on the top nozzle. Inconel grids at the upper and lower end positions. The eight intermediate grids are the M5 alloy High Thermal Performance HTP grids. We have of course M5 clad fuel rods, the MONOBLOC guide thimble or guide tubes or the 24 guide tubes.

This is a rounded cage structure to provide structure rigidity. And also one note is the fuel rods are lifted off the bottom nozzle at manufacture, beginning of life.

This fuel assembly has been tested. We made full-size prototype testing. We performed various mechanical tests including a shaker test to determine fuel assembly frequency, pluck test to get the fuel assembly damping, static stiffness, both axial and lateral stiffness, fuel assembly drop and damping.

CHAIR POWERS: How much growth are you going to get during irradiation?

MR. TUCKER: Excuse me?

CHAIR POWERS: How much growth do you get during irradiation?

MR. TUCKER: It is the order of I think like three-quarters of an inch.

We also performed hydraulic testing on the full-sized bundle; pressure drop; life and wear testing; a flow-induced vibration testing where we actually run the rods at the condition to check for fretting. We run an RCCA drive line test where we monitor trip times and control rod wear of the rods in the guide thimbles.

CHAIR POWERS: When you test for vibration in the flow, what range of vibrational frequencies do you look for?

MR. TUCKER: The testing, I guess we do a life and wear test. It is in a full-size loop for a thousand hours where we run it in environmental conditions. But we have an additional testing system that we introduce cross-flows into the stream and fuel-seam alignment and measure fuel assembly frequencies that are like maximum frequencies.

Then we take another set up, an individual grid span with three grids and run that test at that maximum cross flow and frequency for a thousand hours to look for wear.

CHAIR POWERS: So you don't actually look for a vibrational frequency spectrum on this material, on these assemblies.

MR. TUCKER: Can I refer that to my

colleague?

MR. WILLIAMS: This is Gary Williams from AREVA. I have been with AREVA for 30 years, a graduate of Georgia Tech with a BSMS --

CHAIR POWERS: Oh, you are going to be really popular around here.

(Laughter.)

MR. WILLIAMS: If I understood the question with regards to our flow-testing, there is a flow-testing that is done that is primarily a life and wear, primarily focused on wear. It is under hot conditions, of course. That precludes any measurements of assembly vibration of rods.

As Jeff alluded to, we also do cold flow testing where we do monitor the assembly vibration and also rod vibrations under axial flow and also lower span cross-flow conditions.

Of course we are interested in all modes of the assembly response for six modes. Primarily, it has a first and second mode type of response under those flow conditions. We are also interested in the rod response frequencies. Typically the test is done under end-of-life conditions which are most suspect for FIV and fretting and those rod frequencies on the order of 30 hertz or so.

So we are looking for anything that is expected and also interested in anything that is unexpected.

CHAIR POWERS: Yes, we have been troubles with the higher frequency modes and I was just curious.

Thank you.

MR. TUCKER: Okay, we have some pictures here of the assembly components. I would like to now go into a little more detail on each of the components. The next slide just is a summary of the materials used for the reactor, I mean for the fuel assembly. Of course, M5 alloy is used for our fuel rod cladding guide tubes, HTP spacers, the sleeves that attach the end spacers and the fuel rod end caps and guide tube end plugs are all M5 alloy.

Stainless steel --

CHAIR POWERS: I am going to have to confess to a lack of familiarity with M5. I just don't know much about it. But I have heard that there are problems with it. The irradiation growth for some of these zirconium-niobium alloys, I don't that M5 in particular. What is the status here on M5?

MR. TUCKER: I am going to go back to my colleague here.

MR. WILLIAMS: With regards to the M5 growth in particular the use of it for guide tubes, we use M5 quite extensively, aside from the plants that use BEPR, but much of our domestic fuel and also in our European fuel.

To date, we continue to collect and expand the database. We have over 500 measurements worldwide with the M5 guide tube applications, 27 different reactors. Burnup ranges anywhere from seven to 68 gigawatt-days per metric ton burnup. Out of those numbers, approximately 400 assemblies are domestic application measurements.

With regards to the M5 growth issue, we have observed anomalous conditions with regards to certain applications. We have followed up in those applications with more extensive measurements. We have performed our root causes for evaluations, according to our procedures. And we continue to collect data on all of our designs incore to calibrate back to our existing growth models. We continue to assess those growth models relative to the datasets that we continue to collect.

Specific to the growth characteristics, with any zirconium alloy which M5 is, it is largely an axial creep phenomenon that is design-dependent and also guide tube stress state-dependent. And those considerations are used in the course of the evaluation of the EPR and the application of certain data sets that are usable for the EPR design.

There is also certainly a pre-growth component to the alloy, which has been measured under irradiation with no loading on it and typically that particular characteristic is a low contributor to the overall growth phenomena.

But we have observed it is design-dependent, stress state-dependent and that has been largely our focus of evaluation and justification for the growth limits that are use for each.

CHAIR POWERS: Yes, I mean, this core is different because it is so long and, like I say, I don't know anything about M5. Zircaloy I am a little more comfortable with. So I am just curious with what kinds of things we get in trouble with here. Thank you.

MR. TUCKER: And then just briefly, the rest of the materials that have been used before. A 304 stainless for our nozzles, it is 321 and 302 for the fuel rod internals, the nickel Alloy 718 for our in-grids and their springs, and of course the uranium gadolinia fuel pellets.

The bottom line is these components and materials are consistent with those we have used. We have operating experience in the field currently improving these designs.

Next slide. This is a little schematic of our first component I would like to discuss is the fuel rod assembly. This is based, the dimensions, on our 12-foot fuel rod design that we currently have operating in plants in the U.S. In that I mean we have the same pellet OD, density, same pellet OD/ID wall thickness. We use the same proven end plug designs and weld process. The pellets are uranium and uranium gadolinia.

We typically use axial blankets at the top and bottom of the fuel pin to prevent leakage with gadolinia rods as an additional cutback zone above and below the central region, about six inches in length. There is an upper plenum spring that serves to prevent the fuel column from shifting and prevent gas from opening up in the column. That is standard. And then there is a tubular spacer at the bottom to position the bottom fuel column location.

CHAIR POWERS: What is the helium pressurization?

MR. TUCKER: It is 250 psi, I believe.

DR. WITTER: Yes, it is set, preset to ensure the initial rod pressures and gap conductance and then also the maximum pressure at the end of the burnup cycle. But I think it is around 200 pounds, 250 pounds.

CHAIR POWERS: And the clad thickness with M5 is what?

DR. WITTER: Oh, boy.

CHAIR POWERS: Oh, I am terrible. Huh? You can get back to me. It is not a crucial issue.

MR. HAIBACH: This is Brian Haibach. To answer the question, my background is a Bachelor's and Master's from the Pennsylvania State University. I have been with AREVA for about 14 years.

The answer is a 22 and a half mils standard thickness.

CHAIR POWERS: Again, in Christian units. Right? I am terrible, yes.

MR. TUCKER: Again, this is the major difference is a 14-foot fuel rod design. I mention this based on our 12 but is also very similar to the 14-foot fuel used in the 24 French 14-foot plants since 1983. Again, I think Mr. Williams mentioned earlier the type of FIV testing we have done on the rods. So we do have experience with this rod design.

Okay, the next slide. This is another key feature is the MONOBLOC guide thimble assembly. And the unique thing about this design is it has two IDs and only one OD. If you see the figure in the gray box, it is a comparison of the MONOBLOC design to the conventional sway design.

The large OD at the top, of course, provides for clearance for insertion and cooling flow. The bottom reduced diameter for damping, for deceleration of the rods after a trip.

One of the differences is the transition between the two IDs. It is a longer transition zone for the MONOBLOC during the forming process. That longer transition is tuned in the location and length. So we maintain the same flow and the same deceleration as the sway dashpot guide tube.

The guide tube wall thickness at the bottom in the dashpot zone is thicker to maintain the same OD and the result of that is a more robust cross-section of the guide tube to help reduce fuel assembly bow. It is a more rigid structure. So again, a longer transition and a thicker guide tube at the bottom provide structure.

It does also have a quick disconnect feature at the upper end for reconstitution, should that be necessary for inspections or repair and a conventional grid attachment at the bottom. So that is there.

And this operating experience, we currently have this guide thermal design deployed through a number of fuel assemblies in Europe. We have got lead assemblies in the U.S. and our feeling is that it is --

CHAIR POWERS: Where are these lead assemblies?

MR. TUCKER: We have the TVA. We have --

MR. HAIBACH: There are LTAs in the Braidwood reactors from Exalon and the Sequoyah reactors from TVA.

MR. TUCKER: Right with two sets in Braidwood and Sequoyah. Thank you, Brian.

But as I said, AREVA looks at this as maybe not the fix but as a positive design feature, an enhancement to reduce fuel assembly distortion.

The next feature we have mentioned is the high thermal performance or HTP spacer grid. This is a unique feature for AREVA. The thermal mixing, as you can see on the figure on the right is provided by a flow channel we refer to as castilations. The flow enters at the bottom of the grid, it is channeled through these castilations and there is a curved or a slanted exit at the top that induces a swirling pattern in the flow. Another important feature of the HTP grid is the long line contact versus the six-point contact of our conventional or bimetallic or monometallic grids. This has eight line contacts so you have got a larger bearing area, more contact with the fuel rod. It has been proven very effective to reduce rod fretting.

And in fact, with this grid design in the 18 years it has been employed, we have had no incidents of classical flow-induced vibration fretting with the HTP grid design in fuel rods, both Zirc-4 and M5 cladding.

For the U.S. EPR fuel, we did extensive CHF testing on an actual 14-foot bundle with the HTP design. We ran both uniform and non-uniform shapes, heat flux shapes in the CHF perform and the 14-foot bundle.

And we have submitted a unique CHF correlation for the HTP PR fuel assembly to the NRC which was approved. That is topical report ANP-10269. So we have evaluated the thermal performance, especially with for the 14-foot fuel with this grid. We have extensive experience with this grid throughout the U.S. and throughout the world with the HTP but we still ran the unique 14-foot CHF testing.

These grids, as I mentioned earlier, also since they are M5, they are welded to the guide tubes that provides again structural rigidity to resist fuel assembly distortion.

As far as testing these grids, we have performed, as well as the hydraulic testing for pressure drop and CHF, we have performed dynamic mechanical testing, dynamic and static impact crush tests, fuel rod slip load testing to determine the slip load of the rod through the grid cell, and then handling an interface test to test the grid lead-ins.

The dynamic impact test we use to determine the acceptance criteria for faulty condition loading and also to provide inputs fuel assembly structural models used to evaluate the fuel.

As I mentioned, we have over 18 years of operating experience with the HCP up to 62,000 gigawatts and all types of fuel from 14 by 14 through 18 by 18 both in the U.S. and Europe. So that is the HTP intermediate spacer.

The next feature is very similar, the HMP or High Mechanical Performance spacer used at the top and bottom. The main difference between this and the previous HTP is the flow channels are not curved. They are straight. You don't need the mixing in the unheated regions at the top and bottom. Also, it is made if Inconel or Alloy 718 versus M5 because of its low radiation relaxation and the enhanced rod support provided by the alloy 178 material.

They also are attached to the guide tubes to maintain the structure rigidity but they are attached by a series of sleeves. With this HMP design, spring contact is maintained throughout the life of the fuel assembly. So we maintain rod position and rod fixity at the ends where the higher cross flows usually occur.

The HMP operating experience, we have used that in Europe since 1992 and in several U.S. designs. In fact, it is a staple design now in our BNW 15 by 15 product with very good results. It has resisted things like the baffle flow jetting problems we have had in some of those plants.

The next feature is the low pressure drop top nozzle assembly. Again, this is very similar to the nozzles that are used on our 12-foot designs and very similar to the 14-foot nozzles used in France. We thought it had low pressure drop because the flow holes through the grillage are optimized to maximize the flow, minimize the pressure drop while maintaining our strength requirements for our different loading conditions.

It consists of a 304 stainless steel structure. It interfaces reactor internals so we have the diagonal corners to core guide pin holes that interface with the upper core plate. It is very classical to our 12-foot designs.

The Alloy 718 springs consist of four sets. It is a five-leave spring pack in each set. Those springs have been tested and proven again, very similar to the ones used in our 14-foot European designs and we have had good performance with nozzle and the springs. No issues with these springs breaking or cracking.

The next component is our FuelGuard bottom nozzle assembly. This is another unique AREVA design. It consists of a frame of stainless steel. A machined frame with the feet that interface with the bottom nozzle, I mean the lower core plate. The grillage consists of a series of unique curved blades and connector rods that are brazed in position to form the grill structure. So there is really no line of sight through this grillage but the flow goes through the curved path. This is used for enhanced debris trapping to reduce debris fretting failures in the fuel. Again, it has a conventional interface with the reactor internals.

In our operating experience, we currently use this in operating plants in the U.S. and Europe and this has been tested for load carrying capability again for validation and for input into our fuel assembly structure models.

CHAIR POWERS: Have you tested this debris trapping against the kinds of fibrous type particulate that they are worried about for this some blockage issue and things like that?

MR. TUCKER: Those testing are currently ongoing. I don't think the results are conclusive yet.

MS. SLOAN: We are still evaluating the test results right now. But the answer is yes. We have performed testing related to GSI-191.

MR. TUCKER: Right.

MS. SLOAN: I assume that is what you are referring to.

CHAIR POWERS: That's right. And it is particularly the fibrous stuff that I think comes up the most. And I have to confess a certain amount of ignorance on this -- actually none at all.

Who is testing? Is it your own testing or somebody else's testing?

MR. TUCKER: We are testing in a lab.

MR. HAIBACH: A third-party independent.

MR. TUCKER: A third-party independent laboratory.

MR. HAIBACH: The third-party laboratory is doing a testing for AREVA.

CHAIR POWERS: And sooner or later we will find out about that.

MS. SLOAN: You will. In fact that gets addressed under the guise of GSI-191 in Chapter 6.

MR. TUCKER: Okay. The next new and different that I would like to show here is the interface with the incore instrumentation. Classically our fuel designs have had a center instrument sheath from a bottom-loaded detector or probe. With the EPR we have no bottom penetrations. So the detectors come through the top.

Where they enter is through one of four potential guide tube locations as shown in the figure. Depending on the location of the fuel assembly incore where it is located under the instrument lance, it will be in one of the four symmetric locations to the one shown.

In some cases where we have both an SPND and an aeroball probe, it could be in two diagonal locations. So there again, it is top loaded through the guide tubes. And since of course, a fuel assembly that has this doesn't have any component, no control rod or RCCA in the assemblies with the instrumentation.

And then I guess I have mentioned this as we go along, but this is just a summary of the operating experience for some of our key components. Just to show you, we have had 33 reloads in 15 reactors in the U.S., over 2300 fuel assemblies provided to date. Globally over a million and a half M5 rods with excellent corrosion performance. The low oxidation low hydrating have been confirmed by post irradiation exams on a number of these fuels as I think Gary Williams referred to.

We have over 3,000 fuel assemblies that have both the M5 guide tubes and the fuel rods in 37 reactors. And as mentioned, we have been in everything from the 14 by 14 to the 17 by 17, 18 by 18 fuel. And we have achieved burnups of 68,000 gigawatt days in lead assemblies. So we have, in special cases we have gone and pushed the burnups to above the 62 and the lead assemblies I think in the North Anna reactor.

CHAIR POWERS: The oxide on this cladding, I mean, this is kind of not part of our review but I will ask anyway. The oxide on this alloy is different in some respects to the zircaloy oxide. How does it do with respect to boric acid absorption and things like that?

MR. TUCKER: I don't know the answer to that question, sir. I don't have a materials person here. I know our oxidation is low in the range of 30 to 50 microns.

MS. SLOAN: Unless we have somebody who can answer that, we are just going to have to make a note and come back to you.

CHAIR POWERS: Yes, that's fine. I mean, it's not really pertinent here. But like I say, I am just inexperienced.

MR. WILLIAMS: I would only supplement it in the realm of numbers of reactors type and use of boron, we have not seen anything unusual with regards to the boron deposits as it relates to the affects on the oxide for the M5. The oxides measured on M5 are very typical and as expected if I understood your question.

CHAIR POWERS: Does the niobium get

incorporated into the oxide or does it partition back into the alloy as oxidation progresses?

MR. WILLIAMS: That I do not know. I cannot answer that.

CHAIR POWERS: Yes, let's not get hung up on that and progress ahead.

MR. TUCKER: Okay. Like I said, as I have mentioned the HTP spacers, again another unique AREVA feature, the robustness and resistance to fretting has been proven. Eighteen years of experience, discharge burnups I think in Europe as high as 70 gigawatt days. Again, all fuel assembly types from all reactor vendors.

As I mentioned earlier, no fuel rod fretting failures with the classical grid to rod fretting vibration. Like I said, we have deployed this in problem areas as baffle gap areas, lower --

CHAIR POWERS: The codicil at the end of this just begs a question. It suggests that you have flow-induced vibration fretting due to non-classical grid to rod fretting.

MR. TUCKER: Well just to -- I think the cases we have had where the HTP grid has been employed as the intermediate grid and we have had an Inconel or a bimetallic end-grid, it had no problem. But I think in one or two cases where there was a zirconium grid at the end location that didn't maintain that fixity, then those rods were allowed to move because of no fixity and then they wore but it wasn't fretting. It was a motion.

So again, when we had an Inconel or a bi-metallic end-grid, there were no fretting problems. That is what I meant by the classical. I just wanted to get that out. I knew that was what you were going to ask.

CHAIR POWERS: I mean, it just kind of begged the question.

MR. WILLIAMS: To supplement Jeff's comment, of course the EPR design uses the bottom Inconel grid and we have identified that that is indeed critical to the design in making sure that there is a capture of the rod at least at one grid elevation. Typically at the intermediate grid elevations with the grid relaxations, grid growth, fuel rod creep-down. Gaps do indeed form.

The HTP design, by nature is robust under those types of boundary conditions and also the levels of excitations that it might be exposed to. So the HMP is a critical component with regards to the mix of the grid configurations, which is represented on the

EPR. We continue to, as the industry goes into its baselining for grid-to-rod fretting, we are monitoring those designs that utilize the HMP bottom grid with the HTP intermediates and uppers.

And with the OE and with the data collected to date, the results of have been very promising with regards to this particular configuration.

MS. SLOAN: And I would just remind the AREVA staff, for the benefit of the transcriber, please state your name before you --

MR. WILLIAMS: Each time?

MS. SLOAN: Each time I think would be --

MR. WILLIAMS: I'm sorry.

MS. SLOAN: Oh, no? Okay. All right.

MR. TUCKER: Okay. And then of course, the 14-foot fuel is the next feature. But again, we have extensive 14-foot fuel experience in France, 24 reactors in the 1300 and 1450 megawatt plants. We have been making 14-foot fuel since 1983. And over 25000 fuel assemblies delivered to date.

And of course in the U.S. it is not AREVA designs but you know, we do have the South Texas which is a very similar design to this with a 14 foot Westinghouse-style 17 by 17 fuel.

So that is just a summary of the OE that I think I have mentioned along the way. The next slide just to briefly discuss some of the design evaluations done from a mechanical point of view.

Design criteria that we have looked at is the fuel assembly damage criteria; stress, cladding strain, fatigue, fretting oxidation, hydride, crud buildup, rod bow, actual growth. That is the damage criteria.

Fuel rod failure criteria including hydrating, cladding collapse, overheating of pellets, PCI, cladding rupture, full fuel coolability, mostly of which have been addressed in Tier 2, Chapter 15 safety analysis and the LOCA region.

CHAIR POWERS: At what point do we get to that discussion on the thermal hydraulics on this point?

MR. TUCKER: Thermal hydraulics in this presentation?

CHAIR POWERS: No, no.

DR. WITTER: The safety analysis portion?

MS. SLOAN: Do you mean methodology or performance?

CHAIR POWERS: Yes.

DR. WITTER: Well, I will talk a little

bit about the thermal hydraulic design aspects in my section.

CHAIR POWERS: Yes, it is really the accident --

MS. SLOAN: I think it is Chapter 15. You are going to see most of that in Chapter 15.

CHAIR POWERS: Yes. When are we scheduled to get to that?

MR. WIDMAYER: We don't have a schedule or I don't remember.

MS. SLOAN: Still under discussion.

CHAIR POWERS: Professor Banerjee just asked me and I said, you are going to love that one.

MR. TUCKER: So again, we analyzed for these criteria consistent with the NUREG SRP 4.2 requirements. Again, as I mentioned earlier, the design evaluation was performed with NRC approved codes. And this is documented in detail, the codes in 10263, the codes for neutronics and thermal and mechanical analysis were approved.

And again, these results are summarized and detailed in the topical report 10285 U.S. EPR Mechanical Design Topical currently under review by the NRC staff.

So that is all I have. Dr. Witter is up

next with the continuation of Section 4.3 and 4.4.

CHAIR POWERS: Any other questions on this? That was a superb job. I really enjoyed that.

MR. TUCKER: Thank you.

CHAIR POWERS: That was fun. I just don't understand M5. That's all. My fault. Okay.

DR. WITTER: All right. Good morning. My name is Jonathan Witter. A little bit of background about me. I received my Bachelor's and Master's from Rensselaer Polytechnic Institute in 1983, went into the operations aspects of the Naval Nuclear Propulsion Program, qualified in the prototype system and was qualified then as a nuclear plant engineer for training. And so I did some of the training aspects.

Then after that I moved on to BWR and worked as a station nuclear engineer for four years at the Fitzpatrick plant in Oswego.

After that, I decided I needed to go back to school and get my Ph.D. I went to Massachusetts Institute of Technology. I worked under Dave Lanning and John Meyer.

CHAIR POWERS: Well we don't have any respect for that.

DR. WITTER: Right. Dr. Apostolakis isn't here.

CHAIR POWERS: Yes, we kicked them out.

DR. WITTER: Upon graduation, I went to work at the Knolls Atomic Power Laboratory. I worked for 12 years in the nuclear design and system design in the advanced concept area. And at the last stages in the space partnership that the naval nuclear propulsion program had.

When that program went away, I decided to reinvestigate the commercial industry and ended up at AREVA to work on the EPR project. And I have been at AREVA now for four years.

CHAIR POWERS: You got all that BWR experience expunged out of your system.

DR. WITTER: Yes, a little bit of that.

CHAIR POWERS: And a little design and things like that.

DR. WITTER: Gas reactors from space.

CHAIR POWERS: You don't know anything about that boiling. And Jupiter moons is just no good at all. Right?

DR. WITTER: Right. Too far off. I figure --

CHAIR POWERS: Hey, right now we have got Cassini going up.

DR. WITTER: That's true.

CHAIR POWERS: And it is doing good, too.

DR. WITTER: So that is a little bit of background about myself. I will be covering the next sections of the Chapter 4. The first --

CHAIR POWERS: Sandra is looking at us. What are they talking about Jupiter moons and stuff. Oh, this is fun, Sandra. This is --

MS. SLOAN: We are here to entertain.

CHAIR POWERS: Wait until you see the Jupiter moons design.

DR. WITTER: It was wild. Okay, onward to Chapter Section 4.3, the nuclear design section.

The core, as Jeff Tucker mentioned, the codes and methods were approved in an early topical report 10263, where it essentially outlaid the methods and codes that we would be using. And so we are following traditional design aspects for the nuclear parts of the design and it conforms to the NUREG-0800 SRP 4.3 guidance.

The design goals are to maintain the power distributions and peaking limits through the use of gadolinia loaded fuel rods, radial and axial loading zones, and also multiple fuel types within the batch replacements for the reloads.

An assessment of the reactivity

coefficients are considered through all cycle lifetimes, through the moderator temperature coefficients, fuel temperature coefficients, power defects, and the boron worth throughout life.

One feature that we are using is enriched boron rather than the natural isotopic B10 component. It is enriched to approximately 37 percent, weight percent of atom -- sorry -- atom percent of boron-10 within the boron structure of the boric acid.

CHAIR POWERS: How much difference does it make whether it is atom or weight percent?

DR. WITTER: Actually for boron it makes a difference between boron-10 and 11. Unlike uranium, it is about the same but boron it is significant.

For the control rod functions, of course maintaining adequate shutdown throughout the life cycle and then also the ability to control any axial oscillations that may be induced either through xenon or to maintain the actual power shape throughout life as well.

Some of the new features that I will have a few slides on in the upcoming slides are a little bit about the heavy reflector use for the nuclear design aspects of it. The Aeroball Measurement System and how it is used and has been in used in Germany and how we are using it here for the U.S. EPR.

The incore detector system being used both for monitoring and control but also for the protection system, which is a new feature for the incore detectors being used for a protection system. And then the use of the annular control rods, as Jeff Tucker had mentioned before.

CHAIR POWERS: What is the attraction of silver-indium-cadmium?

DR. WITTER: The attraction is rather than using B4C, let's say, is the helium production for the output production is you don't have the rod pressurizations, the issues with clad growth and rod internal pressures, the swelling.

And so it has reduced swelling and pretty much minimal helium and production.

Okay. Kind of some of the components of the nuclear design aspect. For the first phase, the fuel assembly, loading patterns itself, which shows a notional description of the radial 17 by 17 array of the lattice, where we make sure of 265 positions for fuel rods. Then there are 24 guide tubes. Rather than having a central guide tube, now that center position is replaced with a fuel rod.

And then the distribution of the gadolinia loadings, depending on how flat the power distribution needs to go, what sorts of power requirements or lifetime requirements there are, we will use different enrichments or weight percents of gadolinia loaded fuel rods arranged in patterns. Again, following the standard design practices that we use for our current reload fleets to maintain a relatively flat power distribution and peaking within our design limits.

The figure on the right-hand side shows the axial zonings that we take advantage of. Well the central zone is used for both just normal uranium loading but also the gadolinia loading region is in the central region.

The cutback zones are bout six inches in axial span and that would have the same uranium-235 enrichment as the central zone loadings. So gadolinia is used as a lower enrichment to aid with the power peaking after the gadolinia is removed and also to accommodate the degrade of thermal connectivity of the gadolinia.

The blanket regions are used to reduce the axial neutron leakage. And that is typically used with a lower enrichment of uranium in those regions. I think that kind of covers the fuel assembly axial and radial loadings. And there will be different fuel assemblies used in the batch. There won't just be one fuel assembly to help with the radial distribution, which we will get to that.

The next slide discusses a little bit about the arrangements for the rod cluster control assemblies. Again, we have adopted a traditional approach for a base-loaded operation plant where we have a series of there are 89 control rods disbursed throughout the reactor core. Several of them are used as shutdown banks distributed through the core. But then for power control, we have the control banks. Traditionally the D and C bank used at power operations. And within the D bank, there are nine control rods spread regularly through the core to help ensure that we don't -- well, as they are inserted, that a relatively uniformly distributed radial power distribution will occur when the rods are inserted.

On the left-hand side, it shows our power dependent insertion limits, whereby these limits are set to ensure that we do not exceed axial peaking as the rods are inserted but also to ensure that we maintain adequate shutdown margin with the boron concentrations.

There is an amount of overlap occurring at about 50 percent power as the D bank is inserted. I should explain the X axis is the power level, it is a little small to see, and the Y axis is rod insertion position.

At about 55 percent power, the C bank will begin to follow the D bank with about a 234 step overlap, again, to ensure adequate shutdown margin and axial power peak shapes that are within our limits.

Okay, the next figure shows the radial power distribution at the beginning of Cycle 1. And again, some design goals and constraints that are placed on the nuclear design. The base load operations, assuming 100 percent power throughout the cycle life, and minimum maneuvering for maintenance outages or maintenance power reductions.

For the final safety analysis report, the base reference design is the 18-month cycle design. It has the constraints of a maximum average fuel rod burnup of 62 gigawatt days. The average linear heat rate for the core is 5.21 kilowatts per foot. It is slightly lower than the classical or typical PWR in operation today, primarily due to the 14-foot height, the additional fuel rod in the middle of the assembly and the larger, the 241 fuel assemblies.

The peak LHGR or linear heat generation rate is kept below 13.5 kilowatts per foot, based on the local peaking factor limits of 2.6. And then also a design constraint for both local and DNB is that the maximum axial integrated enthalpy rise or F delta H is limited to a radial peaking of 1.7.

As the figure shows, at the beginning of Cycle 1, the maximum peaking occurs in the middle of the core and in some locations, near the edge of the core, with a maximum of 1.25. It would grow over this cycle and then towards the equilibrium cycle, the maximum F delta H experienced from the nuclear design portion is 1.5, indicating a lot of margin to the 1.7 design limit itself. And I think that covers that aspect of the reactor design.

For the heavy reflector, this shows actual photographs of the OL-3, heavy reflector under construction and then the side radial view of the layout relative to the reactor core fuel assembly locations and then the location of the heavy reflector. Where in the figure, the photograph and also the figure, you can see there are cooling holes drilled into the stainless steel to provide cooling for the gamma neutron heating. It is set up to reflect the fast neutrons back into the core to reduce radio leakage and also to reduce the vessel fluence, primarily the vessel fluence for that. And it does provide a better neutron economy, some better neutron economy to allow the potential for slightly lower batch loading. So there is some fuel economy aspects.

CHAIR POWERS: I presume we will eventually get to the vessel itself.

DR. WITTER: Say again?

CHAIR POWERS: I presume we will get to details on the vessel itself at some point.

DR. WITTER: Yes. Actually that is --

MS. SLOAN: Chapter 5.

DR. WITTER: -- Chapter 5 this afternoon.

MS. SLOAN: Chapter 5, in the afternoon.

CHAIR POWERS: Can you give me any hint on what kind of fluence you expect on the vessel weld?

DR. WITTER: Fluence levels I believe -- I don't want to misspeak.

MR. PARECE: This is Marty Parece.

CHAIR POWERS: I mean, I can wait.

MR. PARECE: Yes, we will give you the exact numbers later but the design is such that this 4590-megawatt thermal core or plant, after 60 effective full power years will have a lower fluence on the welds than a current 3400-megawatt unit after 32 EFP lines.

CHAIR POWERS: I mean, that is same -- we will get to the details later.

MR. PARECE: We can have the number for it this afternoon.

CHAIR POWERS: Yes, that would be great. We are going to put Shack out of business here.

DR. WITTER: And the aspects of this for the nuclear design aspect, much like the axial reflectors up in the upper regions above the core and below the core where there is no fuel special cross-sections are generated to handle the radial reflector in those regions. Rather than just using water, the cross-section sensor generates to accommodate the approximately 95 percent metal and five percent water in those regions.

But moving on to the next section for the thermal hydraulic design, this table shows a comparison of current operating plants and then the U.S. EPR. This has been displayed before except for this time we added the Palo Verde plant because of the 241 assembly plant and a slightly higher power than the current 4-Loop plants without power upgrades.

The key things to note there again are the linear heat rate being a little bit lower and the peak linear heat rates also being slightly lower. But the rest of the plant is pretty much an evolution of within the current plant operating conditions for temperatures, pressures, and flow rates. There is the additional loop flow rates of course because it is a higher megawatt thermal rated system. But in the end, a lot of the thermal parameters for the thermal hydraulics are very similar to current plants.

Because it was a larger vessel and interested in the flow distribution coming into the fuel assemblies for the bounding conditions, there was a one-fifth scale test mockup setup in France to investigate the inlet flow distributions and verify and validate the lower plenum flow distribution device.

There are two phases to the flow distribution. One, the lower support plate has holes drilled into it with various friction factor, K over A squared terms where they will have slightly reduced values in the middle and then in the outer regions to provide little bit more flow, the holes will be slightly larger.

The flow distribution device is meant to help reduce the flow vortices that would, if it weren't there, you would have more flow going to the middle regions of the core. So it helps flatten the flow inlet distributions and was meant to help couple it back to the design experience for the N3 and N4 plants, either make it the same or better than those plants. And that distribution device with the testing, as the figure shows, it shows that the inlet mass flux to the assembly is where you still have a higher flow rate in the middle of the core where typically you will also have slightly higher powers. But then once there, the flow is entered into the assembly. In Chapter 4, it also shows the mass fluxes at different elevations of the core. So you can see how the flow redistributes a little bit once it is in the open lattice of the fuel assemblies.

MEMBER SHACK: Now this scaling was not done matching Reynolds numbers, presumably.

DR. WITTER: That I would expect that they would have some model similar to for the mockups. What it had is it didn't include all the fuel assembly regions. They had different mockups that they would insert into the reactor vessel there. This particular test was focused on the lower plenum region and the flow into the fuel assembly.

MEMBER SHACK: But it was done with water and air?

DR. WITTER: I am pretty sure it was done with water and air, yes.

CHAIR POWERS: I mean you indicate that this took place. Do we have access to the information on this test?

DR. WITTER: I believe there --

MS. SLOAN: I beg your pardon. Could you repeat, Dr. Powers?

CHAIR POWERS: I am interested in the details of this test. Do we have access to those?

MS. SLOAN: Anything that you --

CHAIR POWERS: Is there a report, I guess is an easier answer.

MS. SLOAN: Oh, you are looking for a test report.

CHAIR POWERS: Yes.

MS. SLOAN: I am sure there are test reports. Do you need us to make those available?

CHAIR POWERS: I have a feeling our thermal hydraulics gurus, who happen not to be here right now are going to be intensely interested in this.

MS. SLOAN: Okay. We will work with Derek.

CHAIR POWERS: Yes, if you can just find out about this. Now, Mr. Witter, you said that we could -- I am not sure how to interpret your statement. We had more flow coming in the inside and then of course as it goes up, we get to our cross-flows in there. And you said we can see how the flow distributes. I don't see how the flow distributes. Maybe you could explain that again?

DR. WITTER: Well, with the inlet mass fluxes, you will end up having essentially a pressure differential which will then drive the flow across. The vast majority of it, was a high velocity --

CHAIR POWERS: It was the statement we can see and I couldn't see.

DR. WITTER: Oh, I'm sorry. Yes. Yes, within the chapter itself --

CHAIR POWERS: Okay.

DR. WITTER: -- it has tables of the mass fluxes as you go up --

CHAIR POWERS: It does.

DR. WITTER: -- the S elevation. I don't have them displayed here.

CHAIR POWERS: You said I could see there and I --

DR. WITTER: Correct.

CHAIR POWERS: -- well, I know it is in the chapter but --

DR. WITTER: My bad.

The next figure shows again Jeff Tucker mentioned that we had done CHF testing for the EPR assembly and specifically determined a CHF correlation that is known as the ACH-2 Correlation which was documented and approved for use in ANP-10269. And as Jeff mentioned, the testing was done at the Karlstein Thermal Hydraulic Test Facility in Karlstein, where we used multiple heated lengths and also used uniform and non-uniform axial shapes. And the test assembly was a mockup of a five by five rod array with and without a guide tube. So the correlation has developed for both the fuel rod unit cell and also a unit cell with a guide tube present.

Moving on to another aspect of the thermal hydraulic design, it includes the core instrument as part of this chapter, where we have a series of excore detectors, the traditional source range, intermediate range and power range detectors.

Fixed incore instrumentation which, as Jeff mentioned, there are no bottom penetrations for the incore instrumentation that comes through the top vessel head outer radial dimensions and comes in through lances. And I will have some figures of that. And then also thermocouples to measure the core outlet temperature.

And then coupling with that to provide a three-dimensional view of the actual flux in the core and to aid in the calibration of the incore detectors is the Aeroball Measurement System, which the next slide will have an image of that system.

The Aeroball Measurement System is a new feature for the U.S. -- well for the EPR and for the U.S. EPR but what it is, it is an electric mechanical computer controlled online flux mapping measurement system. A lot of words but essentially there are a series of small severe steel balls with some vanadium concentration in them. It can provide a rapid power measurement of the core rather than using say for a boiling water reactor traversing incore probe system, through various locations. Here these spheres are moved into the or blown into the core with a nitrogen gas system to the 40 locations, 40 radial locations in the core, and then they are transferred back into a measurement room where the middle figure shows the counting table for a typical Konvoi plant where it has been in use for 35 years.

That provides 36 axial locations or 36 detectors axially to provide the detailed measurement. And it measures that after the irradiation or activation of the vanadium, the counting table recreates the power distribution in fine detail.

The figure on the far left-hand side shows the relative size of these spheres. They are 0.067 inches in diameter and you can see they are about the size of a pencil point being moved in and out of the core. And as I said, the Germans have been using this system for about 35 years. More information about this system is presented in the Section 7 of the FSAR.

MEMBER STETKAR: Controls are -- can I stop you here? This is kind of a fascinating neat little system. You have got to love the Germans. You have just got to love the Germans for moving strings of little balls around with little bursts of nitrogen.

Section 7, I think, talks a little bit about the instrumentation controls but it doesn't talk much about the mechanical part of the design, which is something I am a little bit more interested in.

The carrier gas tubes, if you look on the right side of your slide there, the carrier gas tubes are the reactor coolant system pressure boundary. Is that correct?

DR. WITTER: I believe that to be -- Mark Royal is --

MEMBER STETKAR: Okay.

DR. WITTER: -- person that will do the

AMS system.

MS. SLOAN: Mark, you can introduce yourself.

MR. ROYAL: Mark Royal, Instrumentation and Controls. I am from Charlotte, a former Naval nuclear reactor operator. After that, I have a Bachelor's degree in physics from University of North Carolina at Charlotte. Prior to working with AREVA I worked with Norfolk Naval Shipyard in instrumentation and controls, specializing in neutronics, overhaul installation and testing. I have been with AREVA for about four years. I have been an engineer for Aeroball and Incore.

Okay, so to answer your question, you are asking if the carrier gas tube is the pressure boundary. Is that correct?

MEMBER STETKAR: That is correct.

MR. ROYAL: Okay. Actually, you have a tube within a tube, within a tube. The outer tube presents, and there are some figures of this as well as we get on, but you have a tube within a tube, within a tube.

MEMBER STETKAR: Are there figures? I mean, if you are going to discuss this later.

MR. ROYAL: Yes, right here.

MEMBER STETKAR: Oh, this is worse, though.

MR. ROYAL: Then let's not look at that.

MEMBER STETKAR: Okay.

MR. ROYAL: In NAP-10282 topical there is a figure that represents what I am about to describe. The tube within the tube within the tube. Okay, the outside tube is what we call your lance finger, which penetrates down into the core. As Jonathan will tell you later on, the tube fingers, depending on the orientation will have either aeroball tubes or will have SPNDs and correlate thermocouples.

With the aeroball tubes, the outside lance is the pressure boundary which is subjected to temperature pressure conditions of the primary. Within that, you have another tube which is called the protection tube. Within the protection tube, you actually have the aeroball tube itself. So the gas, the carrier gas nitrogen puts the balls into the aeroball system tube within the protection tube within the lance tube. And then after irradiation, the valves realign with the aeroball measurement system and position nitrogen such that it brings the balls back out of the reactor and takes it to the measurement table for measurement.

MEMBER STETKAR: The protection tube that the -- the lance tube. No what they call the protection tube, the central tube, that is perforated, though. Right? That is not a solid. If I remember reading about it.

MR. ROYAL: Okay, once more, it is a terminology approach. The outside tube --

MEMBER STETKAR: Is the lance tube.

MR. ROYAL: -- is the lance tube, which has the perforations.

MEMBER STETKAR: Oh, that one has the perforations.

MR. ROYAL: Yes. That allows the reactor coolants to come in.

MEMBER STETKAR: Okay.

MR. ROYAL: The protection tube is just that it is protection for the inside tube.

MEMBER STETKAR: Right, which is on this drawing. The red deal.

MR. ROYAL: Well that is not the best representation of what we actually have.

MEMBER STETKAR: I couldn't find a good one. That is why I --

MR. ROYAL: We actually gave a presentation on September 16th and 17th for the NRC about the aeroball measurement system. It took two days to do the presentation. So there is ample information. I think we will answer the questions there.

MS. SLOAN: We will send a better figure. We will send it through --

MEMBER STETKAR: I actually have a 58-page response to an RAI that --

MR. ROYAL: And my name is probably on it.

MEMBER STETKAR: -- I made copies of the pictures from and I couldn't find a good diagram that shows the actual tube configurations.

MR. ROYAL: Right.

MEMBER STETKAR: Let me ask -- so we will put that on the table for a moment.

One of my questions and concerns has to do with potential LOCA pathways through the gas system. You know, so obviously the number of barriers is an issue but as you said, we will put that on the table for a moment.

MR. ROYAL: Now that was --

MEMBER STETKAR: That is an important issue.

MR. ROYAL: That was answered in RAI as well.

MEMBER STETKAR: There was a discussion about that in the RAI.

MR. ROYAL: Yes.

MEMBER STETKAR: I am going to ask a few questions here.

MR. ROYAL: Okay.

MEMBER STETKAR: As I understand it, the gas tube is, internal diameter of the gas tube is about four millimeters. Right?

MR. ROYAL: Approximately.

MEMBER STETKAR: So it is a pretty small tube.

MR. ROYAL: Yes.

MEMBER STETKAR: When the aeroball system is in I will call it standby, normal power operation when the aeroballs -- you call them stacks.

MR. ROYAL: Yes.

MEMBER STETKAR: You know, when the aeroball stack is in the rest position, whatever you want to call it, upstream of the block, --

MR. ROYAL: Right.

MEMBER STETKAR: -- what are the normal positions of -- There are little solenoid-operated block valves and then there are three-way valves that work the nitrogen around it.

MR. ROYAL: Yes.

MEMBER STETKAR: Are all of those valves -- is the system depressurized at that time?

MR. ROYAL: That's a good question. Yes, the system, the tubes themselves, now there is, obviously a nitrogen gas bladder that is pressurized.

MEMBER STETKAR: Yes.

MR. ROYAL: But the nitrogen tubes themselves for reading remain depressurized until there is a reading that is taken. If that answers your question.

MEMBER STETKAR: So when you basically turn the system -- in simple terms. You hit the button, turn the system on.

MR. ROYAL: That's right.

MEMBER STETKAR: Then from there on out, the control system determines the cycling of the valves --

MR. ROYAL: That is correct.

MEMBER STETKAR: -- you know, and the timing to shoot the balls.

Where does the -- again, I have got a simple diagram from the RAI. Where is the exhaust gas line? Does the exhaust gas line gout of the containment?

MR. ROYAL: That's a good question. That was also answered in the RAI. The exhaust gas, the nitrogen exhaust I am thinking you are referring to.

MEMBER STETKAR: Yes, the nitrogen exhaust.

MR. ROYAL: That is exhausted back to the pressurizer shed area.

MEMBER STETKAR: So it is inside the containment.

MR. ROYAL: Right. It is not -- well all of it resides in the reactor building. The measurement system itself doesn't --

MEMBER STETKAR: Well, I mean the nitrogen supply line, where you charge the accumulators or whatever actually comes in from outside the containment.

MR. ROYAL: Yes, it does.

MEMBER STETKAR: But the exhaust line stays --

MR. ROYAL: Is returned back to the containment, yes, sir.

MEMBER STETKAR: Oh, okay, good. Good. Thank you. Let me see if I had any other questions while you are here. Yes, one last question.

You rely on the -- you said there is both

I can't remember whether there is an activity monitor. I know there is a --

MR. ROYAL: Yes, there is a radiation monitor --

MEMBER STETKAR: There is a radiation monitor and also a humidity monitor or something that will close.

MR. ROYAL: Yes.

MEMBER STETKAR: Or at least it says there is one.

MR. ROYAL: Actually it is a spark plug. Yes, it is a Bosch spark plug that uses a humidity monitor for any changes in the resistance of the spark plug itself. It works great. Really.

MEMBER STETKAR: You have got to love the Germans.

MR. ROYAL: It could be Autolight for America but I am not sure.

MEMBER STETKAR: No, no, Bosch is good.

MR. ROYAL: But it is actually at the table itself. You are checking for humidity in the lines so you are looking at all the lines as they come in, to ensure you don't have any humidity there.

MEMBER STETKAR: And those things close when they detect either high radiation or high humidity in the line, they close the solenoid valves that isolate them?

MR. ROYAL: It is also looking at a high pressure differential as well if it senses a high pressure, yes.

MEMBER STETKAR: Okay. That I didn't read.

MR. ROYAL: They close the quick-closing valves that are shown there.

MEMBER STETKAR: Are they energized to close valves or are they de-energized to close valves, do you know that?

MR. ROYAL: Those are energized to close.

MEMBER STETKAR: Okay. What is the power supply for this whole thing?

MR. ROYAL: For the power supply in Europe they are using for the whole system is a 480-volt three-phase. Now that comes into a central electronic distribution cabinet, if you will, that changes it to the necessary voltages for the electronics. As you see on the table, those use a PIPS detector, which obviously has a higher voltage but the main voltage that will be used will be a 480-volt source.

MEMBER STETKAR: So that comes in from the out. So the external power supply to the whole thing.

MR. ROYAL: Yes, that is correct.

MEMBER STETKAR: You break it down to D.C. or whatever you need internally.

MR. ROYAL: Yes, that is correct.

MEMBER STETKAR: Is that a safety-related power supply or is that a non-safety power supply?

MR. ROYAL: Since the aeroball measurement system is classified as a non-safety system, it will be a non-safety power supply in that in the event that the aeroball system isn't functional, it does not cause you to have to shut down the reactor.

MEMBER STETKAR: Okay, I just wondered. Thank you.

MR. ROYAL: Thank you.

MEMBER STETKAR: Thank you very much. But someplace, I would like to get a decent picture of those tubes. I don't think there was one in the RAI.

MS. SLOAN: We will take an action to get you a better diagram and we will send it to Derek.

MEMBER STETKAR: There was some brief text description of that.

MR. ROYAL: Right.

MEMBER STETKAR: Thanks. Sorry.

CHAIR POWERS: No need to apologize.

MEMBER STETKAR: This is a neat little

system.

CHAIR POWERS: Yes, I will give you that. It is an unusual little system.

DR. WITTER: Thank you, Mark. Moving on to kind of stepping out for the arrangements of the excore and incore instruments.

Starting off with excore, the large circles ranged at four locations are the locations for the intermediate and power range detectors. There are four intermediate range detectors and then eight total power range detectors with two elevations axially for the power range detectors in the same locations.

For the source range detectors, they are located, there are only three of those located at 90-degree positions, indicated by the triangles in the figure.

Within the core map on the left-hand side, it shows the location of the control assemblies, which are kind of like the larger circles on top of fuel assembly arrangement there. And then the lances, there are 12 lances, as Mark mentioned, that are inserted into to provide the coverage for the AMS system and the SPNDs. Each lance will have one SPND string off of it and then a series of different AMS tubes off it.

Again, these have been used in the German convoy plants for several years. And in the next figure, I will have a little bit zoomed in on where the SPNDs are located.

The right-hand figure shows an axial elevation view of the arrangements for excore detectors, the lances system coming in to be inserted in, as Jeff Tucker mentioned, into some of those instrument guide tube regions. Whether they will fit into one location or two locations, depending on whether or not the AMS system is also where the SPND is located.

There are also upper -- well, let's see. Their core outlet temperature, thermal couples also associated with each lance. And then the side view also shows where the penetrations for the control assemblies, some water level measurements, and some other thermal couples in the upper head are located but that is not part of this discussion here.

The next figure shows a little bit more refined view of the Self Powered Neutron Detectors or SPNDs, also noted as the Power Density Distribution System. In this case, there are 12 radial locations and six symmetric pairs of SPND strings located through the core and then there are six axial elevations to provide the coverage for the axial power distribution monitoring.

They are cobalt detectors to provide near instantaneous response. In this case now with the U.S. EPR, the incore detectors are being used for signals to provide for the protection system, as well as just normal monitoring. And so a fast response was required and therefore the choice was going to use a cobalt detector rather than a rhodium detector.

MEMBER STETKAR: Are you going to explain what protection signals actually come from them?

DR. WITTER: Yes, I have a slide in a little bit that hopefully will describe it a little bit how we use these, how we calibrate them and then how it is used within the operational philosophy.

MEMBER STETKAR: Thank you.

DR. WITTER: If I don't answer it, ask away.

So, the I and C functions both from a reactor control surveillance and limitation function or the non-safety functions, we have a view into the departure from nuclear boiling, the linear power densities or linear heat generation rates and then the axial power shape monitoring.

For protection system functions, we

monitor the DNB and the linear power density or linear heat rates. And the detectors are 21 centimeters in length and distributed axially to provide coverage.

How we actually then use these signals, the AMS systems is used as part of the calibration for the incore detectors, where the power distribution if mapped, combined with the power tracks core neutronics monitoring to develop the full 3-D map of the core power distribution. From that power distribution then the SPNDs are calibrated in three separate ways. So there are three sets of calibration functions for DNB, for linear power density, and then also for the axial power shape. Then the design limits are monitored.

Much of this is developed and explained in the topical report for the static setpoints and transient methods. ANP-10287P, which is in the process of getting the final safety evaluation report from the staff.

So explaining a little briefly about the calibrations. To start off, the DNBR monitoring. In this case, each of the 12 strings is calibrated to indicate the linear heat generation axial distribution for the full span of the assembly or the pin fuel rod of the DNBR limiting fuel rod.

So all 12 locations will then have

calibration sets such that once finished calibrating they will read the linear heat generation rate of that fuel rod in DNB. Then the DNB will be estimated using, after calibration, then the DNBR is constantly monitored by making use of plant system indications for inlet temperature, the mass flow rate, based on the reactor pump speed and then the system pressure, combined with the power distribution or the F delta H indication of the axial shape of the SPND readings. That gets fed into the DNBR algorithm, the online algorithm, making use of the ACH2, the CHF correlation and that provides a sense DNBR for the reactor, both for monitoring, for surveillance, and then also for the protection system.

The protection system would then, is more described in Chapter 7 for the INC functionality aspects. But essentially through the four divisions of the sensing of the DNBR, two out of four second minimum will cause a reactor trip, once it reaches the setpoint threshold level, if it reaches that point.

IN the next figure I show a little bit about the kind of like the holistic view of the layered protection of towards the specified allows fuel design limits.

The next set of calibrations that is done at the same time at the time of calibrations is the high linear power density monitoring. In this case, at each of the six axial slices in the core, all the SPNDs, all 12 at that axis slice are calibrated to read the maximum linear heat generation rate at that elevation. So now we are looking at the local peaking and the local linear heat rate, the maximum local linear heat rate. So, essentially monitoring the FQ at each axial slice.

These then are within the protection system and the control system are compared against the setpoints for the linear heat generation rate, which those are meant to protect the fuel center line melt, total clad strain and then any other abnormal operational occurrences, so that we would avoid violating the fuel specified allowed fuel design limits.

These signals then are also used as an indication of any sort of azimuthal imbalance that may occur within the core. If the total difference within the core elevations reaches a certain threshold, it triggers the fact that there is an imbalance being indicated, therefore, you need to adjust the threshold by which you are measuring the DNB. Essentially, if an imbalance is detected within out setpoint methodology, there are a series of different DNBR trip thresholds. And if an imbalance is indicated, it will trigger into a different higher DNBR setpoint for the reactor trip, essentially.

So the linear power density monitors are used both for the fuel centerline melt monitoring or the FQ and then also to provide an indication of imbalance, such that we can cover that imbalance within the DNBR monitoring.

The last and both the DNB and the linear power density are both used in the reactor control surveillance and limitations non-safety system but also in the protection system. The axial power shape monitoring is only used in the reactor control surveillance and limitation or RCSL system as a non-safety measurement to help provide the operator an aid and automatic control for the axial power shape.

In this case now the third set of calibrations is performed where the SPNDs in the upper and lower half of the core are calibrated such that the axial offset or axial power shape index is measured and indicated from those detectors.

And by basically splitting the top half minus the bottom half and coming up with a ratio fraction, to provide a fraction percent of the core, power shape on the upper and lower half of the core. It is a very course measurement. And typically that is also used with the excore detectors. They are also calibrated to provide an AO signal as well. But for the control and for the surveillance system, the incore detectors are used for the axial offset indications.

MEMBER STETKAR: So in this plant, in any automatic mitigation of axial offset comes from the incore.

DR. WITTER: Comes from the incore detectors, yes.

Okay, as I mentioned, there is sort of a holistic of the protection of the fuels, Specified Allowed Fuel Design Limits or SAFDLs. And this is outlined in great detail in the topical report ANP-10287, the "Incore Trip Setpoint and Transient Methodology." The application of the methodology was then used to verify for the AOO or abnormal operational occurrences within the Chapter 15 safety analysis for all the different events.

How the reactor controls surveillance and limitation system works, there may be a little confusion when we say LCO1 and LCO2. It is not meant to counter or replace the technical specification limiting completely for operations but are meant to aid the operating in ensuring that he maintains the reactor within the tech spec LCOs. So it is a nomenclature aspect of how the RCSL system functions.

In a traditional plant, you have LCO levels from your technical specifications. You then have the next layer of the LSSS or limiting safety systems settings or reactor trips and those are ultimately meant to assure that you do not violate the SAFDL.

For the U.S. EPR, there is more of a series with the automatic control system where essentially LCO1 would function as a more passive feature to provide alarms to the operator, will block rod withdrawals and also block any sort of turbine load increases. If you weren't at 100 percent power and you were coming up on a maneuver, it would block that turbine increase and say hey, wait. You are going to violate your LCO or potentially bring you into a condition where you may take a reactor trip.

LCO2 is at the next level of protection where if the operator didn't pay attention to the alarms or hadn't prevented the alarms in the first place, then a little bit more active participation occurs with the control system. Where now the turbine generator will run back a certain amount of power until the LCO condition clears out of the control system and then it will then also insert control rods to match that turbine generator run back. And also of course, you have alarms come in on that actuation.

If for some reason that those actions both from the operator and from the control system do not mitigate the DNBR or LPD response, then the next level is the limitation setpoint or known as our partial trip, which provides a rapid power runback with the scram of a few selected rods from the decontrol bank. and that is meant to rapidly reduce the core power and rapidly run the turbine generator back to a point where you are better assured of being in a much lower power level, essentially running the power back to 50 percent power, so that typically your linear heat generation rates and DNBR are not an issue at the lower power levels.

MEMBER STETKAR: You go back to 50 percent?

DR. WITTER: Somewhere around 50 percent, yes.

MEMBER STETKAR: Is this -- I haven't seen Chapter 7 yet and I haven't seen the topical report, but in a nutshell, is this very similar to the Konvoi control system?

DR. WITTER: Yes.

MEMBER STETKAR: Okay.

DR. WITTER: Yes, very similar except for now with the protections --

CHAIR POWERS: We understand the Konvoi plants are the ones that the Germans want to shut down.

DR. WITTER: Well the German utilities would like to keep them out.

MEMBER STETKAR: There are a couple of them outside of Germany.

CHAIR POWERS: I'm just trying to understand. That's all.

DR. WITTER: Yes. Essentially, their control system has a rapid power runback or partial trip, I think more on their pellet-clad interaction linear heat generation rate monitoring for the non-safety systems and it provides that partial trip functionality.

The new feature then is as an adaptation and an evolution is to now bring it fully into the protection system. So now if you have exceeded, if the partial trip still didn't work or if for some reason the RCSL system was not in function, the protection system is a separate monitoring of the DNB and LPD signals. So that RCSL can be out of service but the protection system will still be in service to provide if LCO1, 2, and LSSS limitation were surpassed, either due to operator error or the evolution of the AOO events, and RCSL was out of service, then the protection system would kick with a normal function of providing a full reactor trip if the DNBR or the LPD exceed the thresholds.

And the DNBR has a symmetric radial power shape threshold level. As I mentioned before, there is an imbalanced threshold, which is a little bit higher in DNBR and also the settings are set to accommodate certain numbers of SPND string or SPND failures, whereby the more SPNDs that are out of service, the higher and higher the DNBR thresholds go or the lower and lower the LPD thresholds go for a reactor trip.

MEMBER STETKAR: And on this plant, from what you just said, just to make sure I understand, there are separate DNBR trip. There is a separate DNBR trip and an LPD trip?

DR. WITTER: Yes, separate trips. Yes.

MEMBER STETKAR: Thanks.

DR. WITTER: And the LPD trips will occur on any one of the 72 SPNDs on a second max, two out of four trip from the divisions. So we have 72 SPND locations.

MEMBER STETKAR: Right.

DR. WITTER: Any one of those 72 reaching the maximum and then taking the second maximum reading out of the two divisions will cause the reactor to trip.

MEMBER STETKAR: But it is on any one of the 72?

DR. WITTER: Yes. Now, for DNB --

MEMBER STETKAR: That could be very, very localized.

DR. WITTER: Yes. Yes, exactly. So it is meant to protect the local peaking and the fuel centerline melt and clad string.

For the DNBR, it is based on the 12 strings because it is an axial integrated enthalpy rise. Let's see. I think that kind of covers the view of how the protection system makes use of the incore system but then also making use of the DNB correlations, tying it all back to the detailed thermal hydraulics calculations and the codes and methods.

Moving on to the later sections of Chapter

4, it gets into the reactor materials used. And for here the control rod drive mechanisms, they are all pressure boundary materials and they are designed within the confines of the ASME code. The materials that are used for the pressure boundaries are austenitic stainless steels, martensitic stainless steels. And these are addressed within the chapter 5, which we will hear a little bit more about this afternoon.

For the non-pressure boundary materials within the housing, again using austenitic stabilized steels and martensitic stainless steels, there are some cobalt-based materials where in small portions of the assembly where better where is required, primarily in the latching mechanisms for the drive mechanisms. So the decision was used to reduce the amount of cobalt in the system so you avoid the cobalt activations.

MEMBER SHACK: Just out of curiosity, when you -- You call out both English and metric specs on some of these materials. When you call out a metric spec, does that mean you have picked a European supplier?

DR. WITTER: Let's see, our materials person.

MS. SLOAN: We are bringing up one of our materials persons. Sarah Davidsaver.

MS. DAVIDSAVER: Good morning. My name is Sarah Davidsaver. I have been working at AREVA for four and a half years. I have been in the nuclear industry for nine years. I have a Bachelor's degree from Virginia Tech in material science and a Master's degree from University of Virginia in material science.

The materials in the internals non-pressure boundary portions of the CRDM are European materials.

MEMBER SHACK: Okay, so they are given a European spec because you will get them from there.

MS. DAVIDSAVER: Correct.

MEMBER SHACK: The Europeans don't have any limitation on the carbon level in the 347, beyond that in the ASME/ASTM spec, the 0.08 level?

MS. DAVIDSAVER: We actually have an RAI question right now that we are responding to where the staff has asked us that question.

MS. SLOAN: We are not ready to answer that at this point.

MS. DAVIDSAVER: Right. We are currently working on that.

CHAIR POWERS: Go ahead, Bill.

MEMBER SHACK: No. If it is evaluated it is evaluated.

CHAIR POWERS: I mean aren't you surprised a little bit that for these high-wear locations they are still using cobalt alloys and not using something like an ion-implanted alloy or something like that to get the hardness they need?

MEMBER SHACK: Well that was sort of one of my first things when I looked at it is I was sort of surprised there was as much Stellite still in the system. But I am assuming people are comfortable with Stellite, you know even though people have spent a lot of time and money looking at non-Stellite materials. But you know, it is their reactor.

CHAIR POWERS: Yes, and neither of one of us have to go in and do any maintenance on this stuff.

MEMBER SHACK: Right.

CHAIR POWERS: I would be interested in why. I mean, is there a reason? I mean, if hardness is your only criteria, there a lot of materials that are really, really hard.

MS. DAVIDSAVER: We have tried to reduce it as much as possible but it is the best material for the application.

CHAIR POWERS: The ion-implanted materials won't do it for you?

MS. DAVIDSAVER: I can't answer that right now.

CHAIR POWERS: Okay.

DR. WITTER: Okay?

CHAIR POWERS: Oh, they are not that expensive. I mean, you have to take an integral over the cost of the material and the amount of worker dose you are going to accrue here.

DR. WITTER: The next slide talks about some of the materials used for the reactor internals and the core support. Again, the materials were selected based on the compatibility with the coolant environment, that they be exposed to and the radiation environments.

For the coolant materials or exposed to the coolant, they are made of corrosion resistant materials. The components for the internals are non-pressure boundary though the materials are ASME, designed to the ASME code, again, made mostly of austenitic stainless steels. There are some components that do again use the Stellite material for the wear resistance.

And then the support pins and bolting for the heavy reflector and portions of the other internals are mostly the low-carbon austenitic stainless steels. Again, materials that are consistent with current applications in the United States and Europe.

And then the last section of Chapter 4 deals with the design of the reactivity control systems; whereas it describes primarily it points to other sections in the FSAR for some of the actual design features. But indicating that we do have two independent reactivity control systems for normal and abnormal conditions, where the control rod drive system is used to compensate for the fuel and water temperature changes that occur with the fuel depletion and any impacts on the plant system, transients.

It also limits the maximum reactivity insertion rates by limiting by the design of the control bank configurations and also the maximum rod speed, which is about 29 inches a minute. They also assure that we maintain minimum shutdown margin during AOOs with one control rod stuck out of the core.

For the chemical and the volume control system, that is essentially our boron, our boric acid control system where we use soluble boron for the fuel cycle depletion and the xenon burnouts and there are limits on the rate and the duration of the dilution or boration. Again, limiting the reactivity addition rates.

And then two systems that are described more in Chapter 6 are safety injection system for LOCA and then also the extra borating system for cold shutdown. Again, these provide our two independent reactivity control systems for shutdown and for control.

MEMBER STETKAR: Jonathan, we are okay on time. So in the SER and in the FSAR there was some discussion about your protections again boron dilution events in the CVCS system. As I understand it, there is a setpoint for boron concentration that is manually set by the operators and that if boron concentration decreases below that setpoint, you isolate both letdown and charging flow form the VCT. Is that correct?

You are looking around but you are not a systems guy.

DR. WITTER: I am looking around the room to see who we have here.

MEMBER STETKAR: You are not the systems guy.

DR. WITTER: Right.

MS. SLOAN: Unless there is somebody that can answer it, we may need to defer that, --

MEMBER STETKAR: Okay.

MS. SLOAN: -- particularly to the Chapter 6 discussion, where you have the system engineers here.

MEMBER STETKAR: Well it is actually Chapter 9 but it was discussed an awful lot in the context of boron dilution events in this chapter.

In particular, the thing that I am concerned about, I think I understand how it works during normal power operation because I sort of understand how the system works. There is some discussion about how that protects -- I have to be careful because it is not a protection feature, how that automatic isolation feature protects you against boron dilution during shutdown modes, in particular Modes 5 and 6. I think the staff had some questions about that.

My curiosity is that the discussion talks about isolation of the normal letdown line as one of the features that protects you against these dilution events. During Modes 5 and 6, it is not at all clear to me that the normal letdown line is actually in service. Most typically plants have let down a line to the RHR system. I noted in this plant RHR Trains 3 and 4 indeed do have letdown connections and, in fact, the description of those letdown connections says that they are used during shutdown. So it is not at all clear how the described protection against boron dilution during cold shutdown actually protects you against boron dilution during cold shutdown if you are not using the lines that are actually isolated.

There is one valve that may, from the charging side, from the water injection side, protect you but I got kind of confused trying to understand how the system works in reading the discussions in the FSAR. Now the description is actually in Chapter 9, so admittedly, we are kind of across issues here between Chapter 9 and this chapter. But a lot of the discussion of the protection in terms of reactivity control and hydraulic control was in here. That is why I asked the question in this context just to see if there is any.

MS. SLOAN: And what I am doing is taking down the question so that when we get to that point in the chapters.

MEMBER STETKAR: Take down the question and when we get to Chapter 9 I hope I remember it also. I was just curious if you had anybody here who said, oh, yes, yes. We understand how that works.

Take the notes. I am not so interested during power operation. I am more interested in the configuration during shutdown modes and how it provides protection.

Thank you.

DR. WITTER: Okay. So just to summarize what Jeff Tucker and I have presented and a few other participants with their aid have discussed the sections of Chapter 4, where we have provided an overview of the reactor system and providing indications that a lot of the design is using our existing methods and incorporates some of the new features of the design for the aeroball measurement system, the incore monitoring system, the heavy reflector, adapting those methods or generating new topical reports to apply those methods for the reactor design.

CHAIR POWERS: Okay.

DR. WITTER: And that concludes our presentation.

CHAIR POWERS: Any other questions on this Chapter 4 presentation?

(No response.)

CHAIR POWERS: Okay, well we have a few

listed for future resolution and we will get to those. What I propose we do now is we go ahead and take a break for 15 minutes and come back and listen to the staff's SER on this. Is that okay with you?

MR. TESFAYE: Yes.

CHAIR POWERS: Okay, let's break until 10:35, according to that clock.

(Whereupon, the foregoing proceeding went off the record at 10:18 a.m. and went back on the record at 10:34 a.m.)

CHAIR POWERS: Let's come back into session.

MR. TESFAYE: Good morning again. I would like to introduce Jason Carneal who is the chapter lead for Chapter 4, Chapter 6, and Chapter 16. You will be seeing a lot of him on your own.

CHAIR POWERS: Another guy that drew the short straw here. Chapter 4 not too bad but 15? Oh, my God.

MR. CARNEAL: All right. Thank you, Getachew. My name is Jason Carneal. I am the NRO project manager responsible for coordinating the staff review of Chapter 4 of the U.S. EPR design certification application.

I have a BS and MS in engineering

mechanics from Virginia Tech. From 2004 to 2008 --

CHAIR POWERS: We seem to have a lot of people from Virginia involved in these particular chapters. I mean, is there some trend I am missing here?

MR. CARNEAL: Well, it is pretty close by and the job opportunities in Virginia are slim.

(Laughter.)

MR. CARNEAL: From 2004 to 2008 I was a mechanical engineer and the hydrating mechanics director at the Naval Surface Warfare Center Carderock Division, where I performed experimental studies on the hydrodynamics of naval vessels. I joined NRC in November 2008 and have since served as Chapter PM for Chapters 4, 6, and 15 of the U.S. EPR design review.

The NRC technical staff involved with review of Chapter 4 are Fred Forsaty, John Budzynski, and Shanlai Lu from the Reactor Systems, Nuclear Performance, and Code Review Branch. And for the Section 4.5, Reactor Materials, we have John Honcharik and Robert Davis from the Component Integrity Branch and we will rotate them in as we get to their respective sections.

During this meeting, the staff plans to make a presentation of their review of Chapter 4, including a safety evaluation report with open items. Chapter 4 covers fuel system design, nuclear design, thermal-hydraulic design, reactor materials, and functional design of reactivity control systems.

The U.S. EPR FSAR Chapter 4 SER with open items was issued as publicly available on February 17, 2010.

The staff has issued 104 questions to the applicant requesting additional information during their review. Out of the 104 questions asked, there are 14 open items identified in the SER with open items. I should mention that there are 14 unique open items that were issued as part of the review. Several of the questions affect multiple aspects of the safety evaluation, which I will get into on the next slide.

The following three slides contain a general overview of the open items contained in the SER with open items for U.S. EPR Chapter 4. I would like to note that the first open item listed in this slide, RAI 339, Question 04.02-17 tracks the open review of topical report ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report." This open item is referenced multiple times throughout the safety evaluation report. However, there are three remaining technical issues with the technical report that will be discussed in detail by the technical staff during this presentation.

The path forward for this open item is that there are draft requests for additional information outstanding on this topical report and the staff is currently planning to complete the SER for the fuel assembly mechanical design topical report, during Phase 4, which is currently scheduled in June 2010. And we will discuss the remaining open items that are summarized on the following three slides with the technical reviewers as we get to their respective sections.

The next slide lists a summary of the open items, specifically with regard to reactor materials. And this slide contains an overall summary of questions asked on Section 4.6.

In the interest of time, we would like to get straight to the detailed discussion and I would like to turn the presentation over to the technical reviewer for Section 4.2, Mr. Fred Forsaty of the Reactor Systems Nuclear Performance and Code Review Branch.

MR. FORSATY: All right. Good morning. My name is Fred Forsaty. I am a little more than four years with the Agency at the Office of New Reactors.

I started my education in Madison, Wisconsin. I got a Bachelor's in chemical engineering and a Master's in nuclear. And then I made a tour of the some of the universities in California. I ended up at Penn State.

CHAIR POWERS: Nothing in Virginia?

MR. FORSATY: No.

CHAIR POWERS: Sorry. Okay. I want some diversity here.

MR. FORSATY: I had an admission from Polytech but I didn't go. I ended up at Penn State University and spent five or six years at the school. My interests and background includes thermal hydraulics, neutronics fuel management. And after school I started building simulators for Arizona Power Plant BWR and PWR. Then I moved on an worked as a systems engineer at Beaver Valley, Yankee Atomic, Niagara Mohawk, ended up at the D.C. Cook. Spent four or five years there during the shutdown and then I am at the Agency.

CHAIR POWERS: One of us had a tour of duty at D.C. Cook don't we?

MR. FORSATY: Section 4.2, right now we have two open items. One is related to Seismic LOCA and the other one is pointed at the topical report. What we have done here is we had put together some of the remaining issues that we have in the topical report that give you some information on that.

On the seismic-LOCA, the applicant has stated that no crushing deformation will occur during normal operation and operating basis earthquake conditions based on the crush data and test results. And also that with the load limitation as the 95/95 one-sided confidence of the mean elastic limit for grids at beginning of their life condition, they are correcting for the operating temperature.

Here, the open items basically in our open item we are not questioning the methodology. However, we have some questions on the methodology to document the applicability of the methodology that they have used, which is an approved methodology for the operating plant. And we are doing that as a requirement or as SRP Guidelines Section 4.2.1.

Now I am going to move to some of the topics that were remaining, technical issues on the topical report for the fuel. And I would also like to recognize Mr. Carl Beyer. He is with PNNL and he is supporting their staff and fuel issues, related issues, and he is present here today.

M5 growth issue. The current plants with M5 guide tubes have experienced much higher than expected irradiation growth, resulting in gap closure between the top nozzle and the core plate during cold shutdown creating assembly bow and control rod becoming stuck in the guide tubes.

CHAIR POWERS: Why does the M5, why does it experience this growth?

MR. FORSATY: The experience with the growth?

CHAIR POWERS: Yes.

MR. FORSATY: We have spent quite a lot of time on this issue reviewing the information that is provided to us. And at this point, we are still reviewing the result of the information and we also have looked at the root cause and root analysis. You know, one reason that we are experiencing this growth issue could because of stress.

But again, as we speak we are going through all of the available information to make a determination and a position on that.

Did I answer your question?

MR. TESFAYE: Maybe Carl Beyer can expound on this?

MR. BEYER: Yes. Why it is experiencing more growth than --

MR. FORSATY: Carl, could you please give your background.

MR. BEYER: Yes, excuse me.

CHAIR POWERS: Yes, we have got to do this. Yes, we just like to know who you guys are.

MR. BEYER: Sure and for the record. Right? Carl Beyer, a BS and MS from Washington State University and I have been in this business for nearly 40 years now. And so, and consultant to NRC for about 35 of those 40 years in the field performance area.

CHAIR POWERS: You still haven't and straightened them out.

MR. BEYER: We are still working on that. It is a non-stop process.

In regards to why M5 is growing, perhaps a higher growth at equivalent fluence levels as compared to Zirc-4 and particularly for the implied growth, it looks like it is stress-related. And each assembly design has different guide tube stresses on them. And so you can get different growth, depending in what your assembly design is on the stresses.

But in addition to that, it still looks like at least for M5 anyway, for guide tubes, they seem to be growing, for equivalent stresses, seem to be growing at a higher rate than for Zirc-4.

CHAIR POWERS: Is it associated with the

alloy or is it something else?

MR. BEYER: Well, yes. And AREVA has been doing a lot of SEM/TEM work in order to try and figure out why. Because theoretically, growth happens due to dislocation loops and their arrangement and density. And right now, though, based on their experience, they haven't been able to determine the exact why M5 is growing differently than the Zirc-4 from dislocation densities or orientation at this particular time. They are still spending quite a bit of effort in order to try and determine that.

CHAIR POWERS: My limited familiarity with these niobium alloys is, like with the E110 is nominally the same as M5

MR. BEYER: Right but phase differently.

CHAIR POWERS: Phase radically differently. And so they are very strange. Very weird.

MR. BEYER: Right.

CHAIR POWERS: And of course that raises all kinds of questions. You know, does the manufacturing process yield the same product every single time and things like that.

MR. BEYER: Right. And then in some cases, minor impurities can make a difference.

CHAIR POWERS: Well, what you know about zirconium is that it is the one alloy where all these electronic state rules actually work to tell you something.

MR. BEYER: Yes, and we are still learning a lot in terms of the theoretical behavior.

CHAIR POWERS: Yes, okay, well good. I'm glad I am not the only one that is completely in the dark here.

MR. BEYER: Exactly.

CHAIR POWERS: I am more in the dark than most but at least I am not alone.

MR. BEYER: Exactly, yes. But AREVA is still working on the --

CHAIR POWERS: Go ahead.

MR. FORSATY: As Mr. Beyer indicated, we are still pursuing some of the information that has been provided by AREVA. We are in almost weekly or monthly contact with them and --

CHAIR POWERS: I seem to be in the same state. Spending altogether too much time with them. They can put me on the payroll, I think, pretty soon.

MR. FORSATY: We are making progress and we have Mr. Carl Beyer here to guide us through this. So we are very fortunate on that.

Just to give you a little background on the staff evaluation. The staff have issued several questions related to the applicability of M5 growth data from the current fuel designs, the EPR design. And we also have asked question on the growth uncertainties and on the end-of-life gap closure analyses.

At this point, we have requested an accurate growth limit that would use more the higher burnup data rather than a combination of low and high burnup data. And we are also expecting to get that information within a short period of time.

CHAIR POWERS: Okay, good.

MR. FORSATY: Anymore question on growth?

The next remaining technical issue on the fuel topical is the cladding strain. The COPERNIC code is used by AREVA to determine the linear heat generation limit that meets the one percent cladding strain for the AOOs.

We have then audit calculations and a confirmatory analysis using FRAPCON. FRAPCON is the computer code that PNNL is using and Mr. Beyer has helped us with that.

And the conclusion that we have right now at this point is that the FRAPCON which is used as a confirmatory code that predicts a lower linear hat generation limit compared to the COPERNIC, using the same input for both computer codes. Further examination of the data used to verify the COPERNIC code had demonstrated that it also under-predicts all of the strain data for power transient for a time period on the order of seconds to minutes for the AOOs while over-predicting time periods in the order of hours.

On the staff evaluation, we have issued several questions for further COPERNIC data and code comparisons and also are in the process of looking into conservatism exists or AREVA was claiming exists in the calculation to support the one percent strain limit for M5 cladding.

If there are no further questions, we will go to the next remaining issue for the topical report. That is the M5 hydride limit. We do not consider that as significant as the other ones. The applicant has responded that a hydrogen limit is not necessary for M5. However, this is not consistent with the SRP Guidelines.

In our evaluation, we have requested that a hydrogen limit be proposed by AREVA to justify this. So it would be Guideline Section 4.2.

CHAIR POWERS: In the presentation this morning, it was indicated in that presentation that M5 has a lower hydrogen pickup rate than the zircaloy. But I mean, it still pick up hydrogen. And we know that now we learned from the investigations conducted at Argonne that hydrogen operates synergistically with oxygen to embrittle in DBAs.

So yes, I mean, it seems like hydrogen is important here.

MR. FORSATY: Yes, the data that is provided by AREVA supports that both corrosion and hydrogen remain significantly below the corrosion and hydrogen limits. So there is quite a good amount of margin to the limits. But again, we are pursuing that issue and Mr. Beyer is supporting us on that.

MR. BEYER: Yes, the reason why the standard review plan in the last update, it was stated in there that a limit needs to be established based on issues with --

CHAIR POWERS: What kind of --

MR. BEYER: Both for normal operation because you get hydrogen embrittlement if you get enough hydrogen in there due to normal operation but also due to the LOCA phenomena.

CHAIR POWERS: What kind of hydrogen

pickups do they get for fuel that has gone up to you know, 50 gigawatt-days?

MR. BEYER: Actually it is pretty low; 100 ppm or less.

CHAIR POWERS: Well, that is low.

MR. BEYER: Yes, it is. Right. It typically is in its soluble range during normal operation because soluble range is around 85 ppm, 90 ppm of hydrogen.

So there really isn't a problem here with the M5 with hydrogen but the SRP states that a limit should be established.

CHAIR POWERS: Okay. Okay, I understand. Yes?

MR. LANDRY: Ralph Landry from the staff. We have seen one sample that we tested Argonne of M5 at 63 gigawatt-days exposure, a sample that absorbed 110 ppm of hydrogen. We looked at samples of Zirc-4 and ZIRLO, both of which had 68 gigawatt-days that absorbed 550 ppm of hydrogen.

So the hydrogen absorption was significantly less with the M5.

CHAIR POWERS: And I shall probably go to my grave not understanding exactly why.

The other issue is how uniform is the

hydrogen pickup in the alloy?

MR. LANDRY: Well, in operation, the difficulty is measuring the localized hydrogen because we were measuring 110 plus or minus. I think it was 20 ppm hydrogen circumferentially around that particular thing that we are examining. And with the other materials, it was 550 plus or minus. It was more than 100 around the circumference.

So the only way you could measure it was to take an example and melt it. Once you have done that, you can't do any more testing.

CHAIR POWERS: Well you can, but it is just not anything that you are interested in.

MR. LANDRY: It doesn't respond to ductility.

CHAIR POWERS: Well I bet it is really ductile after you have melted it.

MR. FORSATY: If you don't have further questions on 4.2, we will move to 4.3.

On 4.3, I would also like to recognize two of the NRC staff that are supporting us in review of this 4.3, Mr. Jack Rosenthal and Mr. Lambrose. Mr. Lambrose has quite a lot of experience in fluence and we have used his expertise to support us in making decision and determination on the fluence.

On fluence, the applicant -- we have two open items on 4.3. The first one is on the fluence calculation methodology. The applicant has provided a generic fluence calculation methodology for the U.S. EPR design. This methodology basically relies on the approved methodology for the operating plant and the U.S. EPR incorporates also the -- The difference between the operating plants and the U.S. EPR is the heavy reflector, which is an evolutionary change from the operating fleet.

On our evaluation, until we have determined that until the first measured fluence value is available, the actual vessel fluence is only a small fraction of the end-of-life fluence. Basically what we are saying is that the amount of fluence that is going to be accumulated is conservatively low and compared to the operating plants.

The NRC-approved vessel fluence methodology is described in BAW-2241 and that is approved by the NRR in April of 2006. And that is the basis for the fluence calculation that has been submitted to us for the EPR design.

The calculation for the EPR design meets Reg Guide 1.190. And with the exception of benchmarking of the methodology for the U.S. EPR design.

And the next slide. The staff approved BAW-2241, based on surveillance capsule and dosimeter data from operating reactors. The approved version of BAW-2241 requires the use of surveillance of the capsules and dosimeter data points to verify the applicability of the methodology to any particular plant.

So basically, we need to do some kind of benchmarking at some point to ensure that the methodology and the calculations are applicable. WE don't have data. There are no operating plants right now. And based on the information available and the support of Dr. Lambrose and his expertise, we believe that there should be no issue getting the plant started.

And then however, in contradiction of this, we would have a COL item. The COL items would require AREVA to, after the effective ten full power days of operation, they will take the capsules out, doe their measurements and verify that their methodology and the calculation they have submitted to us is still applicable and within a reasonable range of the test.

I think you had some question initially on these fluence level. There are some information in

Section 4.3.8 that the relative flux levels are documented. And Mr. Lambrose did a hand calculation. I think he would be able to give you a better information on what the flux level should be. Mr. Lambrose?

MR. LOIS: Yes, I am getting a microphone. I am Lambrose Lois and I am from the staff. My education, I came to this land, it has been about 35 years or so. I came through academia, University of Maryland, University of Athens, Greece. I got a degree from the University of Athens in mechanical engineering, a Master's from Stevens, and a Ph.D. from Columbia University in New York City.

There was an open question as to the value of the fluence of the vessel. That is about 1.5 10 to the 19th. That is calculated by the methodologies of BAW-2241, which is an approved methodology for conventional plants, dating plants for BNW and other PWRs.

However, the provisions in Reg Guide 1.190 provide a benchmarking for all codes that use the calculation in fluence. Now there is of course this difference between the conventional BWR plants with the EPR. And that is, of course, the heavy reflector around the core.

However, there are several conservatisms in the value of the fluence with respect to the embrittlement level provided in 10 C.F.R. 50.61. And there maybe several reasons for that.

I don't know if you want to go into it but the major reasons, one of the major reasons the material does not contain the culprits for embrittlement, namely, copper and nickel. So therefore, the different rate for the unit of fluence is extremely small. That is one of them.

The other one of course is the presence of the heavy reflector, which is currently one MeV neutrons away from the vessel. And of course, they don't count anymore. And there are several other minor factors that contribute in that direction.

So the value of 1.5 10 the 19th with respect to embrittlement, which is measured in terms of delta RTNDT is huge. Because this vessel does not have axial welds, the criteria is 360 degrees Fahrenheit for RTNDT or RTPTS, if you wish. And the calculated value at the end-of-life for 60 effective years, which of course is not going to have because 60 calendar years of its life is not the service life. Anyway, it is going to be much smaller than that.

So to give you a sense, it is that the

calculated embrittlement the RTNDT is about 100 degrees, the worst for material. And that is quite a ways away from the 300 degrees provided by the 10 C.F.R. 50.61.

Now I want to give you a sense of how far this is is that one of the previous reactors we had RTPTS issues for a license extension from 40 to 60 years, it would take, we would need about 88 degrees Fahrenheit to get the last 20 years.

So if the value here is 100 degrees and the criteria is 300 degrees, the conservativeness is humungous.

Thank you.

CHAIR POWERS: Thank you.

MR. FORSATY: All right. The last item here is on the rod ejection. This is basically a pointer to a topical report.

You know, in general when it comes to neutronics, neutronics is really a standard. No matter what core you look at the neutrons is the same. If you put them in an APWR design or an EPR design, in general, 90 percent of the time they would behave the same.

We have not seen anything different, you know, in the EPR design when it comes to neutronics.

However, we have an open item here that points to the rod-ejection topical report. On the 4.3 side, we calculate the rod worth, the ejector rod worth and that information is given to the topical report or to the rod ejection group and then they would do the calculation. The only reason for this open item is they are waiting for the completion of that topical report.

And that concludes my presentation of 4.3, if you don't have any questions.

CHAIR POWERS: Any other questions?

MR. CARNEAL: For Section 4.4, the discussion will be led by Mr. John Budzynski from the Reactor Systems Nuclear Performance and Code Review Branch.

MR. BUDZYNSKI: My name is John Budzynski and my experience goes back to two years working on the Advanced Test Reactor and about 25 years with Pico Energy, which is now Exelon. I worked at Peachbottom as the reactor system engineer.

And my education is a BS degree in nuclear engineering from the University of Maryland and a Master's in mechanical engineering from Drexel University.

And I would like to say that Dr. Jose

March-Lueba assisted me on this section here, 4.4.

Okay, the first slide. We reviewed the hydraulic loads analysis and we thought that we needed additional information. They identified that the reactor coolant pump over speed transient was a limiting event. When we asked for additional information, we reviewed their information and we decided that we needed a follow-up RAI for them to give us a full description of the hydraulic load analysis for just he fuel ended vessel components for normal operating conditions and for design basis accident conditions. And we are waiting for that response. Next slide.

Most of this was went over, AREVA actually described most of this. But the one thing I wanted to note, to state in here is that the older plants, they didn't use the SPNDs to part of their instrumentation for scram. It is used now in the EPR system before us.

Setpoint methodology, the ANP-10287P SER was evaluated and it was found to be acceptable under the ANP-10287P.

MEMBER STETKAR: John, it is not problem using non-safety systems to provide the calibration data for those safety-related trip functions?

MR. LU: This is Shanlai Lu from the staff. And I just give you background before I jump into that.

In terms of education, I got my Bachelors degree in mechanical engineering, Master's degree in nuclear, and Ph.D. in nuclear from Penn State. And several years with the industry and teaching in the university for three years. And I joined the NRC ten years ago. I worked on GSI-191, you know, research, and then you know.

Back to your question related to the safety system, we actually have probably one slide related to the classification on that point.

MEMBER STETKAR: Okay.

MR. LU: And in the aeroball system it is a calibration system. The purpose is to calibrate and it is not a part of the trip setpoints, online trip setpoints systems based on last PRD.

So calibration time interval does not require you do that online and like every two minutes or three minutes. So therefore, that is classified as a non-safety system for calibration of SPND. However, there is a limitation on that.

And then for the pressure boundary part of the AMS system, we still define that as the --

MEMBER STETKAR: No, I understand that. I was just curious. I am not personally familiar enough with the use of non-safety equipment systems to actually calibrate the instrumentation that is used for protection functions. That was my basic question. I need a little education.

MR. MARCH-LUEBA: I'm sorry. I apologize you won't have to wait for my interaction. I am Dr. Jose March-Lueba from Oakridge National Laboratory. I am an NRC consultant. I have a Ph.D. in nuclear engineering from the University of Tennessee and I have been working on a variety of topics. One of them is I was hired Instrumentation and Controls Division in Oak Ridge National Laboratory.

Let me give you a mental picture of what you are asking. Are you familiar with the term Class 1E?

MEMBER STETKAR: Sure.

MR. MARCH-LUEBA: Class 1E is safety grade. And what it requires, one of the requirements for Class 1E is that it survives an earthquake. So Class 1 systems are those systems that must work during the earthquake. So as the earth is shaking, the system must work and is designed to work that way.

Now imagine that the AMS system is running on your laptop and is sitting on top of the desk. An during the earthquake, it falls off the desk and breaks into pieces. Nothing happens to the reactor because that laptop is only needed two weeks from now to calibrate. Eventually, if the plant does not replace the laptop, they will not be able to calibrate and tech spec will shut them down two weeks from now. But the system does not --

The idea that you have to ask yourself is does this system must run during the earthquake and then it is Class 1E.

MEMBER STETKAR: Okay, I understand that. Extending the analogy, imagine that my laptop here is the AMS system and that the algorithms that translate the information that comes from the measurement table into signals that are actually used to calibrate the SPND detectors are programmed by Bozo the Clown --

CHAIR POWERS: Hey! Don't talk about me.

MEMBER STETKAR: I used to watch Bozo, too. I couldn't come up with the appropriate -- I was going to say your name.

CHAIR POWERS: If I did the program, you are in trouble.

MEMBER STETKAR: My point is that since this is a non-safety related system, you know, I understand that it doesn't need to meet the seismic qualification, except for the pressure boundary retention function.

I understand it has containment isolation valves on the nitrogen supply so that then indeed the containment would be isolated. However, the algorithms in the AMS system, in converting the measurement information into data that will be used to calibrate the detectors, are also non-safety related.

You know, I am curious about controls over that because -- I was wondering if you could expound on that.

MR. MARCH-LUEBA: It is an issue of nomenclature. We are not saying and AREVA is not saying that it is not safety -- they are not germane to safety. Those algorithms, if they are used to support a setpoint, they need to be approved and reviewed by the NRC.

MEMBER STETKAR: Okay.

MR. MARCH-LUEBA: So that at that point this algorithm becomes like RELAP. It is a code that you use to set a setpoint. It has to be reviewed and approved to be able to use it. But RELAP doesn't need to be seismically qualified.

MEMBER STETKAR: Right. I understand.

MR. MARCH-LUEBA: Okay, so when we say safety grade, you really have to think Class 1E, seismic LOCA.

MEMBER STETKAR: So you have reviewed the algorithms.

MR. MARCH-LUEBA: Yes.

MEMBER STETKAR: Okay, thanks. That is sort of the confidence that I was looking for.

MR. MARCH-LUEBA: Then 287, that --

MEMBER STETKAR: I didn't know, you know, it says calibration methodology. That could mean a variety of things.

MR. MARCH-LUEBA: It is a complex methodology and yes it was reviewed and is on the list.

MEMBER STETKAR: Okay, thank you.

MR. BUDZYNSKI: And we have two open items for the instrumentation part of 4.4. The first one is RAI 308. And we have a question about the methodology to remove the cobalt-60 activation background from the SPND measurements and how will this background be treated by the protection system and the AMS system.

And basically, is that cobalt builds up as a background of radiation and it has a long lifetime. And over a period -- what would you say the lifetime is?

MR. MARCH-LUEBA: Five years.

MR. BUDZYNSKI: Five years. And so therefore, as the plant operation goes on and it reaches a point where this would affect the actual readings. And we requested information on how they will be able to compensate for this background, cobalt-60 background, that it would affect the calculations.

CHAIR POWERS: And that item is still open?

MR. BUDZYNSKI: Yes, that is an open item.

The other RAI, test plan to verify the accuracy of the correction algorithms that are applied to the raw AMS activation measurements, including delays, activation buildup, and detector dead time.

What happens is when the AMS is used, all 40 of the stacks go into the core, it gets aerated and then you come out and sit and wait as ten of them are measured and then the next ten are measured.

And we will, to verify that if this is working correctly, algorithms, we want to reverse the order of how they are measured. The four stacks from A, B, C, and D, we want to go D, C, B, and A and then compare the actual profile that is generated to the first set against the second set.

Do you want to add anything to that, Jose?

MR. MARCH-LUEBA: Sure. The idea is if you want to make -- the making of this measurement is very simple conceptually. And if you want to make a plus/minus ten percent error measurement it is very simple to do. When you want to cut down that uncertainty to plus/minus 0.1 percent, you have to start correcting for the Coriolus effect. Okay?

(Laughter.)

CHAIR POWERS: I am fascinated that Coriolus effects. I am sure it does.

MR. MARCH-LUEBA: But there is a large number of corrections, including the time the detectors pile up, wait times. The fact that the balls that go on the bottom of the core travel through the core and they pick up some variation. So all those effects, this German technology accounts for. Okay?

What we said is divide some -- We ask AREVA, give us a test plant. Divide some of those parameters that you are correcting four and measure it. And assure me that you are measuring within accuracy that you require.

And they have a commitment to give us this response by April 30th. And I have been told that they have collected it and it looked very good.

MEMBER STETKAR: Good.

MR. MARCH-LUEBA: So we will see on paper and close the item, I am pretty sure.

MR. BUDZYNSKI: All right, the conclusions of this would be that the mechanical design and functionality of the SPND and AMS systems is similar to operating plants, and they satisfy applicable GDC criteria.

All vessel penetrations have proper Seismic 1 classification in that is our two we are referring to in that case.

Implementation of setpoint methodology. The staff has reviewed and evaluated the setpoint methodology in the SER on ANP-10287P, "Incore Trip Setpoint and Transient Methodology for the U.S. EPR Topical Report."

The staff evaluation has determined that additional information is necessary in the FSAR to adequately address the implementation of the methodology described in this report. So we generated an RAI open item, RAI 367. We have requested explanation on how the methods described in this report will be implemented and verified for the U.S.

EPR design.

Any questions on that?

MEMBER STETKAR: I do. That is the end of Section 4.4.

Nothing to do with AMS. Nothing to do with SPND. There was apparently, well reading the SER, there was a question, and it was RAI 134 question 04.04-21, just for the record, regarding thermal hydraulic conditions during shutdown and lower-power operation.

And as I don't have the actual RAI question nor do I have the response available, but I tried to do some tracing back through the reference material that I had. The discussion in the SER -- I wasn't sure what the concern was. The discussion in the SER talks about generic thermal hydraulic conditions. It talks about boron dilution events. It talks about mid-loop operations. It refers to the automatic isolation feature that I asked about earlier in the session. And in particular, just because of the PRA stamp on my head, it refers to Section 9 of the FSAR.

And apparently, the question was resolved because the FSAR now contains a reference from Section 4 to Section 19 of the FSAR. So, I dutifully went to look at the reference section that tells me what I learned in Section 19 regarding thermal hydraulic response during cold shutdown conditions. And first of all, I couldn't even find a discussion of boron dilution events in a PRA. They are apparently not considered at all.

The second thing is the PRA insights in Section 19 are two paragraphs that tell me that the dominant contributors to risk during shutdown are loss of offsite power events during mid-loop operations.

So I was curious first of all what credit the staff is taking for the PRA itself to resolve whatever concern you had about thermal hydraulics. That is question number one.

And of course the follow-on question is how does it do it, if indeed you are talking credit for it.

MR. LU: I think we are probably going to make a note of that.

MEMBER STETKAR: Okay, take that as kind of an action item.

MR. LU: It is related to the PRA side and --

MEMBER STETKAR: Well it is the whole close-out of this. Apparently there was a concern on the staff's part and --

MR. LU: I agree with that.

MEMBER STETKAR: -- it is not clear to me whether the concern was related specifically to boron dilution events or to other thermal hydraulic performance during shutdown modes. And if so, what other thermal hydraulic issues, other than boron dilution?

We heard earlier this morning in Chapter 9, I guess, we will get into how the actual boron dilution protection works.

The reference to Chapter 19 doesn't seem to provide me any information about close-out of that, of whatever that concern was. So simply including a reference doesn't seem to matter.

MR. LU: We will get back to you on that. Those are good questions but it is related to PRA side.

MEMBER STETKAR: It is a little bit also of my, I am very interested in understanding for the new plant designs how much and how the PRA is actually being used to support any type of design or licensing issues for the plant. So this happened to be one place where a close-out of the question kind of fed back into the PRA discussion.

MR. LU: Okay. We will get back to you.

MEMBER STETKAR: Thank you.

MR. CARNEAL: That concludes the presentation on 4.4. We will be moving into reactor materials now and I will ask Robert Davis and John Honcharik to join us at the table.

And section 4.5.1 is Control Rod Drive System Structural Materials and that will be handled by Robert Davis. So I will turn it over to him.

MR. DAVIS: My name is Bob Davis. I am a materials engineer in the Division of Engineering in NRO. I have been with the NRC for seven years, today actually. Seven years.

CHAIR POWERS: Hey, happy anniversary!

MR. DAVIS: I do all of the control rod drive systems structural materials for all of the design centers.

Prior to jointing the NRC, I was a senior welding engineer for Constellation Energy. Prior to working for Constellation, I attended Ohio State University where I received a Bachelor of Science degree in welding engineering. Prior to that, I was a welder for 13 years, six of which was in the nuclear navy program.

The control rod drive structural

materials, we only have two open items and I kind of consider these are more of a paper problem than a technical problem. In an RAI response on CRDM fabrication of the pressure housing, which is also covered, I discuss it in this section but mainly it is a reactor coolant compression boundary component and it is also discussed in section 5.2.3.

In a sketch provided by the applicant, they showed a forging material for 347 and they had no forging spec in their table for reactor coolant pressure boundary. So that is an open item. We should be hearing back from AREVA soon as a result of that.

The next issue is the use of 415 martensitic stainless steel. AREVA referenced two specifications. One specification is allowed for use by Section 3. It was included in the tables in Section 2 Part D. The other specification that AREVA specified for 415 is not included in the table.

It just means that nobody wanted to use that before so it wasn't included in the table. They have requested a code case be initiated to extend the properties from the approved material to the one that is not in the table that has been approved by ASME code and we are currently reviewing it. And that should be resolved relatively soon.

The control rod drive reactor pressure boundary materials, which would be the pressure housing all meet ASME Code Section III requirements except the martensitic stainless steel material, which I just discussed that item. But all the rest of the materials, they all meet Section III requirements for Class 1 components.

The non-pressure boundary components are ordered to DIN, RCC-M and SAE/AMS specifications with AREVA special order in requirements. Since this kind of a new area for us for CRDMs to use foreign material specifications, we requested that AREVA provide us with a comparison of all these materials and how they are similar to ASME materials that we are familiar with. They provided an extremely comprehensive RAI response, which compared chemistry and mechanical properties, just about everything that would be covered in the specification, heat treatment.

We reviewed that and found that along with their special ordering requirements, they are pretty much essentially the same as the ASME. They are ASME equivalents. And in their RAI response, some of which is proprietary, they address carbon content of 347, which was asked earlier. That has all been addressed in their response and I have that proprietary response available for anyone who would like to see that.

The selected materials for the CRDM have over 30 years of satisfactory operating experience. Some of the materials --

CHAIR POWERS: I struggle with the significance of that statement. I mean it would clearly be significant if it was unsatisfactory. I mean, what do I do with 30 years of operating experience?

MR. DAVIS: That means that these materials have been used, they provided an RAI response to some German facilities that have CRDMs using these same materials and that these CRDMs, that a certain number of them have been in operation, a certain number of them are still in operation. They listed how many of them have been disassembled and --

CHAIR POWERS: Davis-Besse could have said this.

MR. DAVIS: Well, I mean based on experience, we know that the CRDMs that have been disassembled, CRDMs made like these are going to be made, there haven't been any stress corrosion, cracking issues, or any other materials degradation issues, other than some wear of parts, which would be expected.

So I think that that is significant that there haven't been any problems with these materials.

CHAIR POWERS: I bet you Davis-Besse could have put this on a slide for us.

MEMBER STETKAR: It is good that the frequency isn't 0.03 per year, for example.

CHAIR POWERS: I agree.

MEMBER STETKAR: Or probably not 0.03 per year.

MEMBER SHACK: No but there are certainly plants here that couldn't make that statement.

CHAIR POWERS: Maybe.

MR. DAVIS: And I think the fact that they had information related to these CRDMs being disassembled and all these internal parts being inspected kind of makes their case that they are sufficient for use. And they are in CRDMs for the EPR.

Let's go to the next slide. The CRDM pressure housing is fabricated from grade 347, which is a stabilized grade of stainless steel. It is in a solution annealed condition and type 415 martensitic stainless steel, which is quenched and tempered.

They have also included information that their weld procedure qualification will include hardness testing and corrosion testing to ensure that the completed welds are not susceptible to stress corrosion cracking or hydrogen cracking.

The non-pressure boundary components are niobium stabilized and titanium stabilized, austenitic stainless steels. And they will perform corrosion testing. Our regulatory guide on austenitic stainless steels does not cover stabilized grades because they are typically not used. So, we made sure that we asked them that they would do corrosion testing or they would do some of the things that are recommended for the non-stabilized grades.

Other materials include type 410, which is commonly used in other designs. Alloy X-750 is commonly used. Haynes 25 has been used in foreign plants. I am not really sure if it has been used here. I don't believe so but it has a lot of operating experience in foreign plants.

They controlled abrasive work such as grinding, polishing, and wire brushing to prevent cross-contamination. And the cleanliness is controlled in accordance with the applicable regulatory guides. And based on their satisfactory performance and currently operating plants, these materials, we consider them to be compatible with reactor coolant.

Any other questions?

(No response.)

MR. CARNEAL: All right. If there are no other questions on Section 4.5.1, we will move to 4.5.2, Reactor Internals and Core Support Materials and that will be led by John Honcharik.

MR. HONCHARIK: Yes, my name is John Honcharik. Some background, I have a BS in material science from Brooklyn Polytech. Previously I worked for Newport News Shipbuilding for 15 years as a materials and welding engineer on naval nuclear reactor plants for aircraft carriers and submarines. I have been at the NRC for about seven years, previously in NRR and now in NRO.

And with that, I would like to discuss the topic of reactor internals and core support materials. We have three open items in this area.

The first one would be concerning the materials specifications. The DCD previously stated a low-carbon austenitic stainless steel that we used for the majority of the reactor internals and core supports. The staff knows that this material has very good resistance to stress corrosion cracking, especially in the PWR on the bottom. However, some of the components would be hard-faced, such as pins, with Stellite 6 for wear resistance.

Now in a response to RAI, AREVA stated that the Stellite 6 is based on filler material specification in ASME Code SFA 5.21. And this hardfaced material is also consistent with material use in current operating reactors. However, this ASME Code filler specification was not identified in the FSAR. So therefore this is an open item until the ASME Code specifications are included in the FSAR.

The next slide. I will now talk about some other material considerations. The FSAR previously did not provide an assessment on how irradiation assisted stress corrosion cracking and void swelling will affect integrity on a reactor vessel internals and core supports for the proposed design and materials. I think this goes back to Mr. Shack's question.

An RAI response provided some information concerning the effects of irradiation stress crack and void swelling on the integrity of the reactor vessel internals and core supports. However, AREVA stated that it will use the criteria for current operating reactors to screen the applicable internals and core support structures for irradiation stress corrosion crack and void swelling.

MEMBER SHACK: Now there is an MRP-277, which I guess the license renewal people are going to use. And this is code for saying we are not going to do that. Is that the response that you are getting?

MR. HONCHARIK: Well the response that I am getting is basically they are going to do what the operating fleet is doing.

MEMBER SHACK: Oh, okay.

MR. HONCHARIK: Basically using the guidelines of the MRP.

MEMBER SHACK: Okay.

MR. HONCHARIK: However, --

MEMBER SHACK: But they are not going to say MRP-277 out loud?

MR. HONCHARIK: Well in their response, they mentioned MRP-175 for screening criteria.

MEMBER SHACK: Okay, that looks at the materials but that is not the inspection program.

MR. HONCHARIK: Right. So therefore, that is why we have this open item and basically okay, you need to be able to screen which components may be affected by this. And then once you do that, propose a description of how you are going to inspect these components, this augment inspection that you say you are going to do. So that basically is the open item for this.

Right now, they need to either provide this evaluation screened in a proper description or they can put it to the COL application. The COL applicant will provide that information. Basically that is the heavy reflector. It is mostly the reflector. That is the one that is probably going to see the most fluence.

Also there is another issue on the heavy reflector.

MEMBER SHACK: Is the reflector the only component that is going to see really high fluence, it is going to stay there forever? I mean, well, for 60 years.

MR. HONCHARIK: That is the one that comes out. There is a couple of other ones. You've got the heavy reflector. There is also the radial key inserts and intermediate core barrel shell, it is a low shell well. Those are the areas it is possible.

MEMBER SHACK: To see enough fluence?

MR. HONCHARIK: Right. But as you know, they still need to factor in the fluence but the stresses and everything else for that.

Any other discussions with that?

MR. CARNEAL: Okay, if there are no other questions --

MR. HONCHARIK: Oh, wait. I still have a little more.

MR. CARNEAL: Okay.

MR. HONCHARIK: Okay, in addition, there are vertical keys and keyways that are used in the forged heavy reflector slabs which could introduce some stress concentrations and crack initiation sites. Therefore, the staff had identified this as an open item. The applicant should provide some information that these notches and other stress concentration sites will prevent -- to make sure that they are not present, therefore, to prevent and maintain the integrity of the heavy reflector. So that was also an open item for the heavy reflector. And those are the only open items for the internals.

MR. CARNEAL: Are there any further questions on reactor materials?

(No response.)

MR. CARNEAL: Okay, thank you very much. We will move on to Section 4.6. I am going to call Shanlai Lu and John Budzynski back to the table to cover this section.

MR. BUDZYNSKI: On 4.6 we have one open item and the open item is really a clarification between two different sections. Section 4.6.4 states that no credit is taken for a boron addition via the SIS system, except for the large break loss of coolant accident. But when we reviewed Chapter 15, we found that the main steam line break does take credit for this.

And so we generated an RAI for clarification purposes. It is an open item. Any questions?

CHAIR POWERS: And they will presumably respond at some point.

MR. BUDZYNSKI: Yes.

MR. LU: Yes, they are going to respond, I think. But is this just clarification.

MR. BUDZYNSKI: Yes.

MR. LU: It is really not technical. It looks like --

CHAIR POWERS: They misstated themselves or something like that.

MR. CARNEAL: I think that concludes the staff's presentation on the SER with open items for Chapter 4. Are there any further questions?

CHAIR POWERS: You don't see any -- it

didn't sound like there were any major stumbling blocks here. These are resolvable things.

MR. TESFAYE: Yes, we have clear paths forward for all the open items.

CHAIR POWERS: We don't have to make a major breakthrough in science and technology. No miracle has to happen.

MS. SLOAN: Dr. Powers?

CHAIR POWERS: Yes.

MS. SLOAN: If we could, could we comment on one thing that we heard earlier we would like to --

CHAIR POWERS: You certainly can.

MS. SLOAN: -- make sure we correct that information.

CHAIR POWERS: If we are done. Let me make sure we are done with these people.

MR. CARNEAL: Yes, we are done.

CHAIR POWERS: We're happy? You are happy?

MEMBER STETKAR: Yes, sir.

CHAIR POWERS: You're never happy.

MEMBER STETKAR: I'm delirious.

CHAIR POWERS: No, hilarious. I'm sorry.

MEMBER STETKAR: That, too.

MS. SLOAN: Go ahead.

MR. WILLIAMS: This is Gary Williams, AREVA. Just as a matter of correction for the record, we have had many discussions, very productive discussions with the staff, lots of information exchanged, lots of data exchanges. So with that, recall can be very difficult.

But there was a statement made earlier that the M5 assembly growth was greater than the Zirc assembly growth. And that is not the case with the datasets that we have. For the fuel assembly designs where we have both M5 guide tubes and Zirc-4 guide tube data, the growth for M5 is equal to or less than the Zirc-4 data.

I think the issue or the question is what would be expected in the realm of that specific design application and the level of variance of scatter or expected scatter within that dataset. AREVA has done a considerable amount of work in understanding many of the parameters that affect the variances in growth, mainly as it pertains to the loading on the guide tubes and there are many facets of that.

There is continued work that AREVA is doing in the area to better understand that loading sensitivity to irradiation, whether there are things that are presently not understood that need to be understood to clarify the sensitivity to the loads.

And parallel with that, are efforts made with incore and out of core testing of material itself. It was made reference to earlier as far as the effects of possible C-loop density dislocations as affected by either stress levels or hydrogen levels in the material. But really I just wanted to make the correction about the M5 growth being higher than Zirc-4 growth and that is not the case.

MEMBER SHACK: But what you are saying is that if you have M5 at the same fluence and the same stress level, the growth is not higher than Zirc-4 at the same stress level.

MR. WILLIAMS: It is not higher. For the case where we would classify the anomalous growth, it was equal to the historic Zirc-4 growth data and that was the part that was unexpected. M5 material, guide tube material, structural material, brings the aspects that the cladding has as far as improvements, lower oxide, lower hydrogen levels, the lower oxide would reduce the level of oxide-induced creep stress on the guide tubes. Both the Zirc-4 and the M5 material are fully recrystallized annealed material. So the microstructure from that standpoint would be transparent.

So at worst, the expectation and what has been observed in the data sets is that the M5 is equal to or less than the Zirc-4 growth.

CHAIR POWERS: My understanding on this is almost zero. My understanding is you still find situations where M5 grows more than you would expect.

MR. WILLIAMS: From the standpoint of trying to understand all of the facets of causes for the ranges of growth we are continuing to pursue. But the same is true with Zirc.

CHAIR POWERS: That is true but I am not concerned about Zirc right now.

Right now M5 you find situations where the material grows due to irradiation more than you would expect.

MEMBER SHACK: That is why we call it anomalous. Right?

MR. WILLIAMS: Yes, that is correct. From all of the elements that we have defined that would contribute to growth, the ranges of growth are not fully explained. That is correct. So we are using the data sets available for low growth and high growth to build design limits.

CHAIR POWERS: Okay.

MR. LU: The staff would like to make a

comment on this. I think we understand your view on that. That is the reason I think right now we are saying that the M5 growth and the COPERNIC strain issue right now it is classified as the topic of interest to the committee. It is not an open item. It is an ongoing review as part of a topical report review.

CHAIR POWERS: My recollection, without actually looking it up is your statement was that it was more than expected growth and any comparison was made orally.

MR. WILLIAMS: That's right.

MEMBER SHACK: I did have one question. I mean since the COPERNIC is an approved code, why didn't you spot the discrepancy in the predicted strains in the course of that approval?

MR. BEYER: Yes, that was at that time, when we reviewed COPERNIC, --

CHAIR POWERS: That's a dead mike there.

MR. CARNEAL: Just to help out our recorder, make sure you have a live mike.

MR. BEYER: Yes, this is Carl Beyer again, a BS in physical metallurgy and an MS in material science.

Yes, basically COPERNIC, back when it was reviewed, there wasn't as much attention paid to the cladding strain issue at that time. Because back during the early days, the issue of strain and PCI was addressed mainly through ramp tests and more of an empiricism type of methodology to approve designs. And so the cladding strain calculations done by codes wasn't considered as critical.

But today, there is virtually not very much cladding power ramp data done on new fuel designs and the fuel vendors are concentrating more on proving that they can meet their design criterion for cladding strain based on fuel performance code calculations, rather than empiricism type data from power ramp data.

And so that is the reason why today we are paying more attention to it from a cladding strain prediction standpoint, from a fuel performance code, versus what we did ten years ago when more empiricism type data was used to demonstrate cladding-strain criteria were met.

MR. LU: Any more questions?

CHAIR POWERS: You are done?

MEMBER RYAN: No, lunch.

CHAIR POWERS: Lunch? You are done? Okay, we will recess until 1:00. And Getachew and Sandra, I would like to chat with you just about some mechanics right after this meeting.

(Whereupon, at 11:46 a.m., a lunch recess was taken.)

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

(12:59 p.m.)

CHAIR POWERS: Let's come back into session. We are going to launch into our discussion of the RCS, which I am informed reliably is the reactor system. So this will be new and unique for most of the members here. So you may have to go into things in a fair amount of detail, more detail than I anticipated.

And it has Mr. Tesfaye dumb. He is completely stunned by this revelation and he may have to revise his SER completely here.

So with this, we will turn it over. Ms. Sloan.

MS. SLOAN: Okay, so Chapter 5, Reactor Coolant System, yes, we have one, and the connective systems. Again, our objective is provide an overview, summary level presentation of the material and the organization of the FSAR chapter and we are going to, as always, try to focus on these --

CHAIR POWERS: I now understand. The RCS is the reactor coolant system but the abbreviated overview leaves out the coolant part of it.

MS. SLOAN: Yes.

CHAIR POWERS: Okay.

MS. SLOAN: So consistent with other

presentations, we will focus on those things that are unique or different for U.S. EPR compared to current U.S. operating plants. And with that, I will introduce the two presenters who, consistent with our custom, will give their own brief biographical information at the start of their presentation.

We have our first presenter is Dennis Newton who is our supervisor in reactor coolant system design. And then during the presentation we will switch to Dale Matthews, who is supervisor for RCS component design. So, Dennis?

MR. NEWTON: Begin? Okay. Good afternoon. My name is Dennis Newton and I am supervisor of the group responsible for the RCS system design, as mentioned. And I have over 35 years of utility and vendor experience with the design operation testing of the nuclear island systems, primarily the reactor coolant system.

I have a B.S. degree in nuclear engineering from the University of Massachusetts. I guess the only college up north that we have heard from so far. You will pick it up in my accent.

CHAIR POWERS: What accent?

MEMBER RYAN: Sounds fine to me.

(Laughter.)

MR. WIDMAYER: He's the only one that doesn't have an accent.

MR. NEWTON: Anyway, some specific experience includes reactor engineer, reactor operator, systems design engineer and a system test engineer. Some specific experience includes core physics testing, system readiness reviews, design bases reconstitution, system design modifications, 50.59 reviews and power upgrades.

I am also the Chairman of the ANS Subcommittee on System Design Standards. Give them a plug.

MR. MATTHEWS: My name is Dale Matthews. My current role is supervisor of component design for U.S. EPR. My group has responsibility for design of the primary components, piping and supports.

My background is I have been with AREVA for approximately 20 years. My most recent background --

CHAIR POWERS: This accounts for the strong French accent. Right?

MR. MATTHEWS: That's exactly right. My most recent experience before coming to EPR was design and manufacturing and installation of replacement reactor vessel heads, pressurizers, control rod drives, pressurizer heaters, some component supports, other plant equipment, heavy load handling equipment, canal seal plates, that sort of thing.

Before that, I spent about ten years in services where I was heavily involved in Alloy 600 repairs, specialty tool design, specialty weld repairs and that sort of thing.

CHAIR POWERS: You are bringing tears to Dr. Shack's eyes.

MEMBER SHACK: I was going to say you gave everybody a good living.

(Laughter.)

MR. MATTHEWS: I am active in --

MEMBER SHACK: God bless Alloy 600.

MR. MATTHEWS: I am active in ASME code committee work. I am the incoming chairman of the Working Group on Vessels and Subgroup Design, Subcommittee Three. I am active in the Subgroup for Industry Experience for New Plants and I am on the Subgroup Design and Subcommittee Three.

And before all of that, I got a BS in ME from the University of Alabama. So I am at the opposite side of the country from my Massachusetts colleague.

CHAIR POWERS: You have to work for the

damn Yankee. And I feel your pain.

MR. NEWTON: When I moved down here I had to tell everybody that I used to work for Yankee Atomic Electric Company.

Okay, as was mentioned, this has to do with this presentation on Chapter 5. And this slide here shows the different sections of Chapter 5. 5.1 is of course the overview of the reactor coolant system; and 5.2 discusses the topics such as overpressure protection, materials, ISE, and leak protection; 5.3 is the reactor vessel; and 5.4 talks about the various components to the reactor coolant system, which includes the pumps, steam generator, pipes. It also has a residual heat removal system thrown in, which is different. The pressurizer, the pressurizer leak tank, high point vents, the pressurizer relief valves and supports. And we will present the material according to the way that it is presented in the chapter itself.

The first thing that I want to mention and sort of emphasize is that the U.S. EPR is a conventional pressurized water reactor. It is a typical U.S. four loop pressurized water reactor --

CHAIR POWERS: Now that is a stunner.

MR. NEWTON: Well, it is just a fact. I

don't think it is anything new except for we do have some features to improve safety and operation.

But it does have four U-tube steam generators and four reactor coolant pumps. And the steam generators are there to make it a coolant system.

Okay, this is a three-D model to show the conventional layout of the reactor coolant system. And as you can see, if I can get the cursor working, you do have the four steam generators, the reactor vessel, the four reactor coolant pumps, the pressurizer, and the surge line going to the hot leg pipe. Okay, that is a general overview.

Okay, we made a summary here of some of the features that make it a little bit unique. The reactor coolant pumps, they have a stand-still seal system. We will talk about that a little bit later but that is there to isolate seal leakage from the reactor coolant pump, if you have an instance such as station blackout.

The reactor coolant pressure vessel, there are no nozzles in the lower head of the reactor coolant pressure vessel. We use pressurizer safety relief valves. They are medium control valves and we use the same valves for overpressure protection at power and also for low temperature overpressure protection.

The pressurizer has a continuous venting to minimize any accumulation of any non-condensable gases. The pressurizer safety relief valve nozzle on the inside of the pressurizer, there is a water container that collects condensation to actually create a water seal between the pressurizer steam space and the pressurizer safety relief valve.

The pressurizer has primary depressurization valves which are used for severe accident. The steam generator has an axial economizer. And again, there is the heavy reflector which significantly reduces the neutron flux on the reactor vessel.

Okay. This slide here shows the performance values of the reactor coolant system. And as you can see, the pressures and the temperatures for the core, for the hot leg and for the cold leg are very typical with existing reactors.

As an example, you just have to look at the South Texas Project 1 and 2. They have a hot leg temperature of 625 and a cold leg temperature of 560. The operating pressure is 2235, the same as U.S. EPR. So the operating parameters are very typical. The one difference is it does have a higher power level and because of that, we do have a higher flow.

CHAIR POWERS: This is not an inconsequential change, by the way. This is not an oh, by the way.

MR. NEWTON: Okay, we made the parts bigger. But if you take the ratio of power to flow, it is pretty much the same.

The reactor coolant pressure boundary components, they are designed and fabricated to ASME III, Boiler Pressure Vessel Code, 2004 edition with no addenda. The reactor coolant system pressure boundary components are Class 1 for the most part but we also do have some Class 2 components. If say the small pipe can break and if the water makeup is within the capacity of the CVCS system, I guess you don't have that acronym but chemical and volume control system.

And there are five ASME code cases that are called out. Only five and they are in the FSAR Table 5.2-1.

Okay, overpressure protection. As I mentioned earlier, we have pressure safety relief valves on the pressurizer and they are used for both overpressure at power and low temperature over pressure protection.

At power, it is used for a turbine trip event and that is discussed in Chapter 15 so I won't go into that. But for overpressure protection at low temperatures --

MEMBER STETKAR: Dennis, first at-power, I think I read that the opening setpoint is 2535 psig. Do all three of the valves open at the same setpoint pressure?

MR. NEWTON: Yes, we made them the same setpoint.

MEMBER STETKAR: Why? I mean, a lot of plants have staggered setpoints.

MR. NEWTON: Yes, well we decided to make them all the same setpoint and we did analyze them for the additional load, assuming they all open at the same time.

MEMBER STETKAR: I am more concerned about popping all three open and having one of three stick open, rather than one of one; any one of three sticking open for an induced LOCA rather than only the first one.

You have three times the likelihood on a minor pressure surge of getting a stuck open relief valve, if they all open at the same pressure. If only one of them has to -- For example, if the pressure relief requirements require only one valve to open, if the first one opens, it might stick open but if all three of them open, anyone of those three might stick open. Other people have used that as justification for why to stagger opening pressures a little bit.

MR. NEWTON: Yes, well we got better results by having all the same setpoints for the sake of analysis.

MEMBER STETKAR: Okay. I just wanted to make sure that I understood that they all opened. Thank you.

MR. NEWTON: It may be sort of a carryover. The BNW plants, they always had them both safety valves at the same setpoint.

MEMBER STETKAR: Yes, they only had two.

MR. NEWTON: Well that is only two, true. So we didn't think that was anything unique.

MEMBER STETKAR: The only reason I ask is that I am somewhat familiar with plants in Europe and I think the ones that I have seen have staggered relief valve settings on them.

MR. PARECE: This is Marty Parece. The staggering of valves in Europe is also because of the difference in their classification. For us, we can only go to 110 percent of design, according to ASME but for certain of their events, they can go to 120 percent. Therefore, it allows a stagger.

MEMBER STETKAR: I understand. Thanks.

MR. NEWTON: There was some discussion about the Appendix G Reactor Vessel P-T limit here in Chapter 4. And I guess from my experience, I guess I was really impressed by how little the Appendix G limit changes because of the heavy reflector we use and I guess we also have some improved materials, too.

To give you an idea, the LTOP setpoint that we have, it is going to stay the same for 60 effective full power years. And the setpoint is 556 psia and the enabled temperature is 250 degrees. So that Appendix G limit just stays way up there out of the way of the operators operating the plant.

MEMBER SHACK: That is why you changed the material and put in the shield.

MR. NEWTON: Yes, it works well. For the overpressure transients at low temperature, we looked at the mass addition and we looked at heat addition. The worst case mass addition was the medium head safety injection pumps coming on and for the heat addition it was starting the reactor coolant pump with the steam generators 50 degrees higher than the reactor coolant system. And we have tech specs to make sure we maintain those assumptions in doing the pressure analysis.

MEMBER STETKAR: Dennis before you -- I know the next topic is you are going to get into more of the materials thing and I was kind of reading ahead here. I had a question regarding the PSRVs and the primary depressurization valves, sometimes they are called severe accident depressurization valves.

All of those valves discharge into a common line down to the pressurizer relief tank.

MR. NEWTON: Correct.

MEMBER STETKAR: Correct? Have you done any analyses to look at that line in terms of can it withstand a simultaneous discharge from all five of those valves structurally?

MR. NEWTON: Well, I know we did for the pressurizer safety relief valves.

MEMBER STETKAR: I read that. My concern is that there can be scenarios and what I am thinking about is the potential feed-and-bleed cooling scenarios where the operators may very well try to open up all the holes they can in the primary system pretty quickly.

I don't know what the operating procedures, the emergency operating procedures for this plant would be and what type of administrative controls would be in place to prevent people from doing that. So I have no information about that.

But I was just curious whether it would handle the dynamic loading if all five of them opened up.

MR. NEWTON: Well the primary depressurization valves, you have two, a gate valve and a control valve series. So they are not going to open real quick.

MEMBER STETKAR: I'm not talking about themselves. I am talking about, you know, I used to be an operator, Joe Operator looking at a situation where I don't have secondary heat removal and my procedures tell me to open up everything.

MR. NEWTON: Do everything I can all at once.

MEMBER STETKAR: And I am not particularly, you know, I want to depressurize the plant. So I am more concerned about if that line fails, you know, I am essentially creating a LOCA. I am more concerned about if that line does fail, where might it fail and can you get the resulting blowdown back down into the IRWST or whatever you folks call it? I probably gave it the wrong acronym.

MR. NEWTON: Yes.

MEMBER STETKAR: Because you are pretty careful. It is a very compartmentalized containment design and you are pretty careful about making sure that water goes where you think it should go. So the reason I ask the question is not because the line might fail and you might have a blowdown, it is if the line does fail, who indeed you get the blowdown with the inventory return back to places that you really want to it to go back to.

Because you are pretty careful about where you blow the rupture discs from the PRT to make sure it goes back into the reactor coolant pump cubicles to get down in.

MR. NEWTON: Yes, all the cubicles do go in the reactor building water storage tank.

MEMBER STETKAR: Even if you get a break high up, like from the top of the pressurizer area?

MR. NEWTON: Do you know about the pressurizer relief tank?

MR. BANKE: Yes. My name is Jim Banke. I work for AREVA. I was a six-year Navy nuc and then 26 years at Constellation's Ginna Station and ROSRO qualified shift supervisor. Then I got my engineering degree, a mechanical engineering degree at Rochester

Institute of Technology.

And then for ten years, I was the RCS/CVCS system engineer. And then four years ago I went to AREVA. So I have been with AREVA for four years.

Yes, the surge line is enclosed within the pressurizer cubicle and that cubicle does have a drain back to the RCS loop area and that in turn drains back to the IRWST. With the volume of water it should go over the two-inch lip that we have that separates the normal floor drains. And it should go through the racks to the IRWST.

MEMBER STETKAR: Now you said, I want to make sure I understand you. You said the surge line. I am talking about the relief line.

MR. BANKE: Well I mean the relief line also within that same --

MEMBER STETKAR: The relief line. Okay. And throughout its transition it goes through cubicles that would drain back to --

MR. BANKE: Yes, they do ultimately drain back to the IRWST.

MEMBER STETKAR: Thank you. I am less concerned about whether the line breaks than --

MR. NEWTON: Yes, it is not doing any safety function.

Okay, is that the last question on this slide?

MEMBER STETKAR: Yes.

MR. MATTHEWS: Okay, again I am Dale Matthews. I am going to just talk a little bit about the pressure boundary materials that are used for the primary components. They are all Class 1 components, so of course they meet the Class 1 rules of Section 3. These components are generally fabricated from materials we are all familiar with. The forgings are primarily SA-508 Grade 3 clad with either austenitic stainless steel or nickel-chrome weld filler.

The forgings were collected in such a way to minimize the number of welds in the plant. That was one of the design goals for EPR. So rather than a bunch of rolled plates with long seams, we went to circular forgings and reduced the number of welds to that which could be done with our current ability to manufacture materials.

These forgings meeting the requirements or the guidelines of Reg Guide 1.43 for fine grain practice to prevent or to minimize the potential for under clad cracking. We do take advantage of some improvements in steel making technology since the first time we built plants in the area of chemistry controls. We are able to do a lot better today on RTndt than was specified for the original plants. The original plants, most of the design specs specified an RTndt of 40. We specify minus four and we have met that routinely with other replacement components that we have built over the last ten years.

Austenitic stainless steel, it is all fabricated low-carbon solution annealed so we follow the Reg Guides, the requirements or the guidelines of Reg Guide 1.44. Other chemistry controls, cobalt and sulfur are limited. Nickel-chrome alloys, we use 690 everywhere there is wetted nickel-chrome alloy. State of the art best practices are followed in the manufacture of that material. The chemistry is controlled more tightly. Solution annealed followed by a thermal treatment to get the right grain structure and grain size.

We have addressed the lessons learned from operating plants in these material selections. I mean, obviously there is no Alloy 600 anywhere in the plant. No 182/82 filler material. The other place where we have seen some failures is cold worked wetted stainless steel parts. So we have eliminated those as well. Next slide.

Fabrication. Again, these are code

components. They are built to the rules of Section 3. In addition to the rules of Section 3, there are some Reg Guides that are applicable, 1.71 in some cases for welder access. The preheat guidelines of Reg Guide 1.50, we followed those in qualification of our weld procedures.

Other lessons learned from the operating plants is that weld repairs to wetted surfaces are minimized to the extent that they could be minimized. If we are in a situation where we do have to do a weld repair to a wetted surface, we do them in both measures to minimize or remove all together any residual tensile stresses at the surface of the weld.

Sensitization again, we talked about Reg Guide 1.44. We follow that for material chemistry, delta ferrite, and also heat input controls while we are welding.

And then the other thing that, you know, in my experience, we have seen failures attributed to was maybe inadequate cleanliness during original construction. So we are really careful to control fluorides, fluorides, low melting point metals, that sort of thing through all phases of manufacturing but in particular before we do any welding or heat treatment or anything of that nature.

Next slide.

CHAIR POWERS: How do you control fluoride?

MR. MATTHEWS: Just in process cleaning. We specify limits for any type of chemicals, cutting fluids, or solvents or whatever that are used on the products during manufacture. We impose chemical limits on those things. And if any product that we use exceeds those limits, then we require the component to be cleaned and wipe-tested to make sure it is all gone before we go into any sort of forming or heat treatment or welding type operation.

Okay, in-service inspection, we actually went to some effort on the EPR to make sure all the components could be inspected, pre-service and in-service inspection, every inspection that is required by Section XI and even some inspections that are required beyond what is in the code right now; 729 is one example, the CRDM nozzles. Those have been designed so that they can be inspected per the rules of the code case that is involved.

So every component has been reviewed with our engineering team and our NDEs. Our NDE level threes that go out and do ISI on the components in the operating plants to make sure that all the inspections that are required to be performed can be performed using existing technology.

Again, I mentioned this earlier, but to facilitate ISI, we have also from a design point of view, done what we can to reduce the number of in-service inspections that are required by our material selections. Next slide.

CHAIR POWERS: Well, I mean you raised the question, didn't you, that -- I like this, that you have looked at what the existing technology is and that you can inspect these things. But have you prognosticated what kinds of technologies might come along in the say 20 years after plant fabrication and said can I inspect it with those, which may then be standard technologies?

MR. MATTHEWS: Well the assumption is that future technologies will be an improvement over the existing technologies. So the assumption would be that if you can inspect it with current technology than future technology should be an improvement upon that.

So I think the assumption of current technology is worst case.

CHAIR POWERS: Yes, I mean I can imagine making that assumption but I can also imagine being dead wrong, too.

MR. MATTHEWS: But any new technology you came up with would have to be qualified against the ISI requirements that are in place. So you couldn't even deploy that technology until you have demonstrated that it could perform all the required in-service inspections.

Okay, next slide.

CHAIR POWERS: But I do think this is very non-explicable. I mean, talk to the guys that have to do it now and say what makes your life hell. And snicker and say no, wait until you see this.

MR. NEWTON: Okay, the next slide pertains to the reactor coolant pressure boundary leakage protection. The containment environment is divided. This is physically divided into two areas. There is a service area and an equipment area. And the equipment area is about one-fourth of the containment volume and all the reactor coolant pressure boundary components are within the equipment area. So that helps the accuracy of measuring the leakage.

There is a sump that is used for detecting leakage. That is safety related. There is a gaseous a particulate radiation detector located in the equipment area. And there are nine air coolers located in the equipment area and they measure the amount of condensate that is being generated.

In addition, there are 13 temperature and humidity detectors located within this area. So to help identify where any leaks may be coming from, there is one near each of the reactor coolant pumps, one near the bottom of each steam generator, near the pressurizer, near the surge line, and a couple of them near the top of the steam generators.

That is all I have to say about this.

MEMBER SHACK: Nothing near the reactor head?

CHAIR POWERS: Why are you concerned about that head here, Bill?

MR. NEWTON: I assume you are relating to the controller drive mechanisms leaking.

MEMBER SHACK: Yes, that sort of thing.

MR. NEWTON: I think -- do you want to address that? It is based on in-service inspection.

MR. MATTHEWS: I can't address the question of whether or not there is any leak detection capability but I can address the control rod drive nozzles. I mean, the original concern was the 600 material that was used for construction to those. And not only was it 600, in many cases, it wasn't heat treated very well. A lot of the annealing temperatures were down around 1600 degrees, particularly the heat that was used at Davis-Besse.

That problem has been solved in a BPR, not only by the fact that we have gone to 690 and using the right material practice, or the fabrication practice for the material, --

CHAIR POWERS: And you have assumed 690 is not going to crack?

MR. MATTHEWS: Well, --

MEMBER SHACK: I mean Alloy 600 was the wonder material of its day.

MR. MATTHEWS: It was. The only thing you could say is a lot of testing has been done on the 690. It has been aggressively tested, compared to 600, and it has been shown to be a lot more resistant but there are in-service inspections that are now required.

Another problem at Davis-Besse was the design of the head made it very difficult to a bare head inspection. There was a solid steel cylinder all the way around all the control rod drive nozzles. You had a couple little holes about that big that let you see up in. So you didn't have good visual access to see the top of the head either. And once you could see in those holes, there was insulation that didn't let a lot -- so it was hard to see.

The EPR is pretty open access to do bare-head visuals. All the old panels, we raised the old panel ring off of the head, you can see right in there. So if you were to have a problem, you would see evidence of it fairly easily, as opposed to Davis-Besse where you had very poor access to the head.

And then the code case requires us to do considerably more examinations on those welds than was ever required for the operating plants before Oconee 3 and some of the other plants started showing degradation.

MR. PARECE: This is Marty Parece. There is one other feature that we have added is that we have got a specific storage location for the head during refueling. That location has a number of design criteria applied. One of course is to prevent shine on people from the head during the refueling operations. But it also sits on a stand that gives us very good under-head access and on-top-of head access to do these inspections during the outage.

So we feel pretty comfortable with the material selection, the fabrication techniques and the access to the head during refueling operations that we don't need a continuous humidity monitor on top of the head.

MEMBER SHACK: And the penetrations that we have for the AMS system, those materials are, --

MR. MATTHEWS: Those are the same materials. All the head penetrations are the same material.

MEMBER SHACK: Are 690?

MR. MATTHEWS: Uh-huh. Welded with 152 to the head. It is all the same material, same practices.

MEMBER SHACK: And what does that weld look like for one of those small tubes?

MR. MATTHEWS: For the instruments that go into the nozzles? Those are canopy seal welds. Very similar to what the operating plants have.

MR. NEWTON: Next.

MR. MATTHEWS: Okay again, here is just a picture of the reactor vessel. You can see where we have eliminated lower head penetrations. You can see the nozzle shell is -- I'm sorry.

So we have upper shell, lower core shell, a weld between them. Lower head no penetrations. The nozzle shell forging is all an integrated forging. So the nozzles are actually set on instead of set into the forging. That is an improvement from an ISI point of view. The inspection volume is smaller. And then there is a 316 safe end welded to the nozzle. And you can see the reactor head penetrations up here. So it is a fairly typical reactor vessel.

This shows a picture of the upper forging with the integral nozzles and this is a picture of the entire reactor vessel being fabricated for OL3.

MR. NEWTON: The reactor coolant pump/motor. The reactor coolant pump/motor is a typical single-stage centrifugal pump. The flywheel conforms to Reg Guide 1.14. The reactor coolant pumps have an oil collection system that conform with Reg Guide 1.189.

CHAIR POWERS: And the flywheel is made out of what?

MR. NEWTON: The flywheel?

CHAIR POWERS: Yes.

MR. NEWTON: What was the question?

CHAIR POWERS: What is it made out of?

MR. NEWTON: I've got to ask Marty.

MS. SLOAN: Is the question what is the material of the flywheel, Dr. Powers?

CHAIR POWERS: Yes.

MS. SLOAN: Is there anybody prepared to

respond? If not, we will have to take it as a follow-up. Okay, we will have to get back to you. We will take it as an action, Dr. Powers.

CHAIR POWERS: I'm sure it is in the FSAR. I just asked.

MR. NEWTON: The flywheel will be tested to 125 percent of speed and then we will do an ISI inspection after the first spin of the flywheel.

There is not much else to say about the --

CHAIR POWERS: You will?

MR. NEWTON: The vendor. We will probably use Juemont.

MEMBER STETKAR: Who is the vendor of these? Do you know yet? You don't have to say if you don't know yet.

MS. SLOAN: The vendor for the RCPs.

MEMBER STETKAR: Who is going to make your pumps?

MR. PARECE: Juemont is going to make the reactor coolant pumps. That is a subsidiary of AREVA.

MEMBER STETKAR: Oh, okay.

MR. PARECE: Juemont fabricated all, I think all of the reactor coolant pumps for the French program.

MEMBER STETKAR: Okay. Yes, I am not

familiar with the -- I am familiar with the KSB up in Germany.

MR. NEWTON: Okay, next slide.

Okay, this slide here talks about the Stand Still Seal System and isolating the thermal barrier for leakage. The Stand Still Seal System is, I guess you would call it a piston ring. And it moves up and creates a metal to metal interface to stop any leakage out the shaft of the pump. It is put into place by nitrogen gas pressure.

If we happen to have a station blackout situation then what will happen is you will have the nitrogen gas pressure will move the stand still seal up and isolate leakage along the shaft. And you will also have the isolation valve for the let down lines closing and also for the seal injection. So the leak seal package will be isolated.

If we have a thermal barrier leakage, this is the thermal barrier, again, we have a situation where we can detect a leakage either by temperature or flow in the component cooling water line leaving the thermal barrier and this allows us to detect the leakage in a single pump. And so we can go ahead and isolate the CCW to that particular thermal barrier.

Any questions?

MEMBER STETKAR: Yes.

MR. NEWTON: Okay.

MEMBER STETKAR: Is there an automatic pump trip on loss of component cooling water flow?

MR. NEWTON: No.

MEMBER STETKAR: Well, it doesn't have to be a flow trip. But I mean, if I lose component cooling water flow to the thermal barrier.

MR. NEWTON: No, that will not trip because you have the seal injection. But if you lose seal injection and component cooling water, then that will start the countdown for your seal, for your stand still seal actuation.

The stand still seal does not close until after 15 minutes to allow time for the pump to close down.

MEMBER STETKAR: There is a trip from high seal package temperature or what trips the pump? Because a stand still seal needs to have a stationary pump. Right?

MR. NEWTON: No. The stand still seal uses gas pressure in the gas tank here.

MEMBER STETKAR: I am a little confused because I can't find much information on this. This is the best picture I could find. It seemed to say that at pressures I believe above 218 psia, if I recall correctly, you don't need the gas pressure because actually the upward force on the pump shaft should give you a face-rubbing mechanical seal. And the gas pressure was required if primary system pressure was less than 218 pounds.

MR. NEWTON: That is correct.

MEMBER STETKAR: Oh, is that correct?

MR. NEWTON: Yes.

MEMBER STETKAR: Okay. So whether or not I get the gas pressure injected at 2250 pounds, primary system pressure doesn't seem to make any difference. My question is, will the stand still seal provide sealing if the pump is rotating? Or does the pump need to be stationary? That is a question.

MR. NEWTON: The pump needs to be stationary.

MEMBER STETKAR: Okay. If the pump needs to be stationary, what trips the pump? What signals trip the pump if you happen to lose both thermal barrier cooling and seal injection, which you would if you lost all component cooling water.

MR. NEWTON: Yes, it is in the INC logic and it looks at the flow rate of the component cooling water and it looks at the flow rate for the seal injection. And if they go below a certain point, then it will trip the pump.

MEMBER STETKAR: But they are on flow rate. It has nothing to do with sensing what is going on in the seal package. It is simply on flow rates or the component cooling water and the seal injection flow?

In other words, what I am trying to understand is, is there for example a high temperature in the seal leak-off line trip or something like that?

MR. NEWTON: Yes. There is also a seal one, a seal two cavity temperature.

MEMBER STETKAR: There is?

MR. NEWTON: Yes.

MEMBER STETKAR: Okay.

MR. NEWTON: And there is also a seal return flow --

MEMBER STETKAR: Okay.

MR. NEWTON: -- if that is excessive but that will do the same thing.

MEMBER STETKAR: Good. Is it, I have never seen this seal design before. Is this seal design installed in any currently operating plants?

MR. NEWTON: Yes. Could I ask Marty Parece, since you are familiar with --

MR. PARECE: I was distracted. Ask the question again.

MEMBER STETKAR: Is this particular seal design, stand still seal with the nitrogen supply, installed in any currently operating plants?

MR. PARECE: Not, it is not.

MEMBER STETKAR: It is not. So this is --

MR. PARECE: This is a first of a kind.

MEMBER STETKAR: First of a kind. Okay.

MR. PARECE: And we have got full-scale testing planned.

MEMBER STETKAR: Planned. That is the next thing I was going to ask you is have you done any full-scale testing at rated temperature and pressure to measure seal leak-off flows.

MR. PARECE: Well, I think we have done some testing of a scaled seal but our full-scale testing for the U.S. EPR which includes station blackout conditions in pressures and temperatures, that testing is planned for the future, before we deploy.

I don't know what the date is for that testing. Our pump expert is actually in Europe.

MEMBER STETKAR: You were careful to say station blackout conditions, which to me connotes a possible limitation on duration.

MR. PARECE: Not exactly. What it denotes is the most severe conditions, temperature and pressure conditions.

MEMBER STETKAR: Okay, that is what I wanted to hear because I am familiar with other test programs where they pump it up and heat it up and let it sit there for 24 hours and measure flows. I hope you are going to be doing something like that.

MR. PARECE: We will be using the pressure and temperature profiles from our station blackout analysis in those tests.

MEMBER STETKAR: Okay, good. Thank you. And that is to be done.

MR. MATTHEWS: Okay, this slide shows the U.S. EPR steam generator. It is a fairly typical U-tube, inverted U-tube steam generator. It is a slight scale up of a generator that has been operating in the N4 plants in Europe for 15 or so years now.

Some features that we have done to improve the generator from current plants, one example is the tube support plates have been made from martensitic stainless steel, instead of carbon steel and they are not drilled holes, they are trifoil holes. That is to reduce the affinity of the tube support plates to attract corrosion products. And with the trifoil holes, there is less ability for things to accumulate in the hole and improves the flow through the tube support plates.

The tubes are hydraulically expanded in the tube sheet, all the way through the tube sheet, instead of mechanically expanded. That reduces residual stresses, particularly re-transition from the expanded region to the unexpanded region. It is thermally treated Alloy 692.

Some other things we have done, the feedwater nozzle has been designed to, it is in an upslope to reduce thermal stratification in the feedwater nozzle. The feedwater header is designed to stay foiled to eliminate water hammer.

There is an axial economizer and you can see that in the picture in front of you. It is like a second tube bundle wrapper and all the feedwater goes through that tube bundle wrapper just to improve heat transfer between the cold side of the tubes and the feedwater before it goes down to the bottom or to the top of the tube sheet and then gets directed back up through the tube bundle.

So it is just an evolution of current technology.

MEMBER RYAN: Is there anything operating with this technology now with these upgrades?

MR. MATTHEWS: These generators, the EPR generator is just a very slight scale up of generators that have been operating in Europe for about 15 years now with a very good performance history.

MR. PARECE: This is Marty Parece. These are specifically operating at the N4 units in France.

MEMBER RYAN: Oh, okay.

MR. PARECE: And those units operate at a thermal power, about 4250 and this is a 4590 unit. So, it is a very small scale up. The biggest thing we did was stretch the drum so it had more water in it.

MEMBER RYAN: Thank you.

CHAIR POWERS: When I looked at the lower inlet plenum, it looks like your feed into the lower inlet is shallow, flat. So that in a situation of an accident, say a station blackout accident or a steam generated tube rupture accident, that you have less opportunity for mixing of the hot feed with the return gases. And so you have a higher potential for induced steam generator to rupture.

MR. NEWTON: Marty can you take that one?

MR. PARECE: I couldn't hear the hole question. Dana, you are a low talker.

CHAIR POWERS: Gee, I have never been accused of that before.

One of the issues that we worry about is induced steam generator tube rupture during the course of an accident. And a lot of plants get out of the woods there because when the loop seals are in tact, you get mixing of the lower plenum between the hot inlet gases and the return cool gases and so you don't have hot gases coming from the reactor vessel going right into the tubes. They don't overheat very much because of the mixing.

And it looks like, I can't guarantee that it is true but it looks like it is a strong function of the amount of volume you have in the lower plenum and where that inlet comes in on how much mixing you get in there. And when I look at your design, it looks like it is a fairly flat entrance. It is fairly shallow. There is not much opportunity to design.

Now I will admit that I am not very good at CFD calculations in my head or on a computer but it certainly looks like it is more susceptible to inducing an steam generator tube rupture because you do not get adequate mixing of the vapors in that lower plenum.

MR. PARECE: Well maybe the figure you are seeing, maybe it is partially an illusion.

CHAIR POWERS: It could be.

MR. PARECE: Because frankly the plenum to tube and nozzle geometries are very similar to existing units. So it is really not that different.

CHAIR POWERS: You are speaking of existing French units and I am speaking of --

MR. PARECE: No, no. The existing French units, the lower heads on the existing French units, remember the French technology started with a license from Westinghouse for a plant very similar to Beaver Valley. And so the original steam generators designed for the French fleet looked very much like Series 44 and Series 51 steam generators and the evolution has occurred since then. But the angle of the nozzles and plenum volumes are very similar, the only difference being some change has been made to replacement steam generators to allow a flat bottomed lower head so that you can get some easy draining when you drain down for maintenance.

Regardless, our inadequate core cooling guidelines will drive you to open the primary depressurization valves and blow the plant down in pressure before you get to a temperature that could cause the tubes to melt. So that is handled as our going from inadequate core cooling to severe accident guidelines.

CHAIR POWERS: We will come back to this when we go back to Chapter 19. Heads up, Sandra.

MR. NEWTON: Okay, the next slide shows reactor coolant piping. Reactor coolant piping is forged austenitic stainless steel. And it has pipe bends in it to reduce welds. And you can see how these are the cold leg pipes and you can see how the pipe has been bent here to avoid having to weld on a nozzle.

In addition, the large nozzles are actually forged with the pipe. And in the drawing here, these are the safety injection nozzles. Over here, a hot leg pipe and right there you can see the surge line nozzle. And on this hot leg pipe, you can see the residual heat removal system nozzle.

MEMBER SHACK: Is this the second set of pipes for the Finnish reactor?

MR. NEWTON: I don't know which set it is.

In addition, we have the nozzles that are attached to the pipe. They are internally machined so that the diameter is small enough so that if a pipe break occurs on that nozzle, it would be within the capacity of the CVCS pipes.

Over here on the corner, I didn't point that out, those are the surge line pipes.

Any questions about the reactor coolant piping?

CHAIR POWERS: Let me ask you a question. You may not know the answer to this but I can satisfy myself. How thick of a corrosion layer do you expect to get on these things during normal operation?

MR. MATTHEWS: I am not sure I can answer that. We might have to get back with you on that.

MS. SLOAN: Yes, I think we will just have to take a note, Dr. Powers. I am not sure we have the right people here to answer the question.

What is the thickness of the corrosion layer?

CHAIR POWERS: Yes. And then as soon as they answer that I am going to say, okay what is it made out of and what does it do and things like that.

MR. NEWTON: Okay, the next slide is the residual heat removal system. The residual heat removal system has four physically separated trains that are completely independent from each other. And the design system conforms to Branch Technical Position 5-4. The layout and configuration of the system is such that it is self-venting to prevent any voids from accumulating in the residual heat removal piping.

Regarding the performance of the system, it can cool from 250 degrees to 131 degrees for refueling in 15 hours. For obtaining a safe cold shutdown, it can go from 250 to 200 degrees in 2.7 hours. That is assuming that you just have two trains.

CHAIR POWERS: What do you mean not much to say about it? There is a lot of it. Just joking with the guy. He says oh, we only have four trains. We were going to put in eight but --

MEMBER STETKAR: Dennis, you have mentioned a couple of places this self-venting capability. Could you elaborate on that and what you mean by self-venting and where the vents are installed?

MR. NEWTON: Oh, okay. What I mean by that is that the configuration of the layout of the piping is such that it is always flowing, going up. So you can't have any gases accumulate in the pipe because it will always flow up. There are no high points in the pipe --

MEMBER STETKAR: Well there must be a high point somewhere.

MR. NEWTON: -- to accumulate the gases.

MEMBER STETKAR: Is there no, I mean between the loops and the pump suction --

MR. NEWTON: Yes, between the pump suction and the --

MEMBER STETKAR: -- are there any other high points on the discharge lines where gas could accumulate or are you not worried about those?

I mean, you tend to be less worried about those because they are on discharge lines.

MR. NEWTON: No, the configuration is such that no, we don't have any high points.

MEMBER STETKAR: So it is just the configuration of the piping.

MR. NEWTON: Yes, it is just the configuration of the piping.

MEMBER STETKAR: Okay. A few other questions on this, since this is our only shot, apparently, at the RHR system because it is for some reason in this chapter.

There are, as in all plants, there are interlocks that apparently prevent you from opening the hot leg suction valves at a pressure above 464 pounds and temperature above 356 Fahrenheit. I read in a couple of places it said automatic isolation is not provided.

So, if I am operating the system, if I have aligned it and for some reason primary pressure increases above 464 or primary temperature increases above 356, the suction valves do not close, re-close automatically?

MR. NEWTON: That is correct.

MEMBER STETKAR: It is strictly a manual operator action.

MR. NEWTON: Yes.

MEMBER STETKAR: Okay.

MR. NEWTON: You do have the low temperature over-pressurization valve.

MEMBER STETKAR: I understand that, as do most plants.

There is also a discussion that there is apparently an automatic trip of the pumps in the event of low level. And this is the mid-loop operation protection.

How are the RHR suction lines configured with respect to the loops? Are they dead bottom-center or are they --

MR. NEWTON: Yes, they are.

MEMBER STETKAR: They are.

MR. NEWTON: And in the hot leg pipe,

there is a level measurement in each hot leg pipe. And there is a nozzle at the top and the bottom pipe that are measuring the level during mid-loop operation.

MEMBER STETKAR: Okay. What level? Where is the trip setpoint? I mean, is it six millimeters above the bottom dead center or is it --

MR. NEWTON: I don't have that information with me.

MEMBER STETKAR: Okay.

MR. NEWTON: The actual setpoint.

MEMBER STETKAR: You know, a lot of times these drain down things, if somebody opens up a line, happen pretty quickly. So that is why I was curious about how much margin do you have for the trip setpoint to actually, you know, losing MPSH. I guess that may be a Chapter 7 thing.

MR. PARECE: Well this is Marty Parece. It is going to be one of those integrated system control system questions, too because if we get a low level in the hot leg pipe, the first thing that would happen would be actuation of a medium head safety injection pump to pump the water all the way back up.

MEMBER STETKAR: Oh, I didn't read it. Is that right?

MR. PARECE: I believe that is correct.

MR. NEWTON: That is correct.

MEMBER STETKAR: So you actually get an automatic injection on that low level setting.

MR. NEWTON: Yes.

MEMBER STETKAR: Oh! I didn't read about that anywhere. Thank you. I'll make a note here.

And I think I had two or three others but they are pretty minor. The only other one I had, the staff -- somebody raised questions about the RHR pump mini-flow and test line recert lines. So I dutifully went and looked at that. And this is more just educate me, please.

I notice that whether you call it a minimum flow line or a test line, there is a single line that comes off and then it divides into two lines that are labeled, at least on the drawing that I was looking at, as a radial mini-flow and a tangential mini-flow. In other words, from the discharge of the pump, there is a single line that comes off and then it branches into two parallel lines before it goes back to the IRWST or someplace. And those are labeled radial and tangential mini-flow. And I had never heard those terms so I was curious what they meant.

MR. BANKE: This is Jim Banke from AREVA.

The radial tangential lines allow you to, when you recirculate the IRWST, instead of just sucking or drawing liquid from one point and discharging it to one point and you have literally a 500,000 gallon tank, what they do is that the radial and tangential lines will distribute the fluid evenly to create a greater turnover.

MEMBER STETKAR: Okay, that is what I was guessing.

MR. BANKE: And that is what allows you to do --

MEMBER STETKAR: The drawing that I saw, you know, only had a couple of arrows going out.

MR. BANKE: When you recirc the tank, you get a nice good sample. You turn it over quickly, rather than waiting for days.

MEMBER STETKAR: Good. I was guessing that but thank you very much.

MR. NEWTON: I have some information about the material of the reactor coolant pump flywheel. Would you like to have that? It is made of SA-508 Grade 4N. It just came in.

CHAIR POWERS: Hot off the presses from the design branch. Thank you. Please continue.

MR. NEWTON: Any more questions on the

residual heat removal system?

CHAIR POWERS: Just four trains. Right?

MR. NEWTON: Yes, four.

CHAIR POWERS: We will struggle along, limp along. It will require some pretty good quality control here.

MR. NEWTON: Okay, the next slide is the pressurizer. The pressurizer is made out of low alloy steel and is clad with stainless steel. The volume is at about 2600 cubic feet and the water level is maintained about half-way, so there is 50 percent gas, 50 percent water.

I had mentioned about the water collectors. And this is the water collector at the inlet to the nozzle that goes to the pressurizer safety relief valve. Over here you can see there are three of these pressurized safety relief valve nozzles and they each have one of those water collectors.

There are three spray nozzles, one, here, and here. There are two for normal spray and they come from two different reactor coolant system loops and the auxiliary spray comes from the chemical and volume control system.

The spray valves, they work in unison. So they modulate together, both spray lines. And they both have a continuous spray flow, 0.77 pounds per second in each line to maintain the chemistry of the pressurizer and to keep the spray line and the surge line at the appropriate temperature.

The heaters, down here at the bottom, so you don't have to count all these dots, I will tell you there are 116 elements. There are two groups that are powered off the Emergency Diesel Generator for a total of 288 kilowatts each. We only need 130 kilowatts. In addition, you also have a proportional heater banks and you have on and off control banks.

Then there is the one nozzle here for the primary depressurization valve. After you come off of this nozzle, there is a water seal before you get to the valve. There are two flow paths after this nozzle and each flow path has a gate valve and a globe valve for primary depressurization.

MEMBER STETKAR: Dennis?

MR. NEWTON: Yes?

MEMBER STETKAR: Where in the FSAR are the primary depressurization valves and all of their controls described? I couldn't find them anywhere. They are not in this chapter of the FSAR.

MR. NEWTON: Well the primary

depressurization valves are discussed in this chapter. It just, it doesn't get into controls during operating procedures.

MEMBER STETKAR: It doesn't discuss the valves either, I don't think. Does it?

MS. SLOAN: You are looking for the components themselves?

MEMBER STETKAR: Yes. Yes, I am a valve kind of guy.

MS. SLOAN: We will have to check the FSAR.

CHAIR POWERS: We've worried about you for years on this.

MR. NEWTON: They are shown on the figures that go along with Chapter 5.

MEMBER STETKAR: Oh yes, they are. I mean, I was trying to figure out how big they are and --

MS. SLOAN: We will check in five and we will also check six and 19.

MEMBER STETKAR: I looked in six. I mean, 19 talks about functionally what they do.

MS. SLOAN: Yes.

MEMBER STETKAR: But you know, I happen to know that they are motor-operated valves but osmosis is one thing, reading about things is something else.

MS. SLOAN: We will check in the FSAR.

MEMBER STETKAR: Okay.

MS. SLOAN: We can get you a specific FSAR location. It looks like you have checked the obvious places, though.

MEMBER STETKAR: I looked here. I did word searches. I looked in six. I looked in -- I mean 19 talks about from a functional perspective what they are used for but not anything about the beast itself. There is a lot of discussion in this chapter about the PSRVs but I couldn't find the --

MR. NEWTON: Anyway, the capacity is about two million pounds per hour --

MEMBER STETKAR: Yes, that is why I was kind of interested.

MR. NEWTON: -- at 2535 psig.

Anymore questions about the pressurizer? This is a pressurizer surge line and the main reason that it is here is to show that it is a continually -- that the pipe is continually graded so that you have a continuous flow going through it and you don't have any low points that can create issues with thermal stratification. And the continuous spray flow is enough to keep water continually flowing through the pipe. And that is the point of this particular slide.

Then we have pressurizer relief tank. It is a stainless steel tank. It is a 1413 cubic feet. It is about 80 percent full of water and it is designed to collect and monitor condensate discharge from the pressurizer. It is sized so that it can accommodate the discharge from a turbine trip and it is also sized so that we can do in situ testing of the pressurizer safety relief valves during heatup.

It has two rupture disks and the rupture disks go to tail pipes and the tail pipes discharge where the reactor coolant pumps are so that any discharge will flow back to the reactor building water storage tank. The rupture disks, they are 28 inches. They have a 300 pound psig setpoint. They are kept high enough so that we can do in situ testing without rupturing the rupture disks. The discharge header coming in is 16 inches and we have the sparger there for discharge from the steam and getting it condensed in the water.

MEMBER STETKAR: You said, just to make sure I recall what you said, that the discharge pipe, inlet pipe, whatever I want to call it, indeed is routed through cubicles in what you call the equipment areas of the containment so that if it does rupture, any discharge will eventually find its way back to the IRWST. Right?

MR. NEWTON: Yes. Okay if there are no questions, I will go to the next slide.

The reactor vessel does have high point vents and they are put in to conform with 10 C.F.R. 50.46. They have a very typical configuration. We have two flow paths. And each flow path has two valves. They are each powered from a different power source. It has an orifice here at the end and the orifice again is sized so that if two valves did fail open, the leak rate would be inside the capacity of the CVCS pump. And the valves do have positive position indicators.

Any questions about this? Okay.

Pressurizer safety relief valves. Okay, you have the main valve here. And it has got a disk inside that opens and closes to relieve pressure. The operation of that disk is controlled by either the spring-loaded pilot valve when you are at power or by the two solenoid valves when you are in low temperature overpressure protection.

The two solenoid valves, they are in series so that if one fails to open you can still have one closed so the valve does not spuriously open.

It has got a position indicator to indicate when the valve is open. The valve opens in 0.7 seconds and that includes the operation of the solenoid.

The spring loaded pilot valves are such that they can be taken off and tested to make sure the setpoint is correct. As I mentioned, this is a medium controlled pressurizer safety relief valve. It is used on some nuclear power plants over in Europe.

MEMBER STETKAR: Who makes these?

MR. NEWTON: There are a couple of vendors. There is Simpel, CCI, and also -- well CCI is Simpel now.

MEMBER STETKAR: Okay.

MR. NEWTON: Sorry that ran together.

And the capacity is 661,400 pounds per hour at 2535 psig.

Any other questions about the pressurized safety relief valves?

MEMBER STETKAR: These valves qualified for water relief?

MR. NEWTON: Yes. They will be tested for both water and steam.

MR. MATTHEWS: Okay, the last thing we will talk about is the component supports.

Again, they are Class 1 components so the supports are ASME Subsection NF Class 1 complement supports. Their function of course is to support the components and provide restraint and maintain the components in a configuration where they can perform their safety related functions. And they are, of course, just attached to the reactor building with embedded bolts. And I have got a couple of figures.

This shows the reactor vessel support ring. This reactor vessel is a nozzle supported vessel, which is not unusual relative to the operating plant. What is a little different is that this is a continuous support ring, instead of a collection of individual nozzle supports like a lot of the operating plants have. Each one of these notches is for the bottom of the nozzle on the reactor vessel has a square pad that sits in this notch, so that allows the vessel to expand and contract with thermal expansion. So each nozzle sits in one of these notches.

And these straps here are seismic straps. They are designed to retain the vessel in the event of seismic uplift. So that is the reactor support. And this is where it is embedded into the concrete. And then this shows some of the supports, configurations for the other components.

Steam generator and reactor coolant pump sit on some column-type supports. The nice thing about these columns is it makes it real easy to get to manways during outages.

On the steam generator, you have also got lower restraints that just limit movement of the vessel down here. And then up top you have got two additional supports and two snubbers that are also designed to limit movement of the steam generator. The reactor coolant pump has two snubbers up top, in addition to the three columns. The steam generator has four of these columns and the reactor coolant pumps each have three column supports.

Now in the pressurizer, the supports are actually welded right to the vessel and then the vessel is bolted to the floor and it also has eight horizontal restraints up around the top of the vessel.

The acronyms and abbreviations.

CHAIR POWERS: We wont' make any comments on those.

MR. TESFAYE: We will send them an RAI to describe for us.

CHAIR POWERS: Please. Please do that.

MR. NEWTON: Any comments on the conventional reactor system?

MEMBER RYAN: That is a reactor.

CHAIR POWERS: I am still perturbed about the four trains. That really bothers me as I am struggling along here.

MS. SLOAN: We are done.

CHAIR POWERS: Let's trudge right ahead, then. A little ahead of schedule, which is going to break absolutely no one's heart.

I am going to -- I will interrupt you at 3:00, roughly.

Please lead us on here.

MR. TESFAYE: We will change the order here.

CHAIR POWERS: Like I say, formality is not the hallmark of a subcommittee meeting. It irritate Derek, but that is okay. He is irritated most of the time anyway.

MR. WIDMAYER: Hey, I'm ready for schedule changes. That's no big deal.

MR. TESFAYE: Thank you all. We have several presenters so we are going to be shifting people and we apologize for this disorganization.

CHAIR POWERS: You like the soul of efficiency right now.

MR. TESFAYE: I will like to introduce

again Tarun Roy, he is the Chapter PM for Chapter 5. You have seen him before.

CHAIR POWERS: He has talked to us before. We know all about him.

MR. TESFAYE: And instruction for the people who have been here before, you don't have to repeat your biographical information.

CHAIR POWERS: In fact, we will conduct a test and see how many members of the committee remember.

MR. TESFAYE: That will work. With that, I will just turn it over to Tarun Roy.

MR. ROY: My name is Tarun Roy. I am the NRO Project Manager responsible for coordinating staff review for U.S. EPR FSAR Chapters 5 and 17, design certification.

We have several Chapter 5 technical reviewers here. I can name them right now and they will go by one by one their sections presenting.

Robert Davis is Component Integrity Branch. We have all of them Component Integrity Branch Thomas Scarbrough, Jeffrey Poehler, Timothy Steingass, Joel Jenkins, Steven Downey, John Honcharik and Gregory Makar. And we are from EMB group, John Wu. Then we have Balance of Plant Li Chang-Yang and then we have Reactor Systems, Nuclear Performance and Code Review Branch John Budzynski and Shanlai Lu.

During this meeting the staff plans to make a presentation of the Chapter 5 Reactor Coolant System and Connected Systems Safety Evaluation Report with the open items.

Staff issued a total of 127 questions to the Applicant requesting additional information. Out of that 127 questions, there are 25 open items identified in this SER with an open item. The staff will discuss these open items in detail.

The U.S. EPR FSAR Chapter 5 was issued in a publicly level document. With that, I now turn presentation over to the technical reviewer John Wu and for the Section 5.2.1.1 of the Engineering Mechanics Branch. John.

MR. WU: Okay. Today I am talking about 5.2, Section 5.2.1.

MR. TESFAYE: Just for the record, we are not going to go through the list of open items. They are there for your information. We will start ahead to the section presentations. Go ahead, John.

MR. WU: Okay, just go directly to the section review. Section 5.2.1.1 is related to Codes and Standards combined with 10 C.F.R. 50.55a.

And in the last section you heard for U.S. EPR there is code of the record. They are design codes of ASME 2004 Edition with no addenda. They also use the 1993 addenda to 1992 Edition for piping seismic design, in order to comply with 50.55a item (b)(1)(iii).

As you know, the Code edition and addenda is a Tier 2 start of information. So anyway, if you a difference, different than the code I just mentioned, if a different code edition and addenda is planned to be used, then the COL applicants must identify the code edition and addenda in the COL applications for the staff to review and approval.

Now, we looked at the code edition addenda they used and to satisfy we look at the 10 C.F.R. 50.55. We got one open item. The open items is they used the 2004 Edition and also the 1993 Addenda to 1992 Edition. But these two editions are not accepted by 50.55a(b)(1)(ii) regarding the weld leg dimensions and the applicant did not explain how they are going to meet or satisfy this requirement. So we think it is out there, their response is. They need more detail to add to this, how are they going to deal with this.

But actually this mostly we are talking

about some kind of undersized weld issue. So this is an item that we talked to the applicant and this will be resolved in the near future.

CHAIR POWERS: And we are still waiting for a response?

MR. WU: That's right. Still waiting for a response. That is correct.

MR. ROY: Okay, 5.2.1.1 and 5.2.1.2 is done.

MR. WU: Yes, 5.1.2 because 5.1.2 is already mentioned by applicant. So we had no open item. Yes.

MR. TESFAYE: If you don't mind, we would like to jump to 5.2.3 while Robert Davis is still here and we will come back to 5.2.2.

MEMBER STETKAR: You are going to come back to it?

MR. TESFAYE: Yes.

MEMBER STETKAR: Okay.

MR. TESFAYE: Robert?

MR. DAVIS: My name is Bob Davis and I am the primary reviewer for reactor coolant pressure boundary components.

Okay with the exception of the material listed for the control rod drive mechanism which we discussed this morning that isn't in the code, the exception of that issue, all reactor coolant pressure boundary materials specifications that have been selected by AREVA meet all ASME Section III requirements and all the ferritic materials meet the fracture toughness requirements of Section III. Go the next slide, please.

All reactor coolant pressure boundary materials exposed to reactor coolant are either stainless steel, nickel base alloys, or ferritic materials clad with stainless or nickel based alloys. Only Alloy 690 is reused. Alloy 600 is not used in the reactor coolant pressure boundary or its associated metals.

Unstabilized austenitic stainless steel is low carbon. Materials and processing conform to the guidance in Reg Guide 1.44, which is guidance for welding and processing of austenitic stainless steels to prevent sensitization.

The RCS chemistry is evaluated under SER 9.3.4 and the guidance in Reg Guide 1.36 for the control of leachable contaminants in thermal insulation is followed by AREVA.

The things in AREVA's design that are different from previous designs are one is the forging, the forged pipe, which is we see this in this design and there is another vendor design that is going to used forged pipe but we haven't presented that material here yet.

But other than the forged material, one of the biggest differences is in dissimilar metal welds. We are all concerned about dissimilar metal welds because there is a history of issues with those types of welds, whether they are in a fossil plant, petrochemical plant or a nuclear plant, they seem to always pop up as an area of interest and an area of problems.

In the typical existing plants for all the primary welds, dissimilar metal welds, a buttering was applied to the low-ally steel material and then it would be post-weld heat treated and then a safe end would be welded onto that later, a stainless steel safe end would be welded on to that later or maybe in the field of RCS piping would be welded directly to that. I believe both ways it has been done.

In the EPR design, instead of having using buttering, they are going to weld the low-alloy steel directly to the safe end without the use of any buttering, which is substantially different from the previous designs.

So in order to get more information about this process because they use what is called a narrow gap design which these welds are only a half inch wide but they are vertically -- they are pretty much vertical straight up and down. There is really not much of a bevel at all on the sides.

We conducted an audit at AREVA's office here in Rockville, where they made a presentation to us and we reviewed several documents associated with the research and development of this process that they did in France. They did a substantial amount of work.

This type of welding has been used on OL3 and I think one of the new French plants they have fabricated some components so far. I didn't put it in the slides but I do have a few slides for some of the members here, if they would like to see them.

And as you can see from those, the differences in the two methods. One of the very -- the two very good things about this method is that the passes are very thin. Therefore, it eliminates a lot of the issues associated with ductility duct cracking. So the welds are extremely high quality and the nozzle forging in the safe end are placed in a fixture and they are welded in a flat position so it is a fixed-head gas tungsten arc weld torch. So you are always welding in the flat position, producing extremely high quality welds and it results in a weld volume that are substantially smaller than a typical dissimilar metal weld, making a much smaller volume to inspect.

And in addition, when the welding is complete, there is extra material left from the ID of these. It is machined out. So basically the first three passes of the weld are machined off from the ID, resulting in the elimination of the area of the weld that would have the highest amount of dilution of ferritic material, thus increasing the level of chromium in that area that is exposed to reactor coolant.

Only minor surface repairs are allowed. Any internal defects that are in excess of what is allowed by code, they cut the whole thing off and re-weld it. So this is quite an improvement on their part.

The other dissimilar metal welds besides the safe-end welds for the CRDM tube to reactor pressure vessel closures, those are partial penetration J-groove welds which have been the result of a lot of attention. And so those welds are extremely challenging to make and are very prone to welding defects.

AREVA has special testing requirements for every lot of filler metal that they use to ensure that it is less susceptible to ductility cracking than lots that could be more susceptible. So they have a supplemental testing program for all Alloy 52 filler metals. And those welds are also subjected to a surfacing with a flapper wheel to reduce surface stresses so as to make them as low as possible.

Are there any questions about these metal welds?

CHAIR POWERS: I mean, your judgment is that these are all great things. You know, they have fixed a lot of problems in the past and whatnot. But it didn't answer the question. How do we know it is any good?

MR. DAVIS: Well I mean they did several mock-ups. They have had quite an extensive testing program and they conduct a lot of mechanical testing, corrosion testing. I believe that they went about as far as they could go for testing these welds.

I mean, now narrow gap welding in itself is not new. I mean, narrow gap welding is used in a lot of different industries. It is just new during these types of welds. And narrow groove type welding is not really -- welds this narrow are kind of new but all of your steam generator replacements and all those things are done with a very narrow weld joint design. This is different because of not buttering.

CHAIR POWERS: Right.

MR. DAVIS: But so the information that they have provided, I mean, it seems like they have really done their homework and they have done as much testing I think as someone could expect them to do. But you are right. There could be problems that we don't see. And I guess as with anything that you use, as with Alloy 690 or anything else that you use that is new, it certainly shouldn't preclude somebody from doing something new that could be better or should be better.

MEMBER SHACK: What does the weld look like at the steam generator to pipe?

MR. DAVIS: I will have to remember back. Calvert Cliffs is the only one that I have experience with.

MEMBER SHACK: No, I meant in this plant.

MR. DAVIS: Oh, for the safe ends?

MEMBER SHACK: This -- we are talking reactor vessel welds here or are they same on both?

MR. DAVIS: Oh, they are all the same --

MEMBER SHACK: They are all the same.

MR. DAVIS: -- with the exception that I believe the reactor pressure vessel is, they actually do the nozzle in the safe end and a fixture. There are some where I believe they actually weld it right to the head. But in all cases, they turn the entire component or they just have the nozzle in the safe end. In all cases, they are welded the same and they are welded in a flat position. And in other differences, these safe ends will be post-weld heat treated but they are low carbon. And we have an RAI out asking additional information that but they have been doing it for years in France where they have post-weld heat treated low carbon nozzles and haven't had any issues associated with that.

Okay, we can go to the next slide, which is the open items. The first open items is the same as the open item for Chapter 4 for the use of the martensitic stainless steel material.

In order to provide consistency with ITAAC, we would like ITAAC for all Class 1, 2, and 3 components to be the same throughout the entire application. So we have, our RAI on ITAAC we have left open until we are sure that all of the ITAAC associated with ASME code materials are the same for welding.

And then their table for reactor coolant pressure boundary materials does not list weld filler materials for reactor coolant pressure boundary piping steam generators or pressurizers. We are still waiting to hear back on that.

In addition, we have asked them to alter their limits on ferrite, reduce their limits on ferrite for cast austenitic stainless steels contained in moly to address thermal aging embrittlement.

The next open item, originally AREVA had indicated that they were going to use a stabilizing heat treatment on their Grade 347. And then later on they said that they weren't. So there was an inconsistency. And we are waiting for them to address that.

And then the last one which I just discussed we asked them to discuss their post-weld heat treatment of stainless steel safe ends and the corrosion testing that they will do on their qualifications.

Now does anybody have any questions?

(No response.)

MR. TESFAYE: If there are no questions for Bob, I would to turn to John and Shanlai. Go back to slide number 10.

MR. BUDZYNSKI: Slide number 10 is a summary of the open items. The first item there is RAI 332, 5.02.02-3. And what we asked the applicant to do was three things. It was three things, to identify the analytical methods used in LTOP analyses, justification for the events that were selected, and compliance with GDC 31. And basically, they did answer. They responded but we need some clarification on the justification with compliance to GDC 31.

The second RAI, 5.02.02-12 is the PSRVs and the PDS valves have loop seals. And with respect to NUREG-0737, there should be a loading analysis because the loop seals have been demonstrated during discharge of the safety valves. You can create large forces on the loop seals. And so that is still waiting for a response on that one.

MR. LU: I think this in line with John's question. You were asking --

MEMBER STETKAR: Well, I was looking at the downstream side of it but I mean, it is the same kind of thing.

MR. LU: So the staff has been asking questions. We are waiting for responses. Once the responses come in, we will figure out what do with that.

MR. BUDZYNSKI: The next slide. Okay, there are no COL action items and the design is confirmed by initial test program. We reviewed the test program and it looked sufficient and also by tech specs.

And the test program was in various sections of the start-up test in the RCS, RHR, and also in the main steam line generator. They all had sections, parts that related to the overpressure protection. That's it.

MEMBER STETKAR: Well on the overpressure protection, it is not directly relevant but it is kind of related and it gets back to my question to the applicant regarding where in the FSAR they discuss, to get my acronyms correctly, the PDS valves or SADVs.

The pressure-to-pressure, the primary depressurization valves, where in the staff's review, what area do you actually look at those valves? They are kind of strange because they are severe accident related valves. So they are not, you know, design-basis accident related but curiosity in terms of looking at the valves evaluating will they work, for example.

MR. BUDZYNSKI: Well using the SRP for

this section, I had to search through.

MEMBER STETKAR: Well they are different beasts because we don't have them, for example, in any of our plants.

MR. BUDZYNSKI: Yes.

MEMBER STETKAR: So --

MR. BUDZYNSKI: I had to search through the sections and I found parts in several different sections throughout the FSAR that discussed --

MEMBER STETKAR: But I mean, yes. Well, they may be scattered around. I guess what I am asking is from our perspective, will there be a review performed on those valves and under what chapter of the SER? I mean, it is obviously not a Section 5 or a Chapter 5 issue or there would have been some more discussion of them in here.

So is it Chapter 6? Is it chapter which?

MR. LU: I think the answer is I think from the reactor system perspective, we are mainly looking at the system response for the load issue. And then in terms of a component, I think that is maybe other branch staff can comment on that, specifically for the walls and then that is nor normally our responsibility for our branch.

MEMBER STETKAR: Yes, okay.

CHAIR POWERS: We can take an action on that.

MS. SLOAN: Can I mention one specific FSAR section? We were looking --

MEMBER STETKAR: Were you looking? You found something?

MS. SLOAN: We were looking. So maybe this will help. And I don't know if you have already found this in 19. You said you had looked one place. So the section we have come up with, it is a long one, 19.2.3.3.4.1.

(Laughter.)

MEMBER STETKAR: Let me see if I could read this back from my notes. It would 19.2.3.3.4.1?

MS. SLOAN: You got it.

MEMBER STETKAR: You can also find something on 19.2.4.4.3. I found those. They are basically only summary information about sort of functionally what the valves do and nothing much about it.

CHAIR POWERS: We now have proof that some members of the committee have absolutely no life whatsoever.

(Laughter.)

MEMBER STETKAR: And proud we are of it.

MS. SLOAN: Okay we were looking on the computers. The information level --

MEMBER STETKAR: I mean, Sandra, obviously the SER of Chapter 19 didn't do any evaluation of the valves themselves, whether they worked because the SER for example of Chapter 19 is focused entirely on PRA-type issues.

MS. SLOAN: Well 19.2 is severe accidents.

MEMBER STETKAR: Yes.

MS. SLOAN: So I am wondering if maybe in 19.2 it is just --

MEMBER STETKAR: No. I did the word search in the SER also. I couldn't find them.

MR. COLACCINO: If I could, this is Joe Colaccino, the EPR Projects Branch Chief. I think we are going to take this back and see if in fact the staff, if anybody has looked at this and where it has been characterized in the SER. Because we don't want to guess here at this point.

MEMBER STETKAR: Yes, it is a gray area because it is not a design basis accident response. So it is not necessarily a Chapter 6-type issue. And yet they are relatively important valves that are directly connected to the primary system.

So you kind of want some sort of assurance that the valves themselves are going to work when you want them to work and not work when you don't want them to work, if I said that correctly.

CHAIR POWERS: One of those ways.

MR. COLACCINO: Again, I think the action for the staff is to go back and look and see where this review has been characterized, if at all, and get back with the committee.

MEMBER STETKAR: Thanks.

MR. TESFAYE: Now we will continue with Section 5.2.4 and 5.2.5. Tim Steingass and Li Chang-Yang.

MR. STEINGASS: My name is Tim Steingass and I am the primary technical reviewer for inservice inspection programs for all the design centers.

During the review of the EPR, we found that the operational program -- let's go with the slide here. What is the slide here? Seventeen, there we go.

We found that the ISI operational program complies with the requirements of the ASME Code Section XI and 10 C.F.R. 50.55a. We were particularly pleased to find that the design enables the performance of all the inservice inspections required by Section XI through elimination of interferences due to design, material condition, or material selection and geometry.

Earlier, a couple of times, the discussions wandered off into the area of Alloy 690 and statements were made about Alloy 600 how we thought it wasn't going to crack and yet it did. Well with 690, we are extremely interested in assuring that the design enables the performance of inservice inspection so that if anything should happen, there is a reasonable methodology out there that can monitor the design and determine whether or not there is going to be a loss of structural integrity and the right decisions can be made with regard to repair or replacement or overlay or whatever.

And this also satisfies the requirement under Part 52, which says that operational issues will be incorporated into the design. So we feel that not only compliance with the regulation in Section XI is important, but primarily the ability to perform the inservice inspections provides for a more robust design and a robust operation.

Secondly, the other thing that we looked at was not only the essential elements under the SRP for the ISI program, which were all there under AREVA and the design but the individuals that will be performing the non-destructive examinations will meet the requirements of ASME mandatory Appendices VII and VIII, which require that personnel, the equipment and the procedures are all demonstrated and qualified.

So in short, the ISI program meets Reg Guide 1.26 as it relates to the quality group classification of components. Reg Guide 1.147, as it relates to the code cases, which was discussed earlier.

And let me say one thing with regard to code cases also. Because the design enables the performance of inservice inspection, there will not a tremendous number of requests for relief because of impracticality. As a matter of fact, we won't approve any.

(Laughter.)

MR. STEINGASS: And that is not just for this design center. This is for all design centers.

CHAIR POWERS: This is absolutely an essential feature of the design because of the established reliability of assurances from the metallurgists that the material will not crack.

MR. STEINGASS: Absolutely. Also as mentioned by ASME Code Case -729, as incorporated into the design, which enables the performance of the visual inspections and access for ultrasonic inspections to meet the order as it has been incorporated into the regulations.

Finally, NRC Bulletin 88-05, as it relates to the establishment of a program which is used to detect and correct reactor coolant pressure boundary corrosion caused by boric acid. And 88-05 consists of four elements that are required to be in the operational program to walk down, to monitor, to identify, provide a methodology for evaluation of any issues that may evolve due to boric acid corrosion.

Finally, there were no open items related to the inservice inspection operational programs for this design.

Are there any questions?

CHAIR POWERS: Pretty good brief.

MR. STEINGASS: Thank you.

CHAIR POWERS: Pretty good brief.

MR. LI: I am reviewer Chang Li, reviewer from Balance of Plant Systems Branch for Section 5.2.5 reactor coolant pressure boundary leakage detection.

My review is calling to Reg Guide 1.45 for the compliance of GDC 30.

MR. TESFAYE: Chang Li, could you please

give your background information?

MR. LI: Okay. The review that we go through using the Reg Guide 1.45 going into the review of the leakage detection sensitivity response time capabilities tech spec requirements and seismic qualifications. Of those we found they are acceptable in accordance to Reg Guide 1.45. There are only areas that we found in terms of the procedures requirements in the Reg Guide 1.45.

The procedures required COL applicant to develop some procedures to convert instrument indicators to a common leakage equivalent. In other words, if you got an instrument reading of microcuries per cc's of leakage indicated in the containment radiation monitor, you want to have a quick conversion into how much that is correlated to the leakage of gallons per minute. So that is the procedures we are talking about.

And also looking for the alarm setpoint such that they would have early warning to the operator before they reach the tech spec limit like one gallon per minute tech spec limit. You don't want to wait until it shuts down. You need some early warning signals from the alarm.

So these are the procedures that we are

looking for from the Reg Guide 1.45.

MEMBER STETKAR: Li, on numbers nine and ten open items, you know, I read in the FSAR that they have committed to conform to the guidance in Reg. Guide 1.45 Revision 1. It struck me as curious that the types of issues that you are raising in nine and ten should be open items for the design review. They strike me more as items that should be perhaps applicable to the COL applicant because they are plant-specific procedures.

In particular, number nine, I was curious why a unit conversion requirement in a plant procedure is an open item for design review.

MR. LI: I would like to ask the applicants to answer that question. They defer that into the COL. Where in asking the questions from design certification point of view and the questions, they are answering this as well, that is going to be developed because of the instrument. They are plant-specific design instruments, for example, irradiation or something in terms of their design.

I would like applicants to address this question how they are --

MEMBER STETKAR: Well, no. I am asking you as the reviewer. Because I understood -- I read the questions and the responses. And as I understood it, AREVA's response was it is a COL applicant issue to develop the plant-specific procedures and, you know converting Unit X to Unit Y so that some operator understands.

I am asking from the staff's position as far as the design certification review why in particular number nine and related number ten on this issue has been raised to the level of open items for the design certification review. They strike me as being more operational COL-applicant issues.

So I would like you to explain to me why it is a design certification issue, not --

MR. LI: Oh. We review start from Reg Guide 1.45. So all the review, whether it is operation or whether it is design, it is all, we go one-by-one in that Reg Guide.

And these two issues which are in the operation. So the information that developed these procedures may be able to, in the design stage, AREVA may not be able to specify that. So it has been deferred to them for the COL to develop those procedures. That becomes the COL action items.

However, in the current process, the COL application make the application. They use incorporate by reference without additional information. And they assume all the Reg Guide 1.45 are conformed and are reviewed of design cert.

So there is no procedures being provided from the COL applicant in the COL application. Therefore, we have to ask well, in between the design, AREVA and the COL applicant who is going to provide that information. In the process, if COL doesn't have a COL information items, they use the process, take advantage of the process, incorporate by reference without additional information, it is empty there.

So between the designer and the COL, there is no one to provide that information. That is why we are asking this list of open asks AREVA to specify a COL information items so that it drives the COL applicants have to respond to address. Therefore, I can ask procedure from the COL. Otherwise, they are going out in this way.

MR. TESFAYE: I think, if I can take a crack at this, this is a licensing type question. I think the inclusion of COL information items is design certification activity.

MEMBER STETKAR: Yes.

MR. TESFAYE: So that is why this is --

MEMBER STETKAR: So if I understand it,

you are basically asking the design certification applicant if they inserted a COL information item saying the COL applicant must develop procedures that conform to Reg Guide 1.45 revision one, that would satisfy you.

MR. TESFAYE: Yes.

MR. LI: That is correct.

MEMBER STETKAR: And it is not necessarily down to the level of what type of unit conversions you need to use or the details --

MR. LI: Not at this stage.

MEMBER STETKAR: Okay.

MR. TESFAYE: This is a tracking mechanisms.

MEMBER STETKAR: Okay. Then, I understand the concern.

MR. LI: It is just a process. Where is that process? Where is that CRA information item for applicant? That I have basis to push for additional questions from the COL applicant.

MEMBER STETKAR: I was just hanging up on -- I mean, these are two, as they are worded, they are two rather detailed type issues that you wouldn't normally expect to see here, rather than kind of a broader issue of just a COL information item that specifies developed procedures that conform to the Reg Guide.

MR. COLACCINO: Yes, this is Joe Colaccino and I just want to reiterate what Getachew said. This is something that we have tried to do in working with the applicant for now several years into having the COL information items, which they are ultimately, the applicants would be required to address. This is kind of like a road map of what is left to do in the COLA. So, this is completely consistent with that.

MEMBER STETKAR: Yes, I understand that. And I support that as far as hooks into the certified design. As I said, I was hanging up on the rather detailed wording of these open items. So, thanks.

MR. LI: There is a last open item which is associated with ITAAC. The EPR design, those have ITAAC associated with reactor coolant leakage detection system spreading out in different sections. But we are looking for the key parameters that is the sensitivities, response times, and alarm limits in the ITAAC. That concludes my presentation.

CHAIR POWERS: In your judgment now, sir --

MR. TESFAYE: We can squeeze in one --

CHAIR POWERS: We can squeeze in one more?

MR. TESFAYE: -- more before we go on break. Joe Jenkins?

MR. JENKINS: Yes, my name is Joel Jenkins. I am on staff with -- excuse me. I am a materials engineer on staff at the component integrity branch. My educational background is I have a Bachelor's degree in materials engineering from Virginia Tech.

CHAIR POWERS: I knew that.

(Laughter.)

MR. JENKINS: I have been on staff with component integrity branch since 2008. Prior to that I worked for 20 years in the Navy nuclear shipbuilding industry, where I worked in the metallurgy lab and also in engineering departments responsible for the reactor coolant pressure boundary. And I will be speaking today on reactor vessel materials, Section 5.3.1.

Now the EPR reactor vessel is designed and constructed with ASME Code Section III, as required by 10 C.F.R. 50.55a and, therefore, meets NRC requirements. Also the materials meet fracture toughness requirements of 10 C.F.R. 50 Appendix G.

Now the reactor vessel is constructed from ASME SA-508 Grade 3 Class 1. The inside of the vessel is clad with stainless steel or nickel alloy. To elaborate on that bullet, stainless steel 308L or 309L is used for the majority of the cladding. Nickel alloy is only used at locations where nickel components are welded to the reactor vessel, such as the Alloy 690 radial keys.

Not mentioned on the slide but a major point worth mentioning, in fact re-mentioning because it has been previously mentioned several times is that Alloy 600 is not used in the design. Neither is Alloy 82 or 182.

In our review of the standard design, the staff requested additional information, including clarification of the minimum thickness of the nickel alloy cladding, qualified as buttering, to clarify that bullet point, and also to confirm that low-heat-input weld processes are used for attachment welds that are not subsequently post-weld heat treated.

Now I want to go back to that question 14 and elaborate on that because the design document does specify the thickness of the cladding. In fact is it specified as 0.295 inches and we requested that AREVA clarify whether that applied to stainless steel cladding or the nickel alloy cladding. They have clarified that that applies to both types of cladding.

AREVA had stated that the nickel alloy cladding could be qualified as buttering. So we wanted a clarification to make sure that we knew that the minimum thicknesses apply to that as well. And as a matter of fact, we did get, we already had received a verbal clarification with a promise for official RAI response. So question 14 open item could be resolved very quickly.

Moving on to question 15 to elaborate on that, the attachment weld I am referring to is the radial key, which is Alloy 690, which is welded to the reactor vessel cladding.

Now I would like to move on to the reactor vessel surveillance program, abbreviated as RVSP. The RVSP complies with 10 C.F.R. 50 Appendix H and ASTME 185, revision 82. And this surveillance program meets NRC requirements.

The AREVA design uses four capsules, each containing materials used for various mechanical testings like sharpies, tensiles, compact tensions. The withdrawal schedule for those specimens does comply with ASTME 185-82.

The four capsules are attached to the core barrel by means of guide baskets. There are four capsules, two guide baskets, two capsules per guide baskets. The guide baskets are bolted to the core barrel and staff requested additional information on operating experience and integrity issues with the bolted connections in this application.

The staff has received an RAI response from AREVA in which they stated they will provide a response to this question by March 23rd.

MEMBER SHACK: Just, I mean the NRC requirements, I mean, this vessel has a 60 year design life. I don't think it would be unreasonable to expect that they would be operated for 80 to 100 years. Does the surveillance requirement go out to that potential end of life?

MR. JENKINS: It goes to 60 years.

MEMBER SHACK: I think it is probably time to maybe rethink the requirement. But it is not a design cert issue. But I mean, the baskets will be such that you really couldn't get more specimens in there. Right? I mean, that will be fixed at the design certification stage.

MR. RAY: This is Neil Ray. I am NRO. Briefly my background is I have a BS and MS in mechanical engineering with another Master's in engineering management. I worked at Westinghouse Idaho National Lab U.S. Energy Department, now finally here.

In terms of your questions, the answer is, although they didn't say it but we reviewed it, the capsules are enough there to go all the way to 80 years, 100 years.

MR. JENKINS: I am done with my presentation, unless anybody has any further questions.

CHAIR POWERS: Okay, we are going to interrupt. We are going to take a break until 3:17.

(Whereupon, the foregoing proceeding went off the record at 3:00 p.m. and went back on the record at 3:17 p.m.)

CHAIR POWERS: Okay, we are coming back into session. We are going to complete the staff's presentation and then we are going to chat just a little bit about the festivities for April and how we are going to make an interesting and stimulating discussion for the full committee meeting, not go over time and stay within schedule and everything else like that. Right? Okay.

MR. ROY: We have Mr. Steve Downey representing 5.3.2 and 5.3.3.

CHAIR POWERS: And this is more stuff bringing tears to Dr. Shack's eyes.

MR. DOWNEY: Good afternoon. I am Steve Downey. I am a materials engineer on staff with the Component Integrity Branch. I received my Ph.D. in mechanical engineering with a concentration in mechanics and materials in 2008. And after a six month vacation from all things engineering, I came to work for the NRC. So, I have been here for a while.

CHAIR POWERS: Where did you do this famous degree?

MR. DOWNEY: Oh, Florida A&M, Tallahassee.

CHAIR POWERS: Florida A&M. All right, no Virginia guys here.

MEMBER STETKAR: You went south of the Mason-Dixon line. You went to school in Florida and then you had to take a vacation?

(Laughter.)

MR. DOWNEY: It was a recovery.

MEMBER STETKAR: A recovery. Okay.

MR. DOWNEY: I am the technical reviewer for PT limits, upper-shelf energy, and pressurized thermal shock for the different design centers. And we are reviewing this section to address the requirements of 10 C.F.R. Part 50 Appendix G, fracture toughness. And first the only open item in this section is related to pressure temperature limits.

But first let me say that the upper-shelf energy, based on the staff's review is in compliance with 10 C.F.R. Part 50 Appendix G. These values were projected out to 60 years, although the licensing is for 40 years. The same with the pressurized thermal shock values.

For the three weld regions, the three welds within the beltline region, the highest value was 141.1 degrees F. and the criteria is 300 degrees Fahrenheit. And for the base material in the beltline region of the reactor vessel, the highest value was 70.3 degrees Fahrenheit, which is way below the criteria of 270 degrees Fahrenheit.

CHAIR POWERS: And when you say it is 70.3, what is your uncertainty on that?

MR. DOWNEY: The 70.3 is based on the material properties. So the uncertainty, I don't know what it is but that may be a question you want to ask the applicant.

CHAIR POWERS: What a great position. Every question that comes up, you say, oh, the applicant can answer that question.

MR. RAY: What it the question again?

MR. DOWNEY: What is the uncertainty on the pressurized thermal shock values?

MR. RAY: Well, right at this moment we have to all understand the EPR has not been manufactured yet.

So the values that are provided are all bounding properties only and that is based on currently copper, nickel, with uncertainty values provided by the applicant as well.

So using 10 C.F.R. 50.61, this is the value of what Steven mentioned. So there is no uncertainty but not expecting anything at this point. We will get the part of qualification and the final numbers as and when the facility is manufactured and is due to be delivered.

CHAIR POWERS: So you are telling me that 70.3 is an upper bound.

MR. DOWNEY: Yes, based on the equations in 10 C.F.R. Part 50.61.

CHAIR POWERS: I understand.

MR. DOWNEY: Fluence attenuated to the wettest surface, etcetera.

CHAIR POWERS: Okay and if a copper atom happens to sneak into this vessel we will shoot it.

MR. DOWNEY: These are bounding material properties, as stated, and there is a COL item to where the COL applicant will provide plant-specific values. So if those values are different, then --

CHAIR POWERS: We will recalculate.

MR. DOWNEY: Yes. Now on to pressure temperature limits. And the applicant has decided to address the requirements of 10 C.F.R. Part 50 Appendix G, related to pressure temperature limits by submitting a pressure and temperature limit report, which is generic to the U.S. EPR design.

This report follows the guidelines of Generic Letter 96-03, which provides seven technical criteria to be addressed in order to submit the pressure temperature limits and the complete methodology for their development. In addition to providing this pressure and temperature limits report, the applicant also provided a COL information item, which states that the COL applicant that references the U.S. EPR design will provide plant-specific pressure and temperature limits using an approved methodology.

Now all of the open items in this section are related to the review and approval of this U.S. EPR generic pressure and temperature limit report. And that report has been received and it is currently under staff review and we have issued several RAIs, four of which are shown on this presentation, among others that have already been resolved. And these RAIs, we have had constant interaction with the applicant and these issues have also been addressed and will be formally submitted by, I think the date is April 9th is what the communication says.

Any questions?

CHAIR POWERS: I don't see any. You have just left the committee dumbfounded here. Thank you very much, Steve. That was good.

MR. ROY: Next will be John Honcharik and it is going to be for 5.4.1.1.

MR. HONCHARIK: My name is John Honcharik. I am a materials engineer and I previously did my bio this morning.

I will talk about the reactor coolant pump flywheel. The integrity of the flywheel is important in preventing the generation of high energy missiles that may affect safety-related equipment.

The U.S. EPR FSAR section describes the materials used, along with the fabrication and inspection of flywheel to ensure its integrity following the guidance in Reg Guide 1.14. The materials used, as you heard previously is a quenched and tempered alloy steel, based on ASME code.

For the pre-service inspection, the FSAR states that this inspection includes 100 percent volumetric inspection, a surface inspection, and dimensional inspection. In addition, the flywheel will be spin-tested at a design overspeed of over 1500 rpm. The design overspeed is 125 percent of the normal operating speed.

Related to this section there are two open items. The first open item is that an ITAAC should be included in the FSAR to ensure that a spin test after design overspeed is performed. This assures that the flywheel assembly can withstand the design overspeed event and preclude the generation of missiles as required by GDC 4.

And the second open item deals with the timing of these pre-service inspections as they relate to the spin test. Reg Guide 1.14 guidance specifies that nondestructive inspections be performed after the spin test. This will ensure that any flaws that have initiated or grown during the spin test or any dimensional changes can be detected prior to being placed into service. Therefore, this is an open item.

And as I note, for a status, the staff recently received from RAI responses this week and we are currently reviewing them but we are look optimistic that these two issues can be resolved rather quickly.

Next, AREVA provided a reactor coolant pump flywheel analysis in their AREVA Report, in accordance with the Reg Guide 1.14 and SRP Section 5.4.1.1, in order to meet the requirements of GDC 4 in preventing the generation of missiles. This report evaluated the critical speeds for various failure modes, which included ductile fracture, non-ductile fracture, and excessive deformation.

Based on the reactor coolant pump flywheel analysis, the critical speed due to excessive deformation of the flywheel is based on the circular collar and the thrust runner. The material properties of the thrust runner were used in this analysis. However, to ensure that the analysis in the AREVA report bounds the material that would be use for the thrust runner in the actual reactor coolant pump, the material specifications should be included in the FSAR. And therefore, the staff has identified this as an open item.

In addition, we also received this week a response to that and there is a PAP4 enclosed in that open item, too. And those were the issues related to the reactor coolant pump flywheel.

CHAIR POWERS: Thank you.

MR. ROY: Next is 5.4.2.1 and 5.4.2.2. Greg Makar is going to present that.

MR. MAKAR: Thanks. Hi, I am Greg Makar. I am a materials engineer in the Division of Engineering, Component Integrity Performance and Testing Branch 1. And I have a Bachelor's degree in mining and mineral engineering from Virginia Tech.

(Laughter.)

MR. MAKAR: I have a Master's and Ph.D. in materials science and engineering from Johns Hopkins University. I have been at the NRC for just under seven years and the last three in my current position.

Before coming to the Agency, I was a senior research engineer with a company that makes paper and chemicals and I worked on a corrosion and materials engineering team that provided support to the production facilities there. And before that, I was at the Knolls Atomic Power Laboratory working in plant corrosion technology.

So I will cover two areas in steam generator and the first is steam generator materials. This is SRP Section 5.4.2.1. And first I will say that we found that the materials and design features meet the relevant requirements, GDC 1, 14, 15, 30, 31, 10 C.F.R. 50.55a is related to the standards of quality for the reactor coolant. Pressure boundary and safety related materials and the design reactor coolant pressure boundary integrity I should say, and they are met, for example, by using materials that conform with the ASME code and designing them to the code and using materials that are compatible with both coolant environments.

Principle materials here are carbon steel, stainless steel, and nickel based Alloy 690 and its weld metal equivalents.

Now these steam generators are typical of the recirculating or U-tube steam generators that are used as replacements at U.S. plants. And I mean that in terms of the principle design features, the ones that are discussed in our SRP section. The tubes are of typical diameter. The wall thickness of the heat treatment is the same and the state and the spacing as well. And no corrosion-related degradation has been detected in those materials which were used in the first replacements in 1989.

The martensitic stainless steel, which is used for tube support plates and other support locations is, as you have heard earlier, it is a corrosion-resistant material. It also has, it is made with non-circular holes which promote flow along the tubes and contact the tubes over a small area. And that material is also a good thermal match to the tube material in the shell. And in addition to promoting the flow of deposits, other things that are already in there, it is also, because it is corrosion resistant, it doesn't itself corrode and generate products that then effect the tube. Of course, denting was --

CHAIR POWERS: What is the electrical chemical potential difference between the support plate and the tube?

MR. MAKAR: I don't know in the number but it is certainly more oxidized, a higher, I won't say certainly because I just said I don't know.

So I don't know.

CHAIR POWERS: Similar to the difference between that and carbon steel.

MR. MAKAR: You know, the operating experience has shown that there is no measurable corrosion of that material.

CHAIR POWERS: It seems like such an elementary measurement that you would want to know when designing this thing. I am surprised.

MR. MAKAR: You mean from galvanic corrosion because there is contact?

CHAIR POWERS: Yes.

MR. MAKAR: Well I don't want to imply that I have looked at this for this review but in materials that I have looked at, published materials that I have looked at, the difference in open circuit, the corrosion potential is very small. You know, probably something you would have difficulty measuring in a reproducible way.

CHAIR POWERS: Yes, you kind of wish the guys that just selected carbon steel for certain generators had actually thought about that one.

MR. MAKAR: It would be a good think to look at if we ever need to pull an Alloy 690 tube to see if there is any evidence of corrosion at that contact point.

Okay, so now there is also a -- those tubes are installed into the tube support, sorry, into the tubesheets with full thickness of the tubesheet, with hydraulic expansion through the full thickness of the tubesheet. It is a method that has shown to generate the lowest, a low level of residual stress at the transition area.

CHAIR POWERS: Yes, how do they -- why does it work at the transition area? One comes right out of the support plate where it emerges, it seem to me like it still would be like a high-temperature or high-stress.

MR. MAKAR: Well it is elevated, I am sure and it is probably dependent on the shape. You know, if we are given tube size, depending on the shape of the transition and I think this method gives the kind of shape, you know, a gradual enough transition that there is no, that the highest rest area is lower than the highest for other places. That is my understanding of why.

It is designed for inspection and cleaning. There is a blowdown system, designed to sweep materials to the blowdown area and also there are inspection ports at various elevations that allow you to get to the tube bundle as well as the fluence at the top of the steam generator.

Now all of these and the water chemistry, they are following EPRI water chemistry guidelines for the primary and secondary water. All of those things, all those design features and the water chemistry have been responsible for dramatically reducing the number of problems with steam generator tubes over the recent decades.

There is one COL item, which is for the applicants to identify the edition and addenda of the ASME code section 11 that will be applied to the steam generators.

Now although all of these features have been proven effective over the years, we are still taking the approach that we expect that problems will be found sometime and inspectability and the inspection program is important. And in fact, we have some RAIs related to that kind of thinking on the steam generator program. Next slide please.

And the steam generator programs just refers to the combination of inspection, assessment, monitoring, and repair that is meant to ensure the structural and leakage integrity of the tubing. And we found that the applicant's steam generator program meets the relevant NRC requirements, including GDC 32 and again 50.55a, as well as 50.36 and 50.65 related to the reactor coolant pressure boundary and maintenance and technical specifications.

And so these requirements are meant in part by designing for inspection, in plain English, we already talked about, and describing a program based on nuclear energy institute 97-06 and the standard technical specifications.

Now the program, the NEI 97-06 incorporates the detection of defects, leakage integrity or detection of leakage, water chemistry guidelines, assessment of any degradation and other things. The details of those are in EPRI documents that are referenced in the NEI 97-06.

And so this includes an operational assessment, which is looking forward to see if the tubes will make it to the next, through the next operating period. If there is degradation found for the U.S. EPR, this conforms to the guidance in Reg Guide 1.121 as far as how much through-wall degradation is allowed. The number is 40 percent and it is what most plants use and it is in the standard technical specifications.

And the tube integrity is required in the technical specifications. EPR's tech specs in this area conform to the standard tech specs, except there is one area where the current standard tech specs, we don't want them to use that. They don't apply. And that is, the standard tech specs don't address the initial inspection of a newly-installed steam generator. And so we do have an RAI about the wording of that, to make sure it applies to both newly installed and replacement steam generators. And we received a response recently. So my expectation is this open item will soon be turned into a confirmatory item but we are reviewing that response now.

CHAIR POWERS: It seems to me that the history of steam generator team inspections is a history marked by oh, we found this mysterious indication. We don't think it is a flaw but we can't find the records on the initial inspection to be sure it wasn't there from day one and things like that. We are not going to get into that trouble is what you are telling me.

MR. MAKAR: Well, we have better methods of documenting that, performing and documenting that initial inspection. So now it is not addressed well in the tech specs, --

CHAIR POWERS: I agree with you on that.

MR. MAKAR: -- the pre-service inspection. We do have it, they have it, we made sure they have it in the DCD. And as a COL item, we have it, it is not itself a COL item, we ask COL applicants to add that reference to that section, in this case 5.4.2.5 in the table of operational programs. So it is specifically mentioned there to highlight that under inservice inspection as well as pre-service inspection.

CHAIR POWERS: Now one of the problems that we run into is in this era of a chronically recording information that the ability to read media from one decade to the next falls fairly dramatically. Do we address that anyplace?

MR. MAKAR: I don't know that we do. I think it is possible, if you had an old form of media. You probably have two versions of it. There is probably a book of printed information somewhere and it is in some electronic form. And it may be possible, if you go to the next generation and there is an incompatibility that you could go back and compare something in a manual way. But I think it would be -- it would probably plug the tube.

But if there was a bigger, if it was a major issue that wasn't so simple, then I really don't how that would be addressed.

CHAIR POWERS: I mean, it is one that is plaguing us across the board. I can think of a hundred incidences where people were very diligent, they recorded information very carefully. There are, like you say at the time, they set up a printed document with the information and they set up electronic media. The printed document got lost in the -- the principal investigator retired, the electronic media now is on a Bernoulli device and nobody has the capability to read a Bernoulli anymore.

It is just an interesting point. Please

continue.

MR. MAKAR: Well, that finishes my part. There is that one open item.

CHAIR POWERS: Thank you.

MR. ROY: Now we have John Budzynski and Shanlai Lu for 5.4.7, 5.4.11, and 5.4.12.

MR. BUDZYNSKI: Section 5.4.7, RHR System. This question has really been under review in Section 6.3 but the reason why I put it here was to highlight the fact that the low head safety inspection pumps are common to both the RHR system and the SIS system. And just to say that although we feel that the RHR system is acceptable, we still have to resolve this one issue.

Next slide, please. Okay, this is 5.4.11, pressurized relief tank. No problems with that at all.

MEMBER STETKAR: Before you jump on the pressurized relief valve, I just thought of something. It is more just information. On this plant, if I want to initiate RHR cooling, is that a fully automatic process? I am aware of some other European designs where you basically walk up to the control board, push a button and everything is done automatically. It is a pre-programmed evolution, including cooldown rates and things like that. Is this beast designed that way?

MR. PARECE: This is Marty Parece. To answer that question, the answer it is a mixed bag.

MEMBER STETKAR: So when you say it is a mixed bag, it is --

MR. PARECE: What I am saying is when you get the conditions sufficient to start RHR cooling, when the operator starts the trains, but he can -- now again, all of this hasn't been totally designed yet but the intention is that the operator can set the cool down rate and then the control system will cool the plant and it will do that by adjusting the bypass flow around the RHR heat exchangers.

MEMBER STETKAR: But I mean, that is basically the way the thing is designed. The thing is designed to the operator does not actively control --

MR. PARECE: Does not have to actively control the bypass flow and the flow through the different systems. And the thing --

MEMBER STETKAR: Does it select the number of trains that it needs to start up?

MR. PARECE: Generally not. No. The operator selects those trains. And so generally, under normal conditions we will start up the trains one and four at 250 degrees Fahrenheit. And then when we get below 212 degrees Fahrenheit, he can start up trains three and four. And it is all about maintaining the cool down rate because we are trying to get into the outage in an expedited time.

MEMBER STETKAR: So he selects the trains and from there on out --

MR. PARECE: Right.

MEMBER STETKAR: Okay. Thank you.

MR. PARECE: And we select trains two and three, one and four first and then two and three below 212 because trains two and three are in buildings two and three and the control room is in building two.

MEMBER STETKAR: Okay.

MR. PARECE: So in case there is a leak or other problem, then you don't humidity and energy problems in the building where the operators reside.

MEMBER STETKAR: Trains three and four have the letdown connections to them, though. Right? If I remember.

MR. PARECE: Each train has a letdown connection from that loop hot leg.

MEMBER STETKAR: No, no, no. I mean cleanup.

MR. PARECE: The cleanup, I don't remember the details.

MEMBER STETKAR: I think it is three and four, if I remember it correctly. That's all right. That is a minor point.

MR. BUDZYNSKI: All right. Yes, the next slide, please.

Section 5.4.11, pressurizer relief tank. We found no problems there and it is acceptable. Any questions on that?

MEMBER STETKAR: John, one thing that this again is just information. I read in the FSAR and I read in your SER that it is carefully worded that the discharge from the rupture discs are directed into the reactor coolant pump cubicles in a geometry that prevents damage, or I don't know whether it says damage or impact to any safety-related equipment.

What safety-related equipment is located inside those cubicles?

MS. SLOAN: Anyone from AREVA who can respond?

MEMBER STETKAR: I mean, it is carefully worded and it is reproduced in the SER that the discharge is directed into the cubicle such that it doesn't impact any safety-related equipment.

MR. PARECE: Well, I was just at OL3 and

standing down there where those components go. And it seems to me that we have got to have the normal stuff you would expect like reactor coolant RTDs with signal processing coming out analog and eventually go to an analog to digital converter. And so you wouldn't want a jet, a vapor, or anything.

MEMBER STETKAR: Sure but I mean I think it is kind of proximity to the loops that --

MR. PARECE: So it is just a general --

MEMBER STETKAR: Okay, thanks.

MR. BUDZYNSKI: Any other questions? Next slide please.

This is Section 5.4.12 reactor coolant system highpoint vents. There is one open item, RAI 342. We have received a response from AREVA about two or three days ago and it is under review. And basically what we wanted to confirm that the CVS system can provide adequate makeup if the vents fail open.

And also CVCS is not a safety rated system and is not required to supply reactor coolant makeup to the RCS n the event of small breaks or leaks in the RCPB. And we just want to verify that the system, how they make GDC 33 and GDC 35.

And like I said, this is under review and we should know in a couple of days.

Any other questions?

MR. LU: We got the responses and we are in the process of review so when we prepare the slides at that time, we don't know. That is a logical -- at this point we don't see a problem but we will find out.

MR. BUDZYNSKI: No, we don't see a problem.

MR. LU: Conclusion?

MR. BUDZYNSKI: And the last slide is conclusion.

Except for the open item discussed above, the staff concludes that the design of the RCS highpoint vents is acceptable and satisfies the guidance of the SRP 5.4.12.

MR. LU: So yes, overall, this part is simple. It is straight forward. And then once the open item and the RAI responses are documented, then we are done.

CHAIR POWERS: They will be cheering.

MR. LU: -- related to this item.

CHAIR POWERS: That was very nice. Very nice. Very nice, indeed. Thank you all.

MR. TESFAYE: Any question on the

acronyms?

CHAIR POWERS: On the acronyms? I think you can give lessons to others.

MR. TESFAYE: Thank you.

CHAIR POWERS: Well let's chat a little bit about festivities in April. The strategy, I think that the world wants us to follow, I believe, the world defined as the Commission and the staff and the applicant, is to get kind of a status report on where we stand in this certification review about roughly half-way through the chapter list. And the requirement is, of course, that we make a little bit of an oral presentation in front of the full committee on what we have done so far.

Okay now, a little caution on this of course is as much of the rest of the committee is not paying hour-to-hour attention to this particular certification. So it strikes -- I know it is stunning.

I mean, the fact of the matter is, I will let you in on a dirty little secret, this one is going much more smoothly than the others are going.

It strikes me that the applicant will need to remind what the EPR design is. Okay? And I think that will constitute much. We will have about an hour and a half. Right?

MEMBER STETKAR: You can have as much time as you want.

CHAIR POWERS: No we can't have as much time as we want.

MEMBER STETKAR: Seven minutes?

CHAIR POWERS: I mean, the most that they will give us and I have not spoken to the planning procedures committee but the most I will give for any one topic is two hours. And so and that is for both the staff and the applicant and all the time the ACRS members will take asking questions. So that constrains us enormously.

And quite frankly, the objective is not to go through a chapter-and-verse blow of everything that has gone on the way we do in the subcommittee meeting. It is rather to give them some kind of a status of where we stand.

So it strikes me that the applicant's first objective and representation is in fact is to remind the full committee of what the EPR design is. And the corresponding obligation of the staff I don't think the committee needs to be reminded of what the certification review process is. I think they are up to speed on that but you might want to go through what the strategy is when it is applied to EPR because it is not a passive plant.

So it is a little more classic strategy here, recognizing that when I say classic strategy, no one has gone through the classic strategy in 20 years. So you might want to remind them and emphasize those things where you feel like you need to do an independent evaluation. Those things where you felt like you could just review the applicant's submission, those things that required the staff to do audit at the offices and things like that, it doesn't have to be everything. It can be kind of in a broad-brush kind of approach. I mean, I would not plunge into the details on this thing. And I would confine my specifics to those chapters we have gone through and I would not include Chapter 19 there because we incomplete as yet.

MEMBER STETKAR: I think what I would add too from sort of general committee is I think the committee understands what a four-loop pressurized water reactor is. So if you are limited in time, emphasize, kind of remind the members of the differences. What is different about this thing compared to a standard four-loop Westinghouse plant, let's say.

And from the perspective of both, and I think this is important from the perspective of both AREVA and the staff is to make note of differences in similarities in the design certification and review process related to items that are kind of in gray areas. And what I am thinking about is, on the passive plants we talk about RTNSS. Well you don't have RTNSS when you have a DRAP program that is something sort of kind of maybe a little bit like that.

So making sure that this is both from the staff's perspective and from the applicant's perspective that the committee members understand a little bit about how those might be related or how they might not be related to each other at all. And the other issue, obviously, is the general issue of DAC and ITAAC. You know --

CHAIR POWERS: Do we really want to go into DAC and ITAAC at this stage?

MEMBER STETKAR: I don't know.

CHAIR POWERS: I don't think so.

MEMBER STETKAR: We haven't heard much about DAC and ITAAC so far at least on these chapters. So perhaps not.

CHAIR POWERS: I think it is a little

early to plunge into that. I think I would stay more at the strategic level and as far as we have gone and whatnot. Trust me, I am going to look at the ESBWR guys and -- I mean, it looks smoother there.

But yes, let's leave those details --

MEMBER STETKAR: Okay.

CHAIR POWERS: -- to them. I mean, we are going to get another shot in front of the full committee.

MEMBER STETKAR: Eventually we are going to get to things like Chapter 7.

CHAIR POWERS: We will get into those things but I don't want to go now. I want to stay fairly clean, fairly high level and fairly summary in nature.

MR. TESFAYE: Are you going to include the Chapter 11 and Chapter 16 that will be presented on April 6?

CHAIR POWERS: Since I haven't seen those yet, I will leave that to your discretion. We certainly can if you want to. You know, I just don't know how -- until we have had the subcommittee, I can't really advise you on that. But they are fair game for your inclusion if you want to.

MEMBER SHACK: Dana, there is one other

point here that is different. This is the first reactor design certification. They are not going to get the PRA. And I don't know whether we want to have the staff maybe talk about at least their review approach without getting into the -- I mean, the trouble is, if we bring it up we get into the gory details but it might be helpful to at least outline how that is going to be done.

CHAIR POWERS: It would be useful to go through the strategy. I mean, strategically --

MR. TESFAYE: I thought Chapter 19 was not going to be part of that.

MEMBER SHACK: Well, I am discussing that with the Chair at the moment.

CHAIR POWERS: You are disagreeing on that. I think I agree with him, the strategic elements of it, I don't think we need to go into the details because I think we are still working the details there. But the strategic elements on what you tried to do and your PRA and how you tried to review it are just, I mean I think those are fine. I don't have any troubles with you but I just would not plunge into the details.

MS. SLOAN: I think just to be clear on PRA, I don't think we did anything different from our perspective, except for what we submitted. We did the same PRA everyone did but because of the rule change, we submitted what we did. So from our perspective, the strategy is very conventional.

CHAIR POWERS: And please feel free to make that point that you are open to looking at any part of it.

MEMBER STETKAR: In the sense of generic discussion of the PRA, is there any merit in having AREVA outline how they have used the PRA, rather than getting into too much detail?

CHAIR POWERS: You are going to plunge into the details.

MEMBER STETKAR: Okay. I think you are probably right. But I just thought I would ask it since you saw it.

CHAIR POWERS: Quite frankly, anytime we are going to plunge into the details, then I am going to set up a meeting in front of the ACRS just on that topic because you are going to run out of time really quickly. I mean, you run out of time almost instantaneously. You would be surprised how short two hours is, especially when you have, you know, there are going to be members that have been totally focused on another certification. You are going to be seeing this kind of -- I mean they have seen the design. They know what the design is but they have forgotten most of it. And so they are going to have a lot of questions and things like that probably coming from a different context.

And so you know, you are going to have to align them with the fact that this is this reactor and those others are the other reactors. And so you would be surprised how quickly. You know, we tell you you get 50 percent of the time to present and allow 50 percent for questions. That is guidelines to you. Unfortunately, we don't give the members that guideline and they are perfectly willing to take 90 percent of the time asking questions.

MS. SLOAN: What I am hearing then is a very brief design overview focusing on unique features, methods, strategies, and then maybe on a chapter-by-chapter basis.

At that level, just simply talk about here was our strategy for Chapter 2. Chapter 2 is mostly COL items, a lot of site specific stuff. Strategy for Chapter 8, pretty conventional stuff and just at that level, Chapter 10 conventional with a couple of exceptions there.

CHAIR POWERS: And for your presentation

you are going to have to go a little farther into the details. Certainly you are going to want to say okay and, you know, here is our strategy to review. We have got X open items here with the two of them we think are significant. The rest of them, we are confident will be resolved. You know, they are just clarifications and that's it.

But constrain yourself because you are going to be the one that is going to run out of time. You are the one that we are going to say let's move on, move on, move on, because --

MEMBER SHACK: Don't go first.

CHAIR POWERS: Yes, Sandra will go first and she will eat up half the time, more than have the time. But I would emphasize the strategic.

Although the committee is very interested in where you thought it was important to do an independent analysis and where you thought it was adequate to review the licensee's submission. And all of this interest. I don't think they have hard and fast views on what should be done. It is just they want to know.

And you have been very good in both your SER and in your presentations in making it very clear exactly what the staff has done. In fact, I congratulate you for that. That is usually the biggest problem that we have is understanding exactly what the staff did in doing the review and you have been very clear here.

And for making that apparent and the strategy, for both of you I would minimize the number of speakers. You can use other people but one is the minimum.

(Laughter.)

CHAIR POWERS: You know, you can use your discretion. Two is okay but I would not have -- and you are going to have other people sitting. It will be much like it is here. You know, you are going to have your support staff there to help you answer the question up at the front of the table. I would minimize the use of people on the perimeters, okay, just because of the mechanics of the meeting and things like that.

And yes, I will give some sort of an introduction. And quite frankly, I am going to sing your praises because I think you have done very, very, very, very well here. I think this has been exemplary.

MEMBER RYAN: Dana, I think for the past briefing and this one, some of the figures and photographs, particularly, integrating those into your presentation is very helpful because --

CHAIR POWERS: Very helpful.

MEMBER RYAN: Yes. Particularly the real photographs of arrangements of parts and pieces really helped, I think, communicate a whole lot very quickly. So that helps. And then some of those schematic diagrams, for example, in the radiation protection chapter, you had that detailed discussion was very helpful.

There is one example that I wanted --

CHAIR POWERS: PIDs are not helpful.

MEMBER RYAN: Yes, PIDs are not helpful but the pictorial representations or arrangements I think are often very helpful.

CHAIR POWERS: Very helpful.

MEMBER RYAN: You might want to have a separate packet of that or integrate them into your presentation, whatever works the best.

MS. SLOAN: Very good idea.

MEMBER RYAN: Thank you.

CHAIR POWERS: And what the committee does is we will present them a proposed letter which is mostly a status report in which we will go through and say things about where we stand on reviewing the chapters, where we think that our sticking points are. And there are not going to be too many of those. I think there will probably be some advice going up to the Commission on the ground rules for doing certifications and things like that. We have seen things that have not been helpful to either the licensee or the staff. And that is about all I can tell you right now.

MR. TESFAYE: I have heard an hour and a half and I have heard also two hours.

CHAIR POWERS: Yes. One of the two because I haven't talked to the P and P to know. I will try to get two hours but I know two hours is their limit for any one topic.

MS. SLOAN: So two hours for staff, AREVA, and you folks?

CHAIR POWERS: Yes, everything but the upper bound is two hours. It depends a little bit on what else is on the agenda.

MEMBER STETKAR: This is for April?

CHAIR POWERS: Yes.

MEMBER STETKAR: Right now April is looking a little light. So we might be able to squeeze out.

CHAIR POWERS: That would be a bit more than two hours. I mean, I have tried in the past and it is --

MR. WIDMAYER: You are covering a large number of chapters, though. They might be willing to --

MEMBER STETKAR: I mean the April meeting right now is shaping up to be a little bit light. So if you --

CHAIR POWERS: Guys, the one thing you don't want is more time. You should cut it down to an hour and a half.

Okay, my best advice and as you think about putting your presentation together, feel free to contact Derek and if you have got questions or things like that, to get more advice. You know, you and I talk and if it is other members that are in the lead, you will talk to them.

MS. SLOAN: We will do that.

CHAIR POWERS: And we will try to make this as painless an exercise as it possibly can be and whatnot.

We will have done, we will have another meeting just before the full committee meeting. So I think we will certainly have an opportunity to chat and whatnot like that. But I mean, I think it is a prudent thing to give kind of a status report part way through. I have not felt the need to report on every single chapter. We are just not running into the kinds of controversies that some are running into. So we are probably going to give them an update. I think it is going along swimmingly.

MS. SLOAN: Good to hear.

CHAIR POWERS: Okay, any other comments? Any advice?

MEMBER RYAN: None.

CHAIR POWERS: That is the best we can do for you. I know it is not much but we can try. I mean, we certainly will make ourselves available to advise further as you get into the details of putting it together.

But I think, I mean, Mike is absolutely correct. The more visual you can make things, the better off you are. And the more strategic you can make things, the better off you are. You know, you can pepper them with a little detail to wet the appetite but you want to stay at a substantially higher level than you are here.

With that, I am prepared to adjourn this meeting, unless there are any other comments to make.

MEMBER RYAN: Second.

CHAIR POWERS: We are adjourned.

(Whereupon, at 4:09 p.m., the foregoing proceeding was adjourned.)

Presentation to ACRS
U.S. EPR Subcommittee
Design Certification Application
FSAR Tier 2 Chapter 4

Jeff Tucker, Advisory Engineer
Jonathan Witter, Ph.D. Advisory Engineer
AREVA NP
March 3, 2010

FSAR Chapter 4 Reactor: Section Topics

- **Section 4.1 – Summary Description**
- **Section 4.2 – Fuel System Design**
- **Section 4.3 – Nuclear Design**
- **Section 4.4 – Thermal-Hydraulic Design**
- **Section 4.5 – Reactor Materials**
- **Section 4.6 – Functional Design of Reactivity Control Systems**

Chapter 4, Reactor:
4.1 Summary Description

- **U.S. EPR design features and processes are fundamentally the same as used for previous designs**
  - **Fuel rods in a 17x17 lattice**
  - **High Thermal Performance (HTP) intermediate spacer grids**
  - **High Mechanical Performance (HMP) end spacer grids**
  - **M5 alloy cladding material**
  - **Use of UO_2 fuel with rod arrangements of UO_2 and UO_2:Gd_2O_3 loading zones**
  - **Design methods and codes for mechanical, nuclear, and thermal hydraulic designs approved for use in ANP-10263P(A)**
  - **Standard fuel management strategies used for power distribution and burnup control**

Chapter 4, Reactor: 4.1 Summary Description

- **U.S. EPR design features that are different:**
  - **14 foot active fuel length and 241 assemblies**
  - **Rod control assembly with full span annular Ag-In-Cd absorber rods**
  - **265 fuel rods, fuel rod replaces instrument tube center lattice location**
  - **Incore instrumentation used for protection system and reactor control**
    - Aeroball measurement system used for calibration of core monitoring neutronics computer codes and the incore fixed Co-59 self-powered neutron detectors (SPND)
    - 12 strings of SPNDs at fixed radial positions, each with 6 SPNDs at fixed axial elevations
    - Online monitoring of DNB and LHGR through power reconstruction from SPNDs
  - **Topical Report ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR" to implement incore power measurements**
  - **Stainless steel "heavy" radial neutron reflector**

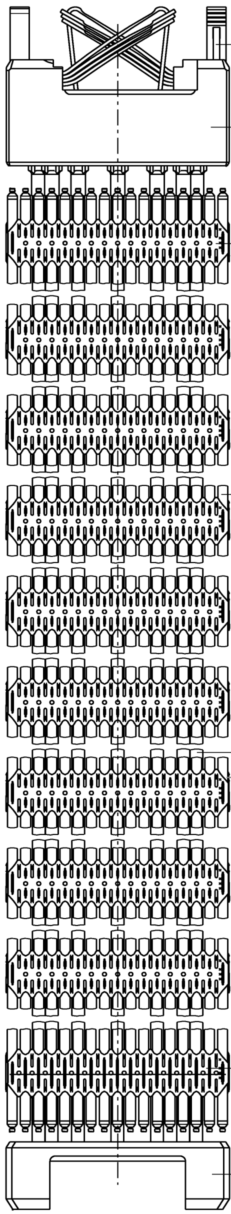
Chapter 4, Reactor:
4.1 Summary Description
Fuel Assembly Design Comparison

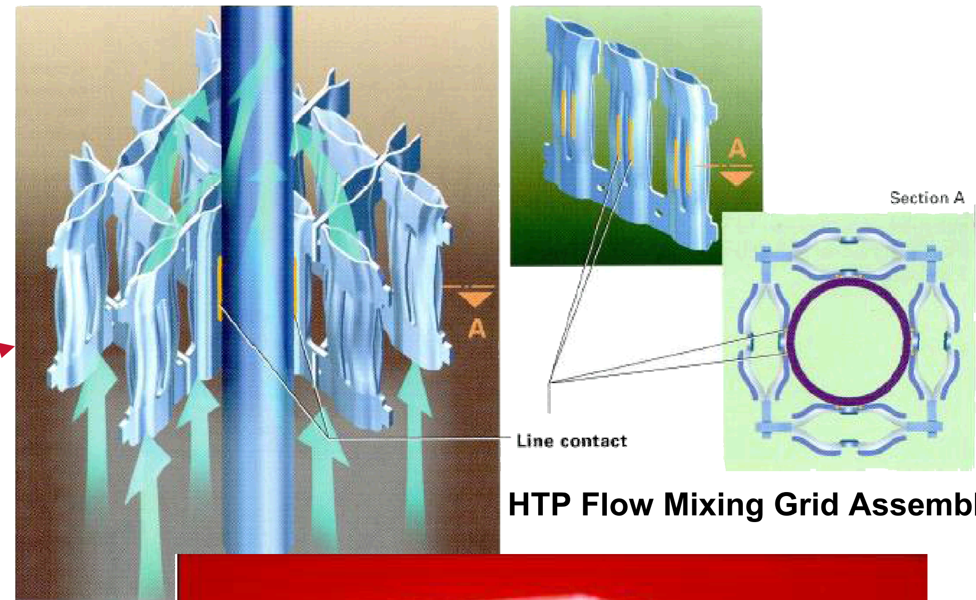
FA Parameter	U.S. EPR FA	Adv. Mark BW	HTP17 12’
FA overall length, in.	189.17	159.85	159.61
FA matrix	17x17	17x17	17x17
Fuel rods / fuel assembly	265	264	264
Fuel rod overall length, in.	179.134	152.16	151.500
FA envelope (SQ.) in	8.426	8.425	8.436
Fuel rod pitch, in.	0.496	0.496	0.496
Guide tubes / assembly	24	24	24
Guide tube type	MONOBLOC™	Standard dashpot GT	Standard dashpot GT
Top nozzle	Low pressure drop	Low pressure drop	Low pressure drop
Top nozzle attachment	Quick disconnect	Quick disconnect	Quick disconnect
Bottom nozzle	FUELGUARD™	Trapper™	FUELGUARD™
End spacer grids	HMP alloy 718	Monometallic alloy 718	Bi-met Z4/alloy 718
Intermediate spacer grids	8- M5™ HTP	6- M5™ monometallic	6- Zirc 4 HTP
Int. grid attachment	Welded to GTs	Swaged ferrules	Welded to GTs

Chapter 4, Reactor:
4.2 Fuel System Design

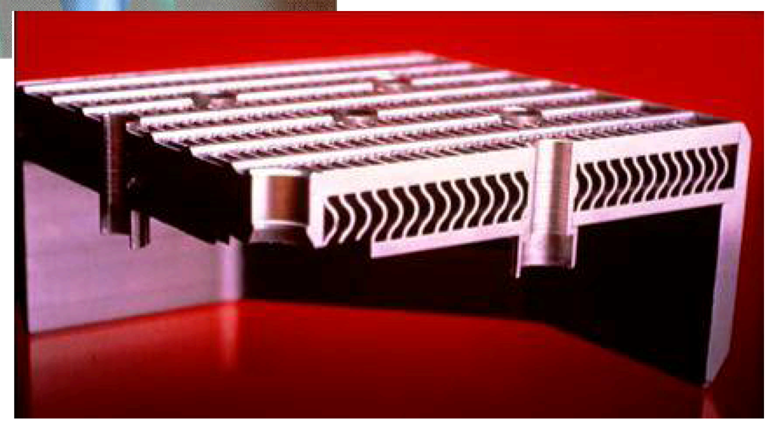
- Fuel Assembly Key Features
- Fuel Assembly Components
- Interfaces with Core Instrumentation
- Operational Experience
- Design Evaluation
- Details of the fuel assembly design for Tier 2, FSAR Chapter 4, are contained in Topical Report ANP 10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report", currently under review by NRC staff.

Chapter 4, Reactor:
4.2 Fuel System Design
Design Features





HTP Flow Mixing Grid Assembly



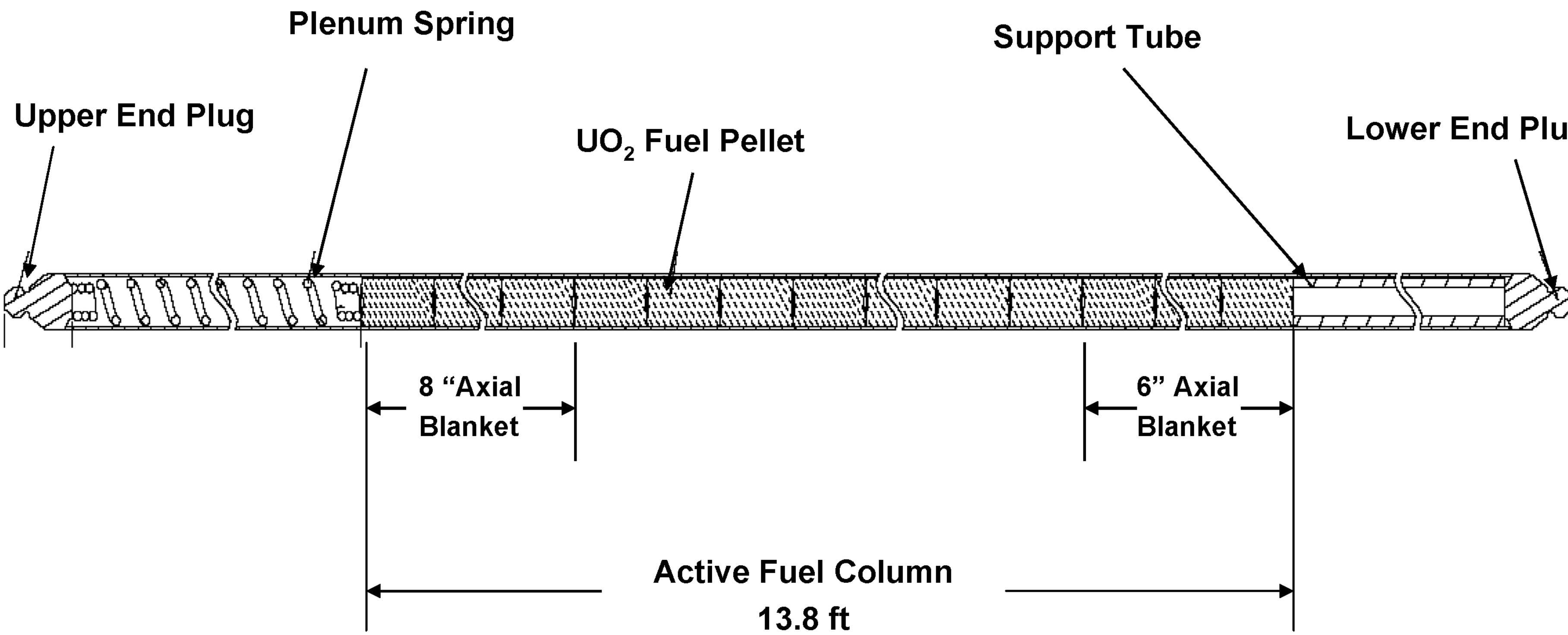
EPR Bottom Nozzle FUELGUARD

Chapter 4, Reactor:
4.2 Fuel System Design
Summary of Component Materials

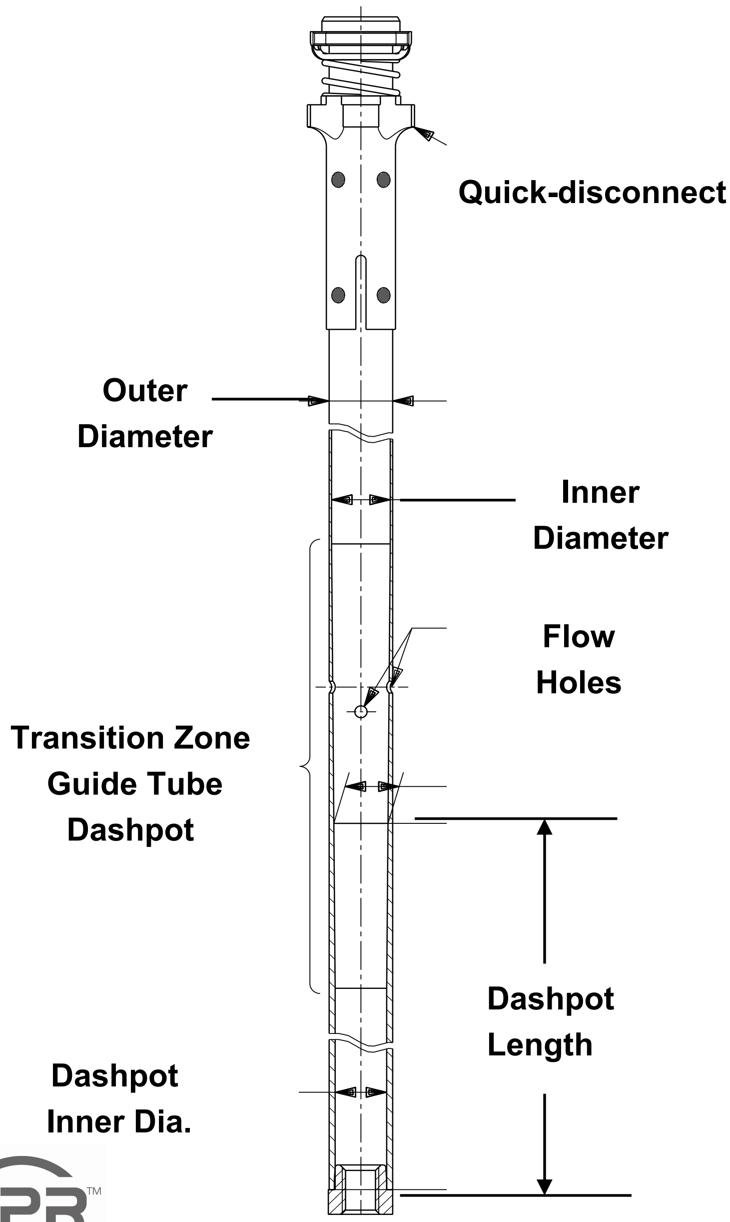
Alloy	Component
M5™	Fuel rod clad, guide tube
	HTP grid spacers
	HMP spacer sleeves
	GT quick disconnect sleeve
	Fuel rod end caps, guide tube plugs
304L S.S.	Top/bottom nozzle structures
	Guide thimble bolt
	Lockwire
321 S.S.	Fuel rod lower support tube
302 S.S.	Fuel rod spring
Nickel Alloy 718	HMP grids
	Clamp screws
	Quick disconnect ring
	Holddown spring leaves
UO_2 and UO_2 + Gd_2O_3	Fuel pellets

Components and materials are consistent with those for existing, proven fuel designs

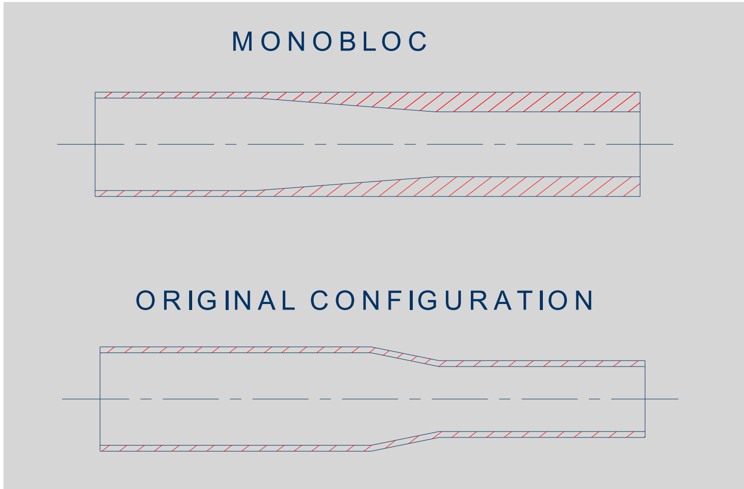
Chapter 4, Reactor: 4.2 Fuel System Design Fuel Rod Assembly



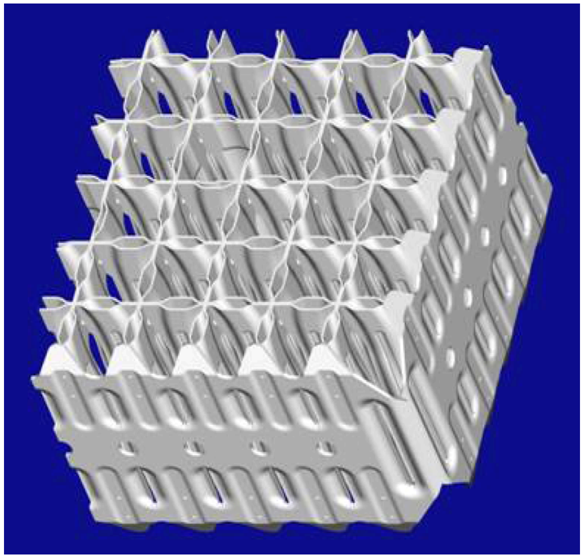
Chapter 4, Reactor:
4.2 Fuel System Design
MONOBLOC
Guide Thimble Assembly



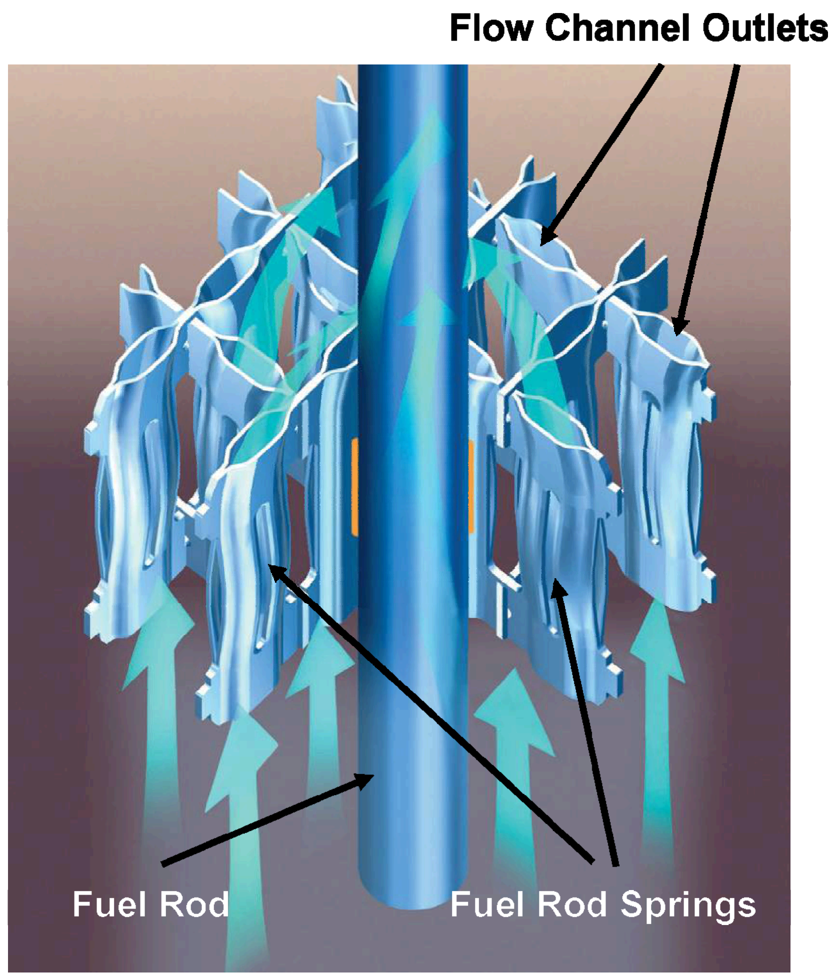
- MONOBLOC guide tubes have two inner diameters (upper dia. & dashpot dia.) and a single outer diameter
- Quick disconnect sleeves at upper end of the guide tube for connection to the top nozzle



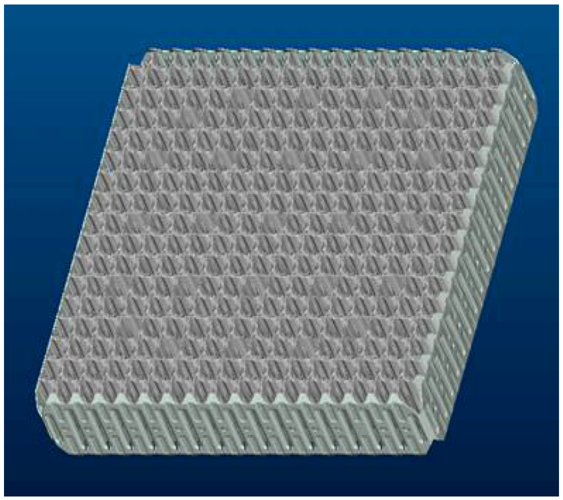
Chapter 4, Reactor:
4.2 Fuel System Design
HTP Spacers for Intermediate Grids



- **Major characteristics**
  - **Material: M5**
  - **Curved flow channel outlets for enhanced mixing**
  - **Large line contact with fuel rod cladding**
  - **Grids welded to guide tubes**
  - **EPR specific CHF correlation approved**

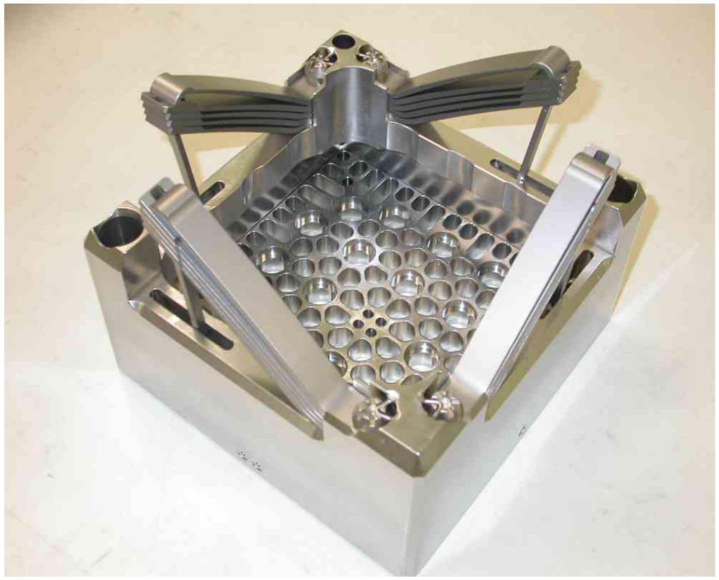


Chapter 4, Reactor:
4.2 Fuel System Design
HMP Spacer for Top and Bottom End Grids



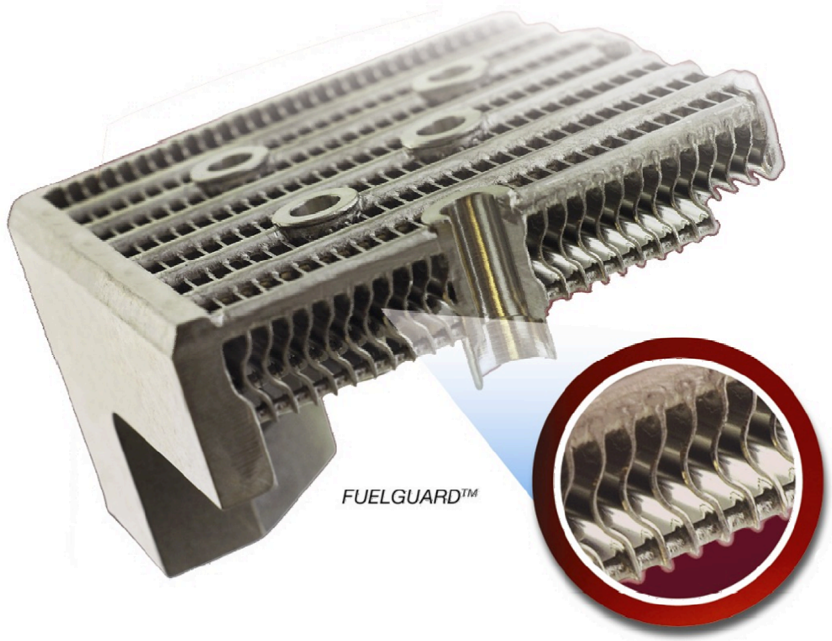
- **Major characteristics**
  - **Material: Nickel Alloy 718**
  - **Straight flow channels**
  - **Large line contact with fuel rod cladding**
  - **Low irradiation relaxation for rod support**

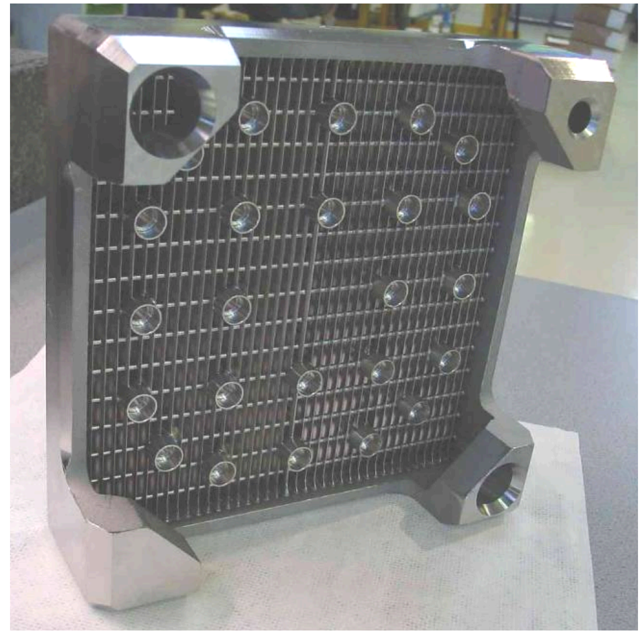
Chapter 4, Reactor:
4.2 Fuel System Design
Low Pressure Drop Top Nozzle



- **Major characteristics**
  - **Material** **304L stainless steel**
  - **Material of springs and screws** **Alloy 718**
  - **Number of springs** **4 sets with 5 springs each**
  - **Grillage flow pattern optimized to optimize strength and pressure drop**

Chapter 4, Reactor:
4.2 Fuel System Design
Robust FUELGUARD™ Bottom Nozzle

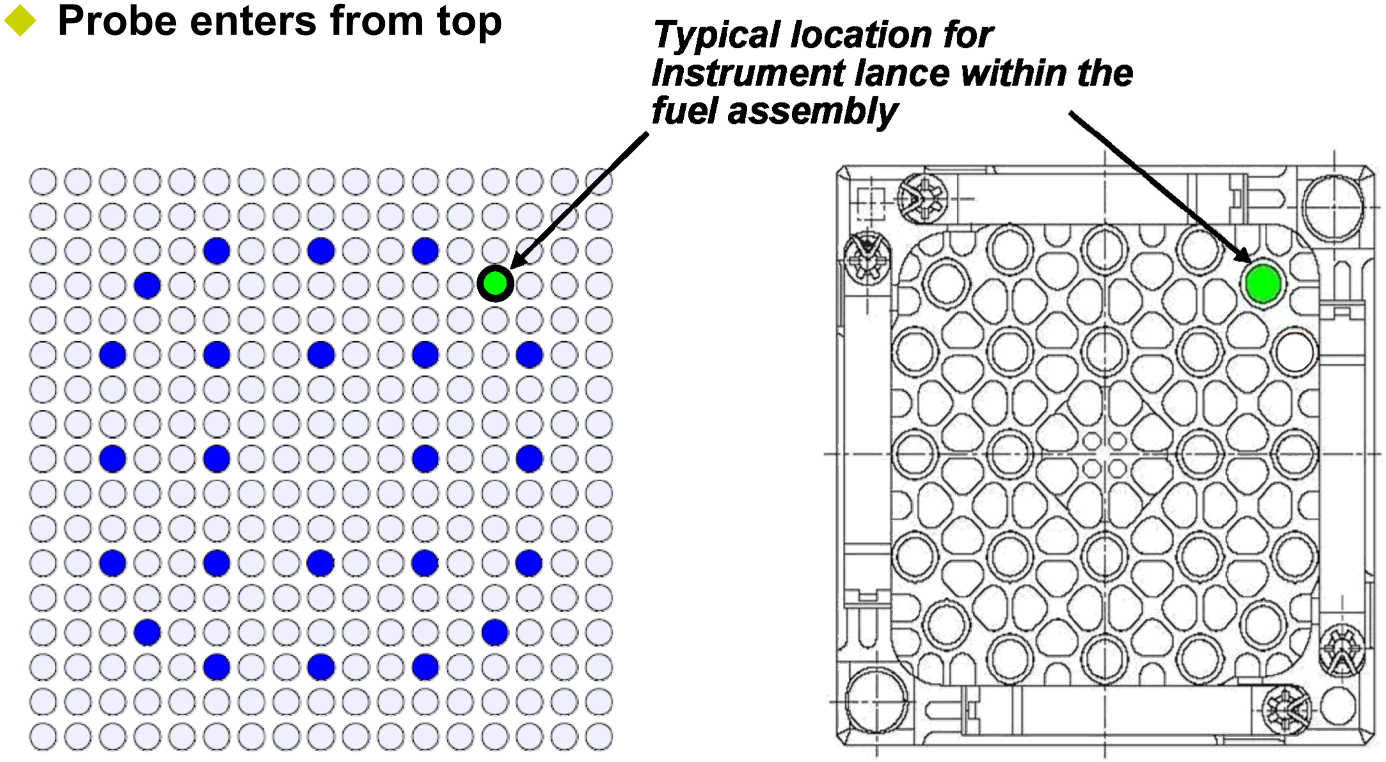




- Frame machined from 304L casting
- Curved blades, rods, bushings made of 304L
- Connected by brazing
- Debris trapping

Chapter 4, Reactor:
4.2 Fuel System Design
Interfaces with Core Instrumentation

- **Instrumentation positioning**
  - **Probe enters select guide tubes on specific fuel assemblies**
  - **No center instrument tube**
  - **Probe enters from top**



Chapter 4, Reactor: 4.2 Fuel System Design Operational Experience

- **M5 alloy**
  - 33 reloads in 15 different U.S. reactors have used the M5 alloy in more than 2300 fuel assemblies
  - Globally, over 1.5 million M5 fuel rods have operated in ~ 6500 fuel assemblies, in 57 reactors
  - More than 3000 fuel assemblies, with M5 fuel rods and guide tubes operated in 37 reactors
  - Globally used in fuel rod arrays from 14x14 to 18x18
  - Maximum fuel rod burnup of 68 GWd/mtU achieved in lead test assemblies
- **HTP spacers**
  - > 18 years of experience, discharge burnups of up to 70 GWd/mtU
  - Range of fuel rod arrays: 14x14, 15x15, 16x16 17x17 and 18x18
  - In reactors supplied by various vendors, Combustion Engineering (CE), Framatome, Westinghouse, Siemens and Babcock & Wilcox, including full cores of HTP fuel
  - No Flow Induced Vibration (FIV) fretting failures due to classical GTRF (grid-to-rod-fretting)
- **14ft cores**
  - Europe (France, first batch in 1983)
    - 24 different 14 foot reactors (2 designs)
      - 20 1300 MW design plants
      - 4 1450 MW (N4) design plants
    - Over 25,000 14 foot fuel assemblies delivered
  - United States
    - South Texas Project (STP)

Chapter 4, Reactor: 4.2 Fuel System Design Design Evaluation

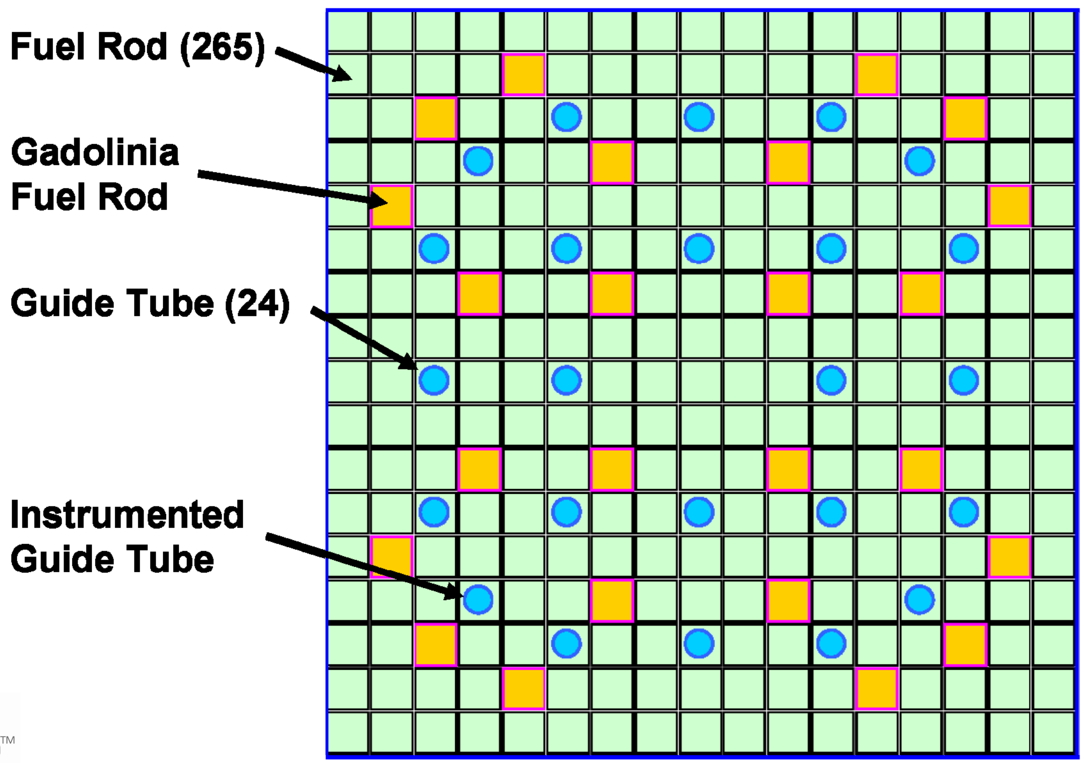
- **Design criteria**
  - Fuel system damage criteria
  - Fuel rod failure criteria
  - Fuel coolability (Tier 2, Chapter 15 analysis)
- **Design consistent with NUREG-0800, SRP Section 4.2**
- **Design evaluation performed with NRC approved codes and methods**
  - Topical report ANP-10263P(A) “Codes and Methods Applicability Report for the U.S. EPR”
- **Results are summarized in topical report ANP 10285P, “U.S. EPR Fuel Assembly Mechanical Design Topical Report,” currently under review by NRC staff**

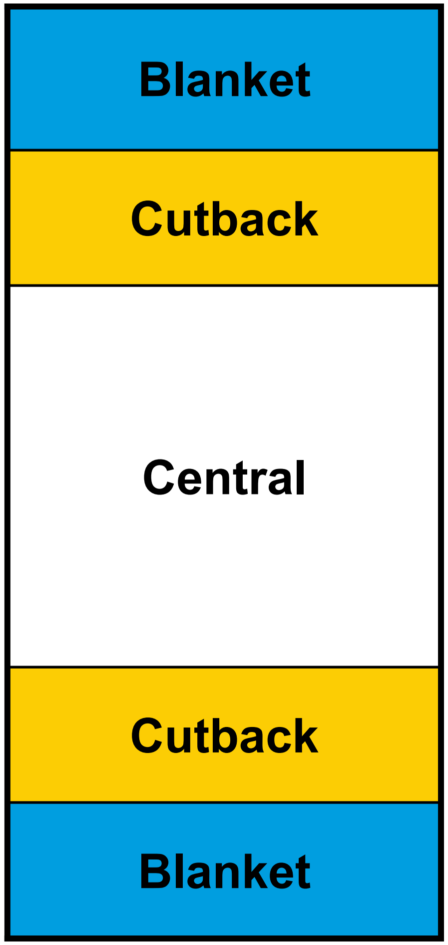
Chapter 4, Reactor: 4.3 Nuclear Design

- Core design conforms to NUREG-0800, SRP 4.3 guidance
- Power distributions maintained within design peaking limits
- Reactivity coefficients considered through cycle lifetime
- Control rod function for adequate shutdown margin and control axial oscillations
- New features to U.S. EPR core design:
  - Heavy reflector
  - Aeroball Measurement System (AMS)
  - Incore detectors contribute to protection system
  - Annular control rods

Chapter 4, Reactor: 4.3 Nuclear Design Fuel Assembly Loading Design

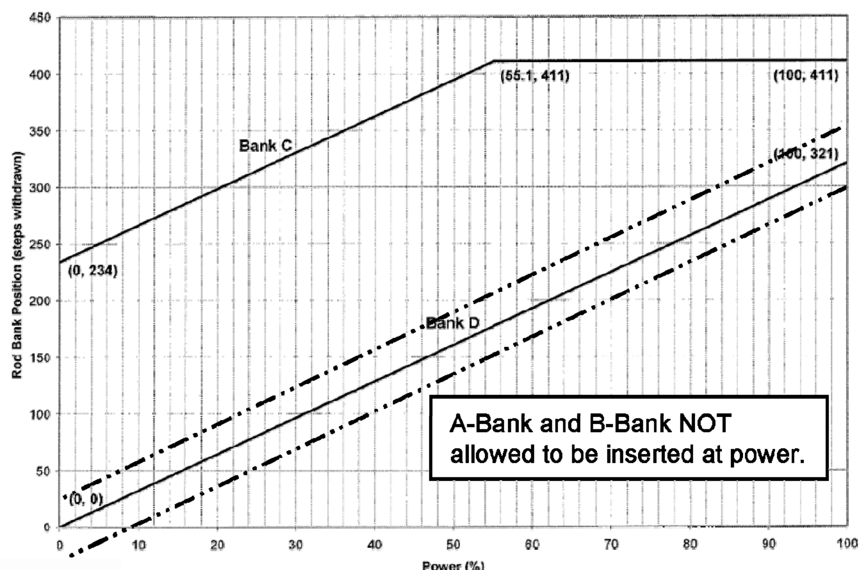
- Fuel and poison loadings follow established design practices to maintain power distribution within design limits and reactivity control for cycle length





Chapter 4, Reactor:
4.3 Nuclear Design
Rod Cluster Control Assembly Patterns

- **Control rod bank worth and arrangements follow standard design approach for base load operations**
- **Power dependent insertion limits for shutdown margin and axial power shape limits**



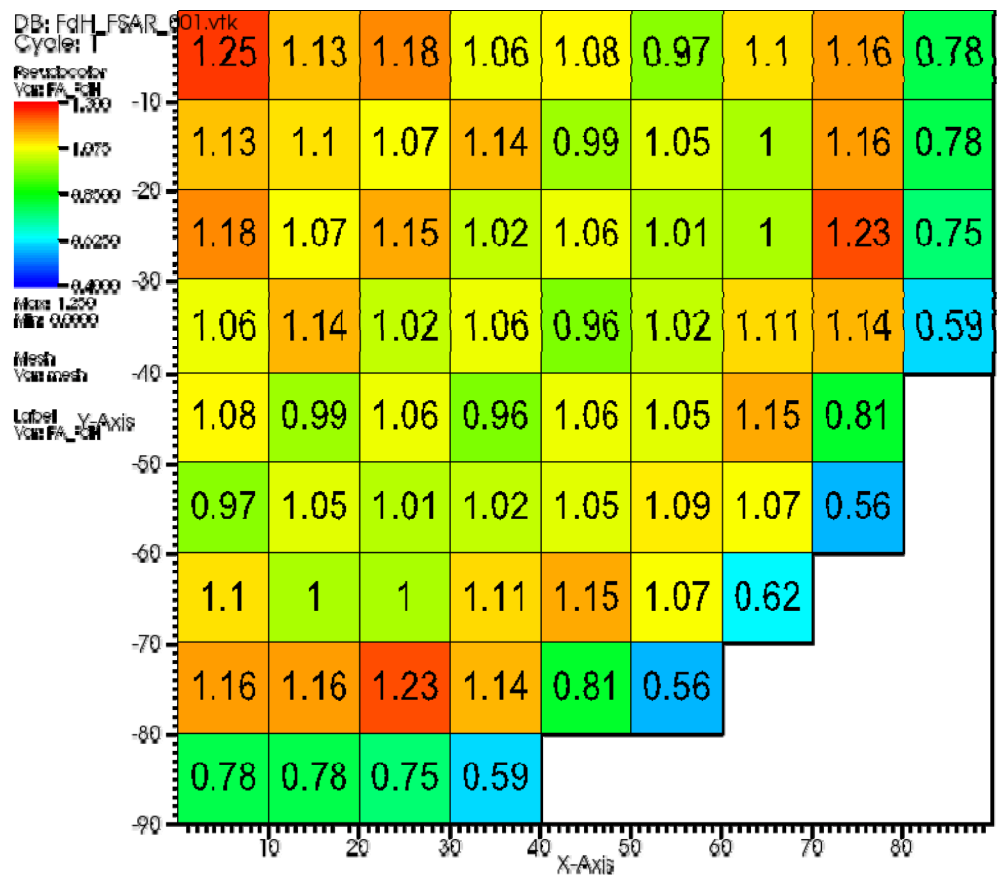
	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T
17									SC								
16						C		SA		SA		C					
15			SC		A		B		D		B		A		SC		
14				SA		SB						SB		SA			
13			A		D		SC		C		SC		D		A		
12		C		SB				SA		SA				SB		C	
11			B		SC		B				B		SC		B		
10		SA				SA		SB		SB		SA				SA	
9	SC		D		C				D				C		D		SC
8		SA				SA		SB		SB		SA				SA	
7			B		SC		B				B		SC		B		
6		C		SB				SA		SA				SB		C	
5			A		D		SC		C		SC		D		A		
4				SA		SB						SB		SA			
3			SC		A		B		D		B		A		SC		
2						C		SA		SA		C					
1									SC								

Control Banks		Shutdown Banks	
D	9	SC	16
C	12	SB	12
B	12	SA	20
A	8		

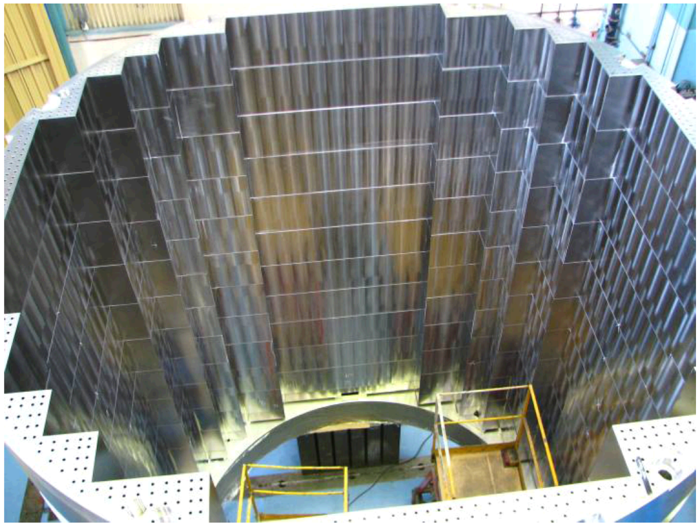
Total RCCA: 89

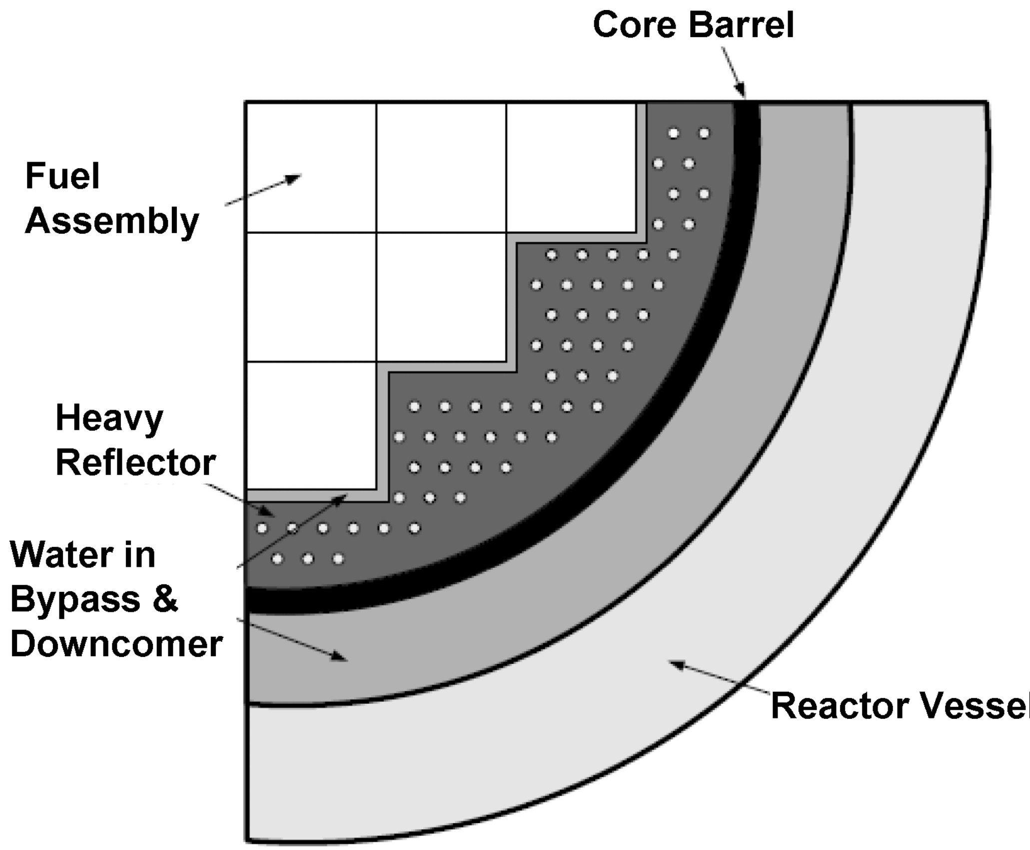
Chapter 4, Reactor:
4.3 Nuclear Design
Radial Power Distributions Cycle 1

- **18-month base load cycle design**
- **Maximum average fuel rod Burnup < 62,000 MWD/MTU**
- **Average LHGR 5.21 kW/ft**
- **Peak LHGR 13.56 kW/ft**
- **100% Power radial power peaking (F∆H) design limit < 1.7**
- **100% Power local power peaking (FQ) design limit <2.6**
- **Excess of 3000 pcm shutdown margin**



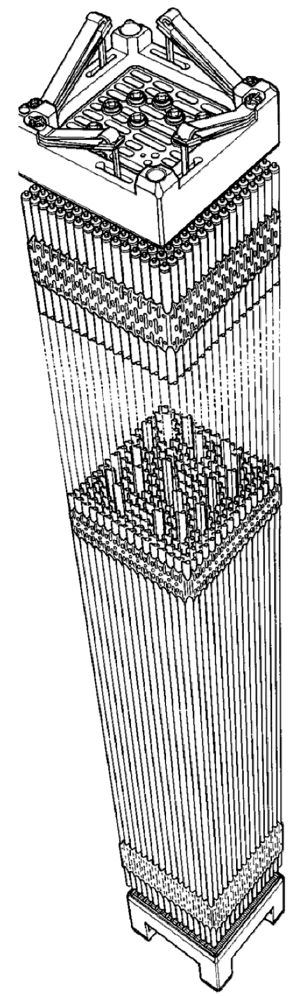
Chapter 4, Reactor: 4.3 Nuclear Design Heavy Reflector





Chapter 4, Reactor:
4.4 Thermal-Hydraulic Design

Parameter	Current US 4-Loop	Palo Verde	U.S. EPR
Thermal Power, MWth	3411	3998	4590
Electrical Power (Net), MWe	1120	1314	1600
Plant Efficiency, Percent	33	33	35
Number of Fuel Assemblies	193	241	241
Hot Leg Temperature, °F	616	620	624
Cold Leg Temperature, °F	559	558	564
Vessel Average Temperature, °F	588	589	594
Primary System Operating Pressure, psia	2250	2250	2250
Reactor Coolant Flow per Loop, gpm	100,500	111,000	125,000
Average Coolant Flow per Assembly, gpm	2082.9	1842.3	2074.7
Core Average Linear Heat Rate, kW/ft	5.58	5.63	5.21
Peak Linear Heat Rate, kW/ft	14.51	14.64	13.56

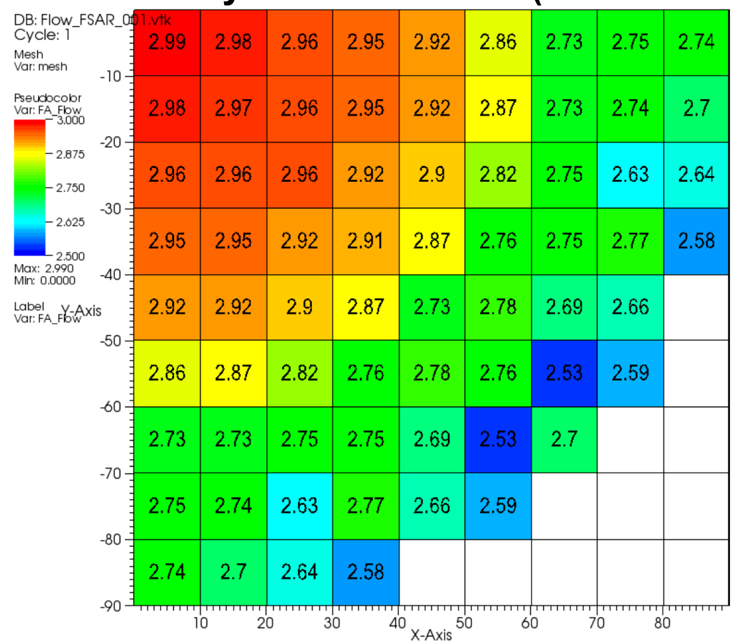


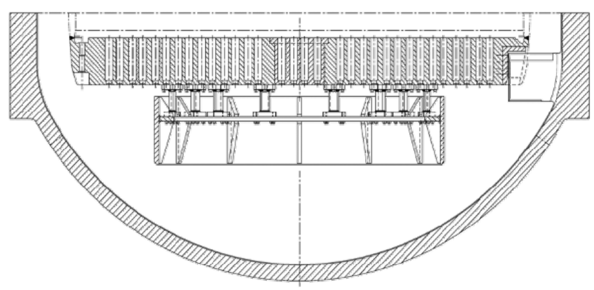
Core design parameters similar to current plants

Chapter 4, Reactor:
4.4 Thermal-Hydraulic Design
Core Inlet Flow Distribution

- Inlet flow distributions confirmed with 1/5 scale lower internals hydraulics Juliette mockup test (Le Creusot)

Fuel Assembly Inlet Mass Flux (Mlbm/hr-ft²)

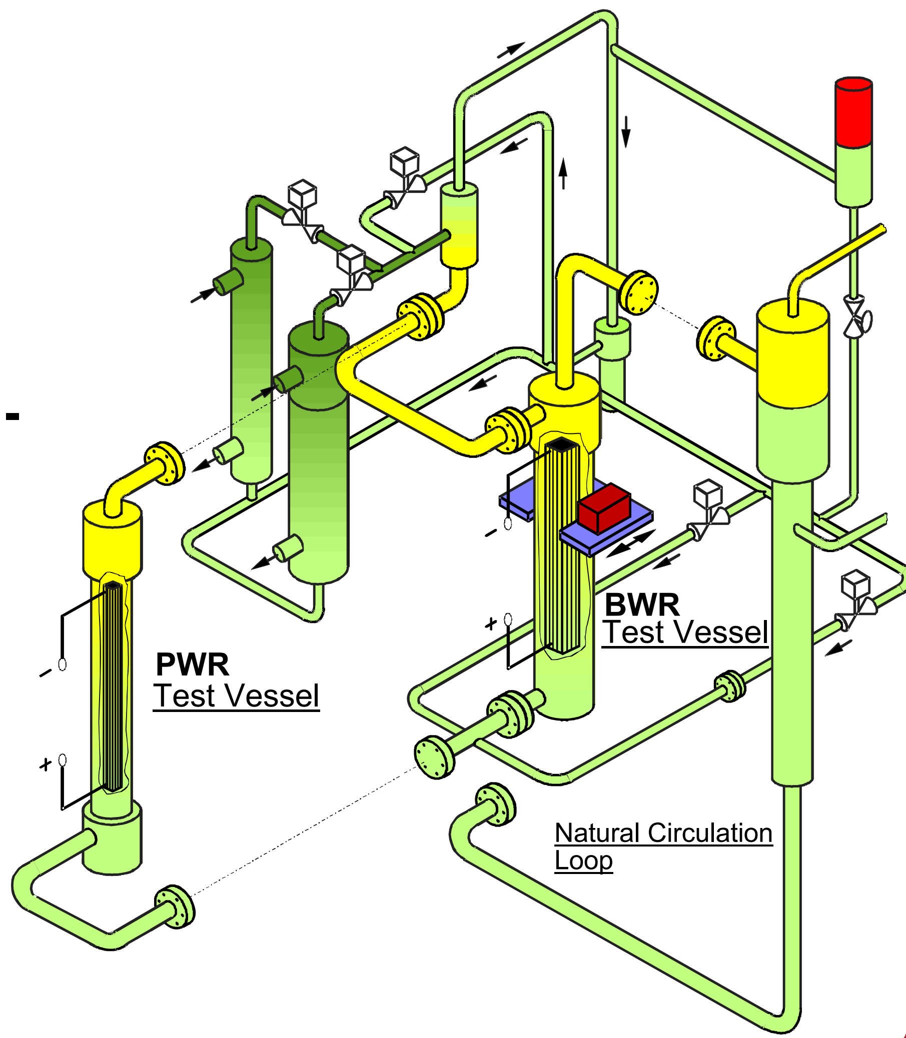




Lower Plenum Flow Distribution Device

Chapter 4, Reactor:
4.4 Thermal-Hydraulic Design
Critical Heat Flux Testing for Correlation

- **CHF correlation derived specifically for EPR HTP spacer grids**
  - **ANP-10269P(A) for ACH-2 CHF Correlation**
- **Testing at Karlstein Thermal-Hydraulic Facility (KATHY) included:**
  - **Multiple heated lengths**
  - **Uniform and non-uniform axial heat flux shapes**

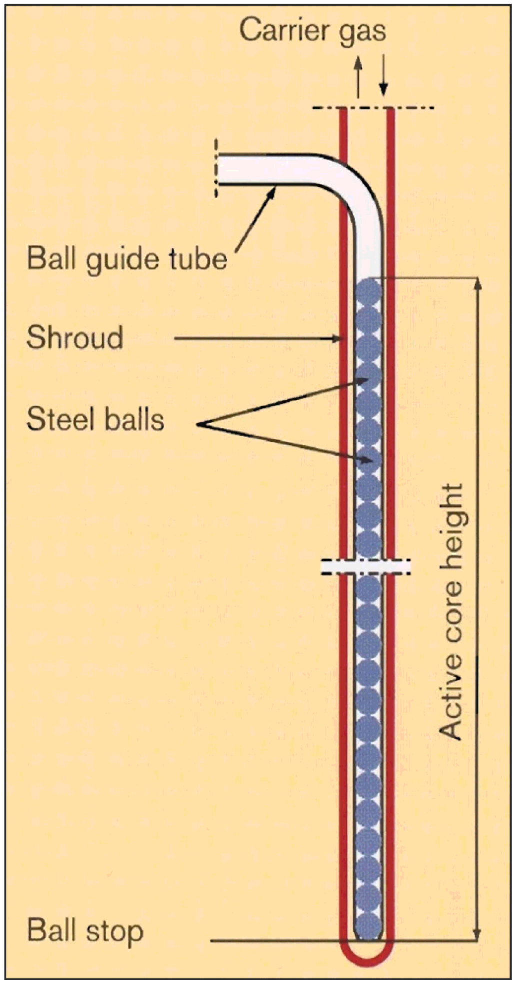


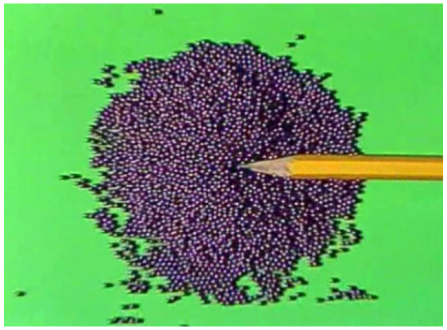
Chapter 4, Reactor: 4.4 Thermal-Hydraulic Design Core Instrumentation

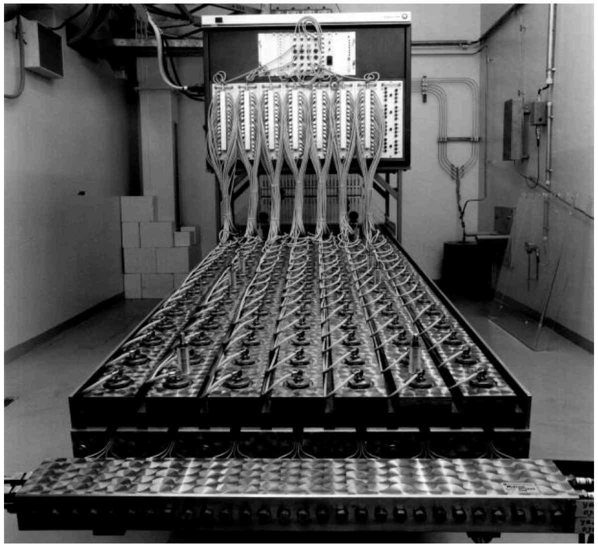
- **Excore Instrumentation:**
  - **Source Range Detectors**
  - **Intermediate Range Detectors**
  - **Power Range Detectors**
- **Fixed Incore Instrumentation:**
  - **Self Powered Neutron Detectors (SPNDs) also called Power Density Distribution System (PDDS)**
  - **Thermocouples to measure core outlet temperature**
- **Aeroball Measurement System (AMS)**

Chapter 4, Reactor:
4.4 Thermal-Hydraulic Design
Aeroball Measurement System

- **Electromechanical, computer controlled, online flux mapping measurement system**
- **On-demand power distribution measurement ~10min**
- **Large measurement array provides 1440 measurement points distributed over the active core volume**
- **γ -radiation emitted by the vanadium-alloyed steel balls read by detector arrays outside of the biological shield**
  - **0.067” diameter spheres**
  - **83.36% Fe, 14.5% Cr, 1.54% V, 0.6% C**

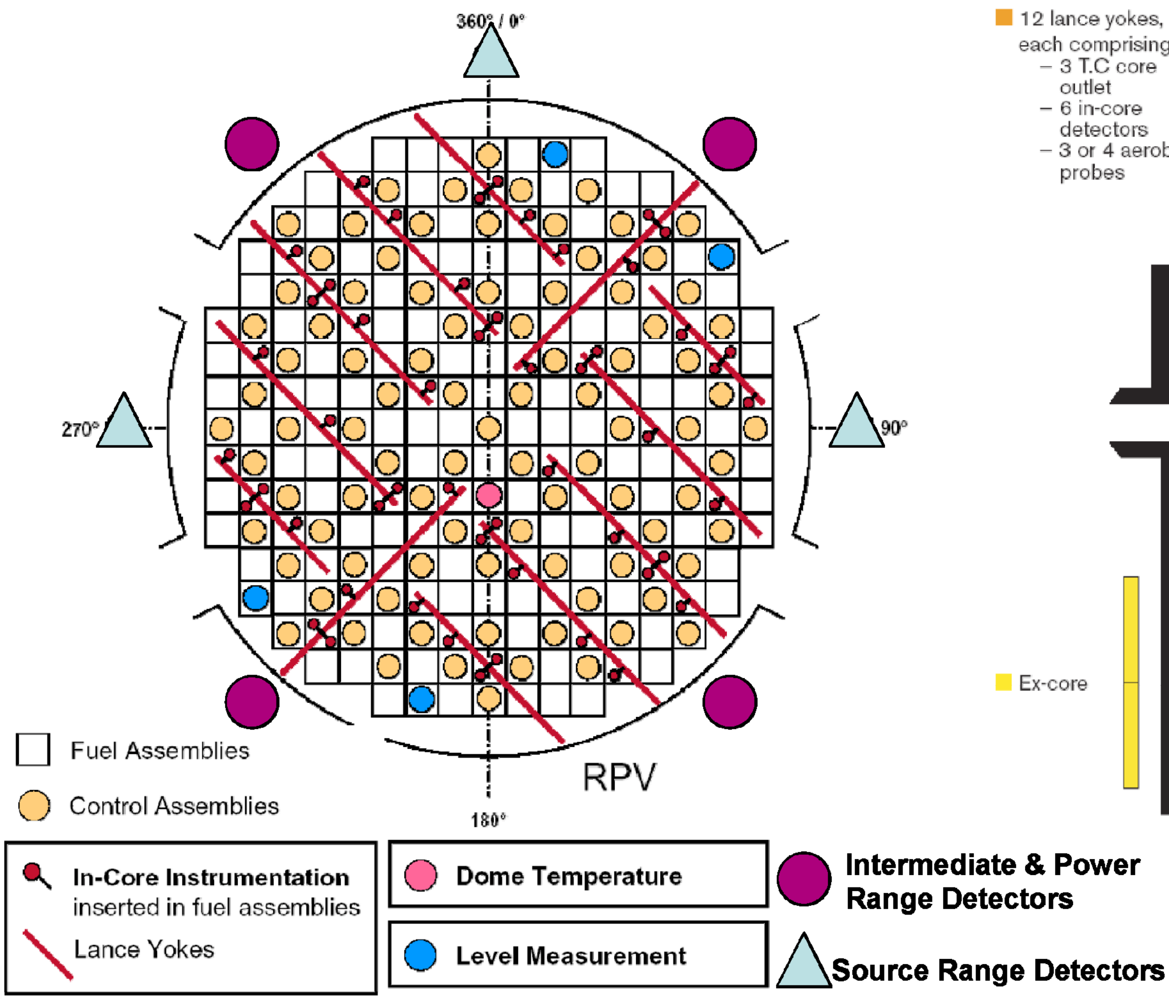


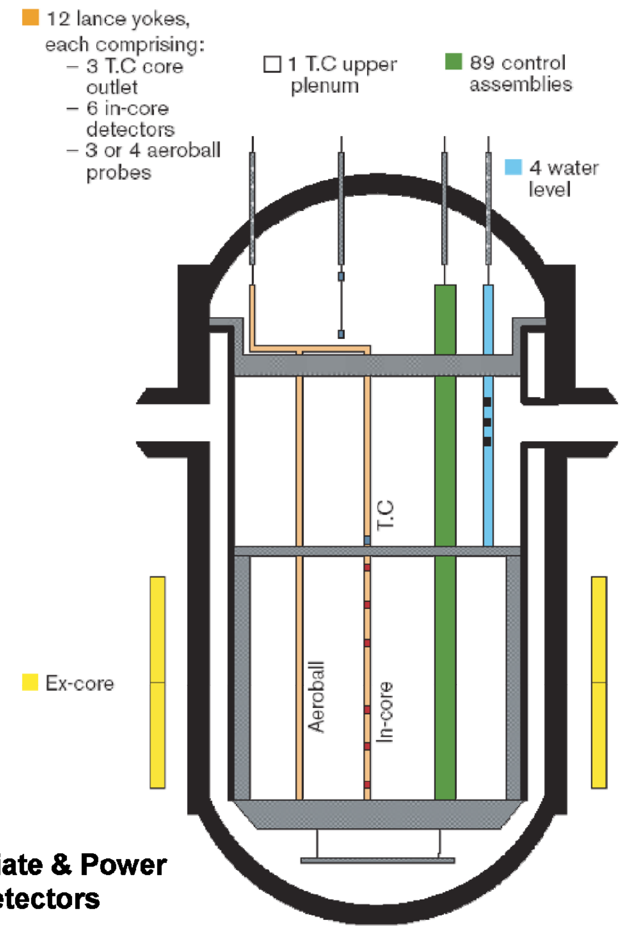




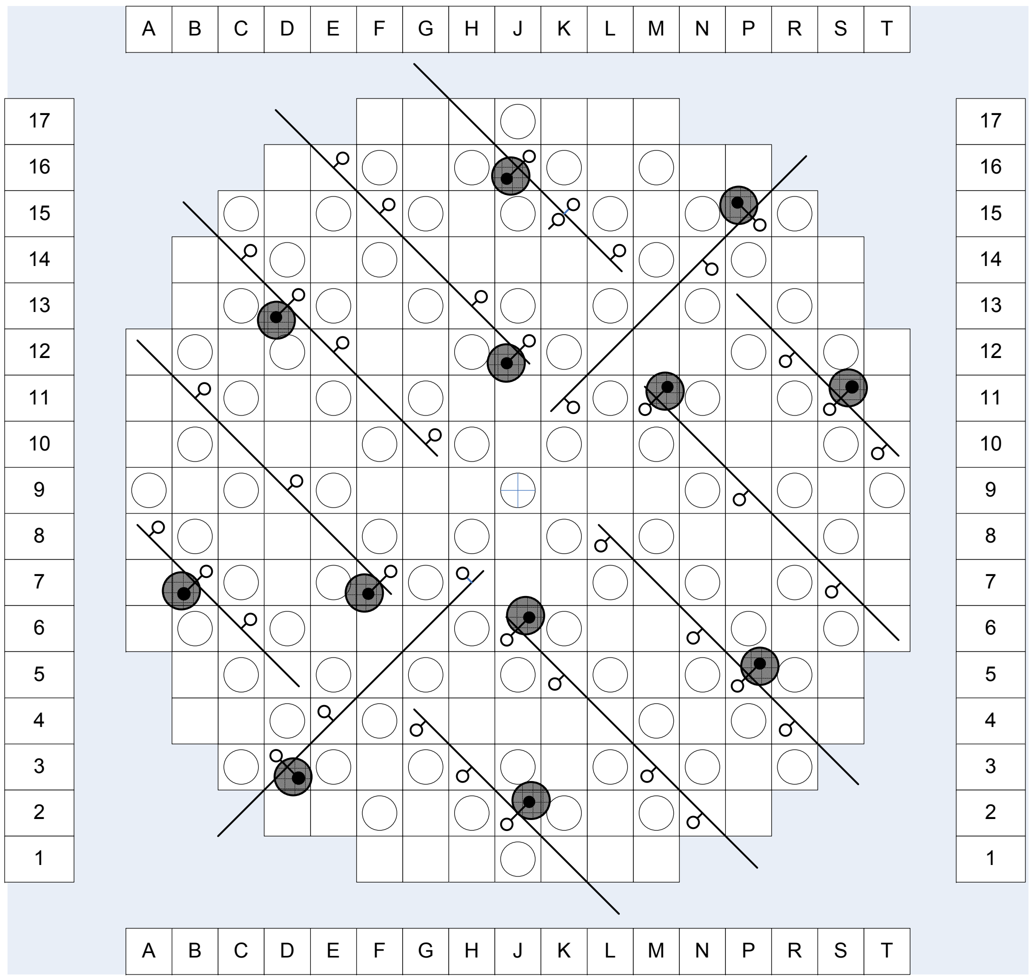
AMS Measurement Table currently used in KONVOI (Type) Plants with more than 35 years of operational history

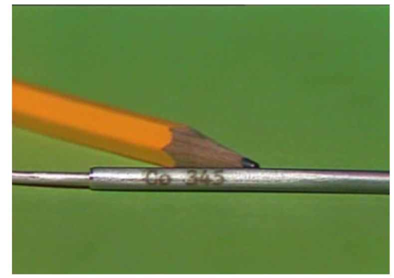
Excore and Incore Instrumentation Arrangements



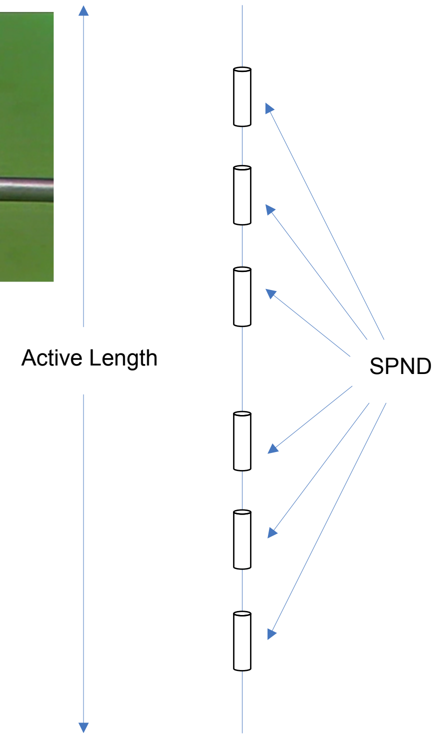


Chapter 4, Reactor:
4.4 Thermal-Hydraulic Design
Incore Neutron Detector Allocation





SPND Length:
21 cm (typical)



AMS probe (40)
SPND string (12)
Control assembly (89)
Instrumentation lance (12)

Co-59 Self Powered Neutron Detectors (SPND)

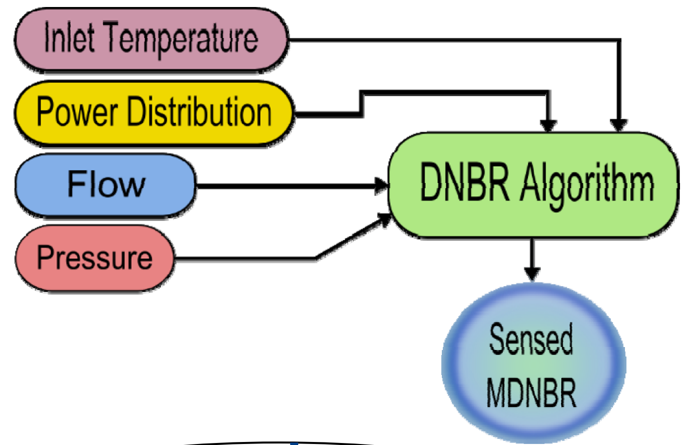
I&C Functions:

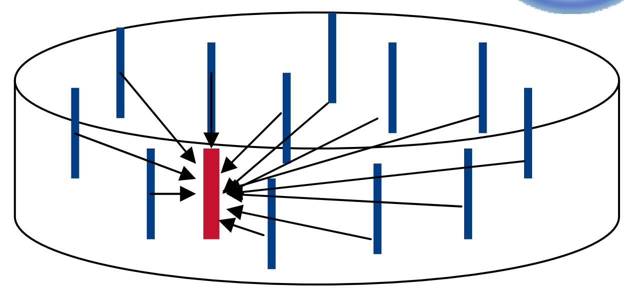
- **Departure from Nucleate Boiling (DNB)**
- **Linear Power Density (LPD)**
- **Axial Power Shape**

Chapter 4, Reactor: 4.4 Thermal-Hydraulic Design Incore Instrument Requirements

- **DNB Ratio Monitoring**
  - Each SPND string is calibrated to indicate the LHGR axial distribution of the DNBR limiting fuel rod
  - DNBR estimated using reconstructed power distribution and thermal hydraulic boundary conditions
- **High Linear Power Density Monitoring**
  - All SPNDs in an axial elevation are calibrated to read the maximum heat rate in that zone
  - Signals also used to provide indication of azimuthal imbalance for DNBR thresholds
- **Axial Power Shape Monitoring**
  - SPNDs in upper and lower half of core are calibrated to read the axial offset, defined as:

$$AO = \frac{P_{Top} - P_{Bottom}}{P_{Top} + P_{Bottom}} \times 100\%$$

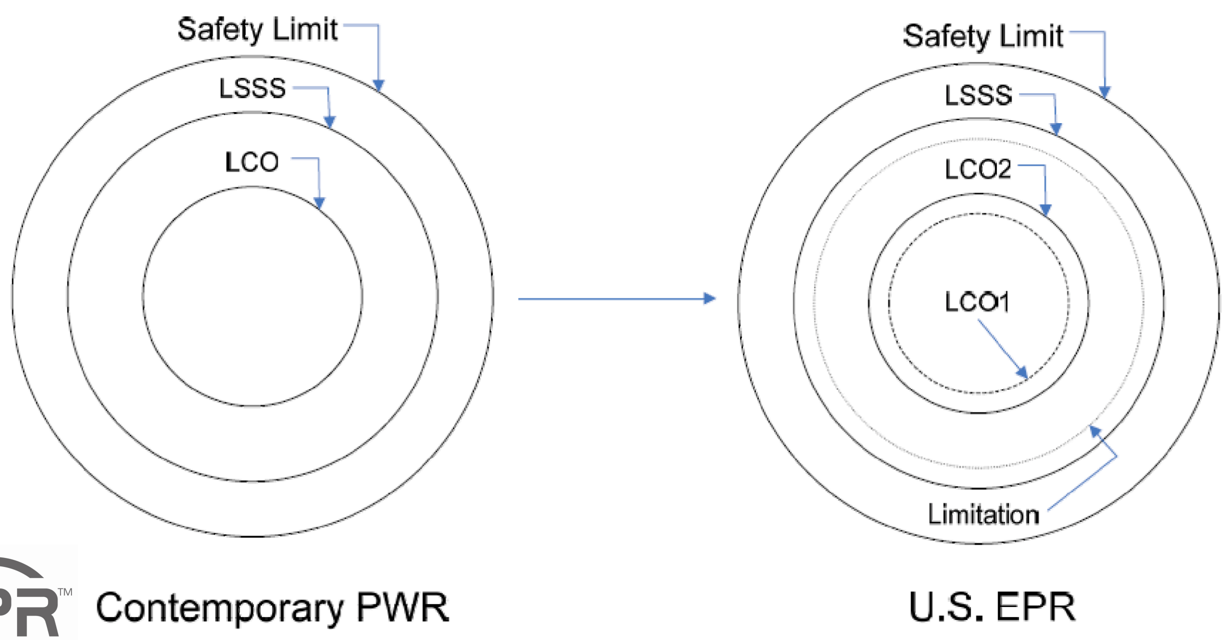




Design limits are monitored and protected with online incore systems instead of implied performance from excore detectors

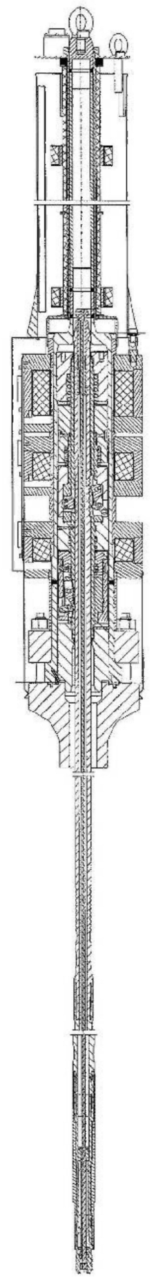
Chapter 4, Reactor:
4.4 Thermal-Hydraulic Design
Holistic Protection of Fuel SAFDLs

- **Graduated approach to SAFDL protection using incore detectors**
- **Reactor Control, Surveillance, and Limitations - RCSL**
  - **LCO1 – alarms, block rod withdrawal, block turbine load increase; passive**
  - **LCO2 – turbine generator runback and insertion of rods to match reduction; active**
  - **Limitation – rapid power runback with scram of selected RCCA Bank subgroups**
- **Reactor Protection System - PS**
  - **Limiting Safety System Settings (LSSS) – full reactor trip**



Approach is detailed in ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR"

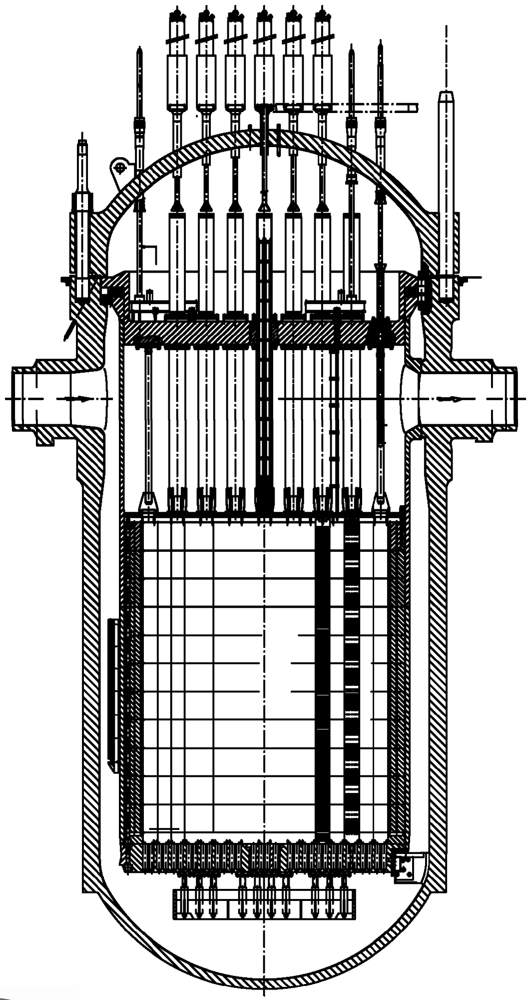
Chapter 4, Reactor:
4.5 Reactor Materials
Control Rod Drive Mechanism



- All pressure boundary materials are designed in accordance with ASME Code
- Pressure boundary materials
  - Austenitic (stabilized) and martensitic stainless steel materials and addressed in Tier 2, Section 5.2.3 and ASME Code with corresponding weld materials
- Non-pressure boundary materials
  - Austenitic (stabilized) & martensitic stainless steels, cobalt-chromium alloy, nickel-base materials and cobalt base materials
  - Cobalt materials are used in a very small portion where alternate material will not perform satisfactorily.

Components and materials are consistent with those for existing, proven designs

Chapter 4, Reactor:
4.5 Reactor Materials
Reactor Internals and Core Support



- **Materials are selected based on compatibility with their environment**
- **Parts exposed to reactor coolant are made of corrosion resistant material**
- **Components are non-pressure boundary, though materials are ASME**
  - **Made mostly of austenitic stainless steel, some martensitic stainless steel coated with cobalt alloy for wear resistance**
  - **Support Pins/Bolting: Mostly low-carbon austenitic stainless steels; some cold-worked austenitic material with a limited maximum yield strength**

Components and materials are consistent with those for existing, proven designs

Chapter 4, Reactor: 4.6 Design of Reactivity Control Systems

- **Control Rod Drive System (CRDS)**
  - **Compensates for fuel and water temperature changes**
  - **Maximum reactivity insertion rate limited by bank configuration and maximum rod speed (29.5in/min)**
  - **Maintains minimum shutdown margin during AOOs with one rod stuck out**
- **Chemical and Volume Control System (CVCS)**
  - **Soluble boron for cycle fuel depletion and xenon burnout**
  - **Limits on rate and duration of dilution**
  - **Safety Injection System (SIS) for LOCA (Tier 2, Section 6.3)**
  - **Extra Borating System (EBS) for cold shutdown (Tier 2, Section 6.8)**

Two independent reactivity control systems for normal and abnormal conditions

FSAR Chapter 4 Reactor: Section Topics

- Section 4.1 – Summary Description
- Section 4.2 – Fuel System Design
- Section 4.3 – Nuclear Design
- Section 4.4 – Thermal-Hydraulic Design
- Section 4.5 – Reactor Materials
- Section 4.6 – Functional Design of Reactivity Control Systems

Reactor is designed and evaluated using existing methods and incorporates some new (to U.S.) features

Nomenclature

AMS	Aeroball Measurement System
AOO	Abnormal Operational Occurrence
DNB(R)	Departure from Nuclear Boiling (Ratio)
FDH	Axially Integrated Radial Power Peaking (Enthalpy Rise) Factor
FQ	Local Power Peaking (Heat Flux) Factor
(H)LPD	(High) Linear Power Density (kW/ft)
HMP	High Mechanical Performance (spacer grids)
HTP	High Thermal Performance (spacer grids)
IRD	Intermediate Range Neutron Detector
LCO	Limiting Condition for Operations
LHGR	Linear Heat Generation Rate (kW/ft)
LSSS	Limitation Safety System Setting
PA	Postulated Accident
PRD	Power Range Neutron Detector
RCCA	Rod Cluster Control Assembly
RCS	Reactor Coolant System
RCSL	Reactor Control, Surveillance, and Limitation
RT	Reactor Trip
SAFDL	Specified Acceptable Fuel Design Limit
SDM	Shutdown Margin
SPND	Self-Powered Neutron Detector
SRD	Source Range Neutron Detector

AREVA

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Presentation to the ACRS Subcommittee

AREVA EPR Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 4: Reactor

March 3, 2010

Staff Review Team

- **Technical Staff**
  - **Fred Forsaty**
    Reactor Systems, Nuclear Performance, and Code Review Branch
  - **John Budzynski**
    Reactor Systems, Nuclear Performance, and Code Review Branch
  - **Shanlai Lu**
    Reactor Systems, Nuclear Performance, and Code Review Branch
  - **John Honcharik**
    Component Integrity Branch
  - **Robert Davis**
    Component Integrity Branch

  - **Project Managers**
  - **Getachew Tesfaye**
  - **Jason Carneal**

Overview of DCA

SRP Section/Application Section		No. of Questions	Status Number of OI
4.2	Section Title Fuel System Design	15	2
4.3	Section Title Nuclear Design	24	2
4.4	Section Title Thermal-Hydraulic Design	37	3
4.5.1	Section Title Control Rod Drive System Structural Materials	7	2
4.5.2	Section Title Reactor Internals and Core Support Materials	11	3
4.6	Section Title Functional Design of Reactivity Control Systems	10	2
Totals		104	14

Description of Open Items

- RAI 339, Question No. 04.02-17: Tracks the open review of topical report ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report." Note: This open item affects Sections 4.2, 4.3, and 4.4 of this report
- RAI 318, Question No. 04.02-16: Requests documentation of the seismic-LOCA evaluation of the EPR fuel assembly design
- RAI 344, Question No. 04.03-27: Provide a COL information item to ensure proper benchmarking of the fluence calculation methodology for the U.S. EPR
- RAI 344, Question No. 04.03-28: Tracks the open review of topical report ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report"
- RAI 308, Question No. 04.04-59: Provide a testing plan to verify the accuracy of the correction algorithms that are applied to the raw Aeroball Measurement System activation measurements
- RAI 308, Question No. 04.04-60: Provide the methodology to remove the ^{60}Co background from SPND measurements and a description of how this background will be treated by the protection system and the AMS POWERTRAX/E calculations

Description of Open Items

- RAI 325, Question No. 04.04-61: Provide a description of the hydraulic loads analysis for the U.S. EPR fuel and vessel components for normal operating conditions and design basis accident conditions

- RAI 343, Question No. 04.05.01-6: Requests that the applicant modify FSAR Table 5.2-2 to list material specifications and grades for all CRDM pressure boundary components

- RAI 343, Question No. 04.05.01-7: Tracks the applicant's submitted request to ASME Code to extend the properties currently provided in Section II Part D for SA-182 Grade F6NM (UNS S41500) to SA-479 (UNS S41500) material.

- RAI 339, Question No. 04.05.02-9: Requests that the applicable ASME code specifications for the hardfacing material, Stellite 6, be included in the U.S. EPR FSAR Tier 2, Table 4.5.2.

- RAI 339, Question No. 04.05.02-10: Requests documentation in the FSAR or a COL action item to ensure an augmented ASME Code, Section XI inspection program will be developed to verify that IASCC and void swelling does not impact the safety function of the heavy reflector or create loose parts

Description of Open Items

- RAI 339, Question No. 04.05.02-11: Provide a discussion on the prevention of notches on the vertical keys and keyways that can act as stress concentrations and crack initiation sites, which could lead to the loss of function of the heavy reflector
- RAI 366, Question No. 04.06-13: Explain the apparent discrepancy between FSAR Tier 2, Sections 15.1.5 and 4.6.4 regarding credit taken for boron addition via the SIS to mitigate large steam line breaks from hot zero power conditions
- RAI 367, Question No. 04.06-14: Provide an explanation on how the methods described in ANP-10287P, “Incore Trip Setpoint and Transient Methodology for the U.S. EPR Topical Report,” will be implemented and verified

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Seismic-LOCA analysis

The applicant states that no crushing deformation will occur during normal operation and operating basis earthquake conditions based on crush load tests with load limits taken as the 95/95 one sided confidence of the mean elastic limit for grids at beginning of life conditions correcting for operating temperature.

Staff Evaluation

- The FSAR did not contain sufficient information regarding the seismic-LOCA analysis

Open Item

- RAI 318 Question 04.02-16 requests documentation of the seismic-LOCA analysis performed by the applicant

Technical Topics of Interest
Section 4.2 – Fuel System Design

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M5[TM] Growth Issue: **Remaining Technical Issue in Fuel Assembly Mechanical DesignTopical Report (ANP-10285P)**

Current plants with M5 guide tubes have experienced much higher than expected irradiation growth resulting in gap closure between the top nozzle and the core plate during cold shutdown creating assembly bow and control rod becoming stuck in the guide tubes. The EPR fuel design utilizes M5 guide tubes with gap clearances questionable at EOL when 95/95 uncertainties are considered.

Staff Evaluation

- The staff have issued several questions related to the applicability of M5 growth data from current fuel designs to the EPR design and how growth uncertainties are included in the EOL gap closure analyses. The staff have requested that a 95/95 upper growth be determined based on high burnup growth data. The staff evaluation will be documented in the SER on ANP-10285P

Open Item

- RAI 339 Question 04.02-17 tracks the ongoing review of ANP-10285P

Technical Topics of Interest
Section 4.2 – Fuel System Design

COPERNIC Cladding Strain: **Remaining Technical Issue in Fuel Assembly Mechanical Design Topical Report (ANP-10285P)**

The COPERNIC code is used to determine the LHGR limit that meets the 1% cladding strain (elastic + plastic) limit for AOOs. Audit calculations with the FRAPCON-3.4 code predicted a lower LHGR limit (higher cladding strains) than that predicted with COPERNIC using the same input for both codes. Further examination of the data used to verify the COPERNIC code demonstrated that it under-predicted all of the strain data for power transient time periods on the order of AOOs (a few minutes to seconds) while over-predicting time periods of hours.

Staff Evaluation

- The staff have issued several questions for further COPERNIC data and code comparisons and the conservatism in the 1% strain limit for M5 cladding.

Open Item

- RAI 339 Question 04.02-17 tracks the ongoing review of ANP-10285P

Technical Topics of Interest
Section 4.2 – Fuel System Design

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M5^{TM} Hydride Limit: **Remaining Technical Issue in Fuel Assembly Mechanical Design Topical Report (ANP-10285P)**

The applicant has responded that a hydrogen limit is not necessary for M5^{TM} , however, this is not consistent with SRP 4.2.II.1A.IV.

Staff Evaluation

- The staff have requested that a hydrogen limit be proposed and justified as per SRP 4.2.II.1.A.IV.

Open Item

- RAI 339 Question 04.02-17 tracks the ongoing review of ANP-10285P

Fluence Calculation Methodology

- The applicant has provided a generic fluence calculation methodology for the U.S. EPR design.
- The U.S. EPR incorporates a heavy reflector which is an evolutionary change from the operating fleet
- Staff Evaluation
  - Until the first measured fluence value is available, the actual vessel fluence is only a small fraction of the EOL fluence.
  - NRC-approved vessel fluence methodology described in BAW-2241P-A, “Fluence and Uncertainty Methodologies,” April 2006
  - The calculation meets RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” March 2001 with the exception of benchmarking of the methodology for the U.S. EPR design.

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- **Fluence Calculation (continued)**
    - The staff approved BAW-2241P-A based on surveillance capsule and dosimeter data from operating reactors. The approved version of BAW-2241P-A requires the use of surveillance capsule and dosimeter data points to verify the applicability of the methodology to any particular reactor vessel
    - No such data exists for any particular U.S. EPR reactor which incorporates a heavy reflector
    - The use of the methodology after the initial 10 EFPYs of operation cannot be justified without benchmarking
- **Open Item**
    - RAI 344 Question 04.03-27 requests the applicant provide a COL information item to ensure proper benchmarking of the fluence calculation methodology

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Rod Ejection Methodology Topical Report (ANP-10286P)

- Ejected rod worth computed using the methods described in Chapter 4.3. These values are then used in the Chapter 15 Ejected Rod analysis to determine energy deposition in the fuel
- The applicant has responded to all RAIs on ANP-10286P and there are no outstanding technical issues.
- Staff Evaluation
  - The staff is currently processing the draft SER
- **Open Item**
  - RAI 344 Question 04.03-28 will track ANP-10286P until the SER is issued.

Technical Topics of Interest
Section 4.4 – Thermal-Hydraulic Design

Hydraulic Loads Analysis

- The applicant states that hydraulic loads on vessel components and fuel assemblies were evaluated but did not provide any description of the evaluation. In a January 29, 2009, response to RAI 134, Question 04.04-26, the applicant stated that the hydraulic loads were assessed and a reactor coolant pump over speed transient was identified as the limiting event. However, The applicant did not provide the results of that analysis for staff review.

- Staff Evaluation
  - Additional information is necessary regarding the hydraulic loads analysis

- Open Item
  - RAI 325, Question No. 04.04-61 requests a description of the hydraulic loads analysis for the U.S. EPR fuel and vessel components for normal operating conditions and design basis accident conditions

4.4.6 Instrumentation: Operating Experience

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Instrumentation

- Co-59 SPNDs have been used in Konvoi plants, Maine Yankee, and Palo Verde
  - But these old implementations do not provide instantaneous scram
- AMS has ~30 years of operating experience in 12 Konvoi plants
  - Extension to EPR involves only including additional ball stacks and detectors.

Setpoint Methodology

- AMS-based Calibration methodology and SPNDs protective functions were evaluated in the ANP-10287P SER and was found to be acceptable.

4.4.6 Instrumentation: Open Items

RAI 308, Question 04.04-60 (Open Item):

- Methodology to remove the ^{60}Co activation background from SPND measurements and how will this background will be treated by the protection system and the AMS

RAI 308, Question 04.04-59 (Open Item):

- Test plan to verify the accuracy of the correction algorithms that are applied to raw AMS activation measurements, including delays, activation buildup, and detector dead time

4.4.6 Instrumentation: Conclusions – AMS and SPNDs

- The mechanical design and functionality of SPND and AMS systems is similar to operating plants, and they satisfy applicable GDC criteria.

- All vessel penetrations have the proper Seismic Category I classification

4.4.6 Instrumentation: Conclusions Setpoint Methodology (AMS and SPNDs)

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Implementation of Setpoint Methodology

- The staff has reviewed and evaluated the setpoint methodology in the SER on ANP-10287P, "Incore Trip Setpoint and Transient Methodology for the U.S. EPR Topical Report."

- Staff Evaluation
  - The staff has determined that additional information is necessary in the FSAR to adequately address the implementation of the methodology described in ANP-10287P.

- Open Item
  - RAI 367, Question No. 04.06-14 requests an explanation on how the methods described in ANP-10287P will be implemented and verified for the U.S. EPR design

Control Rod Drive System Structural Materials

- FSAR
  - In response to an RAI, the applicant provided a description of fabrication of the CRDM pressure housing, including a sketch of the CRDM housing showing weld locations. The sketch reveals that components are made from stainless steel grades TP347, F347, and F6NM. The staff noted that FSAR Tier 2, Table 5.2-2 does not list grade F347.

- RAI 343, Questions 04.05.01-6
  - FSAR Table 5.2-2 should be modified to list material specifications and grades for all CRDM pressure boundary components.

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Control Rod Drive System Structural Materials

- FSAR
  - The RCPB materials specified in FSAR Tier 2, Table 5.2-2 for the CRDM pressure housing lists martensitic stainless steel materials

    SA-182 Grade F6NM (UNS S41500), and SA-479 (UNS S41500).
  - The applicant stated in response to an RAI that in August 2008, a request was submitted to ASME to extend the properties currently provided in Section II Part D for SA-182 Grade F6NM (UNS

    S41500) to SA-479 (UNS S41500).
- RAI 343, Questions 04.05.01-7
  - Tracks the applicant's submitted request to ASME Code to extend the properties currently provided in Section II Part D for SA-182 Grade F6NM (UNS S41500) to SA-479 (UNS S41500) material.

Control Rod Drive System Structural Materials

- Materials
  - ♦ The CRDM RCPB materials specifications and grades meet ASME Code Section III requirements for Class 1 components except as noted in Open Item 04.05.01-7.
  - ♦ Non-pressure boundary components are ordered to DIN, RCC-M and SAE/AMS specifications with AREVA special ordering requirements.
    - Materials ordered to foreign material specifications with AREVA special ordering requirements are essentially the same as comparable ASME/ASTM materials.
    - Selected materials have over 30 years of satisfactory operating experience

Control Rod Drive System Structural Materials

- Materials (con't)
  - CRDM Pressure Housing (RCPB)
    - Grade 347 material is solution annealed. Type 415 material is quenched and tempered. Weld procedure qualification will include hardness testing and corrosion testing to ensure that completed welds are not susceptible to SCC or hydrogen cracking.
  - CRDM Non-pressure boundary components
    - Type 347 (niobium stabilized) and Type 316Ti (titanium stabilized) austenitic stainless steels. Will include corrosion testing.
    - Other materials include Type 410 martensitic stainless steel (quenched an tempered), cobalt-nickel-chromium-tungsten alloy Haynes 25 and nickel based alloy X-750 (solution annealed-thermally aged).

Control Rod Drive System Structural Materials

- Cleaning and Cleanliness
  - Abrasive work such as grinding, polishing and wire brushing is controlled to prevent contamination.
  - Cleanliness of CRDMs is controlled during manufacture and installation in accordance with RG 1.37 and ASME NQA-1-1994
- Materials Compatibility with Reactor Coolant
  - Materials selected have performed satisfactorily in currently operating plants

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Material Specifications

- FSAR
  - Stellite 6 or equivalent is used for hardfacing
    - Radial Key Inserts
    - upper core guide pins and inserts.
- RAI Response
  - Stellite 6 filler material specifications are ASME SFA 5.21, Classifications ERCCoCr-A and ERCoCr-A.
- Open Item RAI 339, Question 04.05.02-9
  - FSAR, Table 4.5-2 should include the applicable ASME Code filler material specification for hardfacing

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Other Material Considerations

- FSAR – no information on degradation mechanisms
- RAI Response
  - ♦ Upon screening an augmented ASME Code, Section XI inspection program would be developed to verify IASCC and void swelling do not impact the safety function of these components.
  - ♦ The heavy reflector uses vertical keys and keyways in the forged heavy reflector slabs.
- Open Items RAI 339, Questions 04.05.02-10 and 11
  - ♦ FSAR should provide:
    - The screening evaluation or a COL Action Item to provide the screening evaluation
    - Prevention of notches on the vertical keys and keyways that can act as stress concentrations and crack initiation sites, which could lead to the loss of function of the heavy reflector.

Technical Topics of Interest
Section 4.6 – Functional Design of Reactivity Control Systems

Credit Taken for Boron Addition via the SIS

- FSAR Tier 2, Section 4.6.4 states that in the safety analyses in FSAR Tier 2, Chapter 15, except for the large break loss of coolant accident, no credit is taken for reactivity control systems other than reactor trip to mitigate the events to achieve a stable plant condition. The staff notes that in FSAR Tier 2, Section 15.1.5 appears to indicate that boron addition via the SIS is credited to mitigate large steam line breaks from hot zero power conditions.

- Staff Evaluation
  - Additional information is necessary to resolve this apparent discrepancy

- Open Item
  - RAI 366, Question No. 04.06-13 requests an explanation of the apparent discrepancy between FSAR Tier 2, Sections 15.1.5 and 4.6.4 regarding credit taken for boron addition via the SIS

Acronyms

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- AMS – Aeroball Measurement System
- ASME – American Society of Mechanical Engineers
- COL – combined license
- CRDM – Control Rod Drive Mechanism
- EFPY – Effective Full Power Year
- EPRI – Electric Power Research Institute
- FSAR – Final Safety Analysis Report
- IASCC – Irradiation-Assisted Stress-Corrosion Cracking
- LOCA – Loss of Coolant Accident
- MRP – Materials Reliability Program
- RCPB – Reactor Coolant Pressure Boundary
- SER – Safety Evaluation Report
- SIS – Safety Injection System
- SPND – Self Powered Neutron Detector
- RAI – request for additional information

Presentation to ACRS
U.S. EPR Subcommittee
Design Certification Application
FSAR Tier 2 Chapter 5

Dennis Newton, Supervisor - RCS System Design
Dale Matthews, Supervisor - RCS Component Design

AREVA NP, March 3, 2010

EPR™ by AREVA

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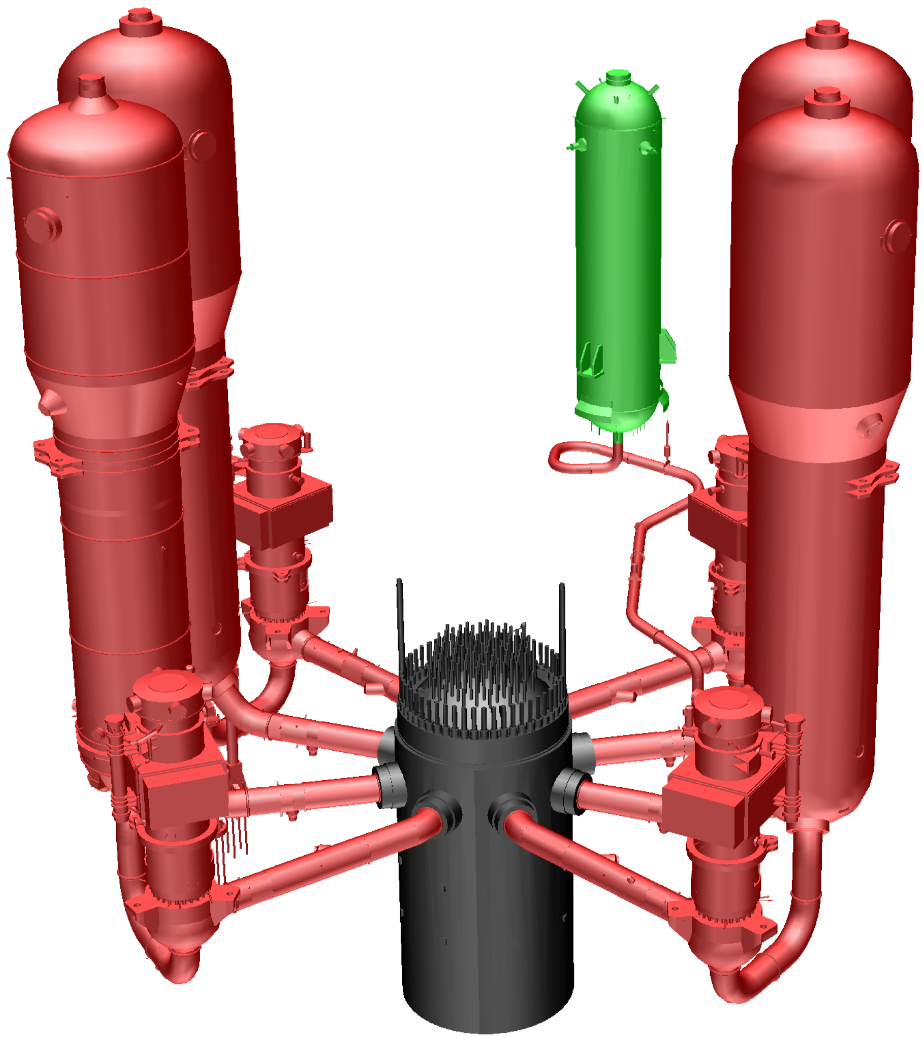
Chapter 5 Reactor Coolant System and Connected Systems: Section Topics

- **Chapter 5.0 Reactor Coolant System and Connected Systems**
- **Section 5.1 – Summary Description**
- **Section 5.2 – Integrity of Reactor Coolant Pressure Boundary**
- **Section 5.3 – Reactor Vessel**
- **Section 5.4 – Components and Subsystems**

5.1- Summary Description
U.S. EPR Design

- Typical U.S. four loop pressurized water reactor with four U-tube steam generators and 4 reactor coolant pumps/motors with shaft seals and flywheel
- Design features to improve safety and operation

5.1- Summary Description
RCS Component Arrangement



5.1- Summary Description Design Features to Improve Safety and Operation

- Reactor coolant pump shaft seal isolation for station blackout (SBO)
- No reactor pressure vessel (RPV) lower head penetrations
- Pressurizer safety relief valves (PSRV) provide overpressure protection at power and at low temperature
- Continuous pressurizer venting to minimize non-condensable gases in steam space
- PSRV loop seal internal to pressurizer to minimize hydraulic loads
- Primary depressurization valves depressurize RCS to prevent high pressure core melt ejection
- Steam generator (SG) with axial economizer to enhance heat transfer
- Heavy reflector to reduce neutron flux and associated shift in NDT

5.1- Summary Description
RCS Performance Data

Parameter	Value
Core Power	4590 MWt
RCP Power	24 MWt /pump
Thermal Hydraulic Flow	119,692 gpm/loop
Best Estimate Flow	124,741 gpm/loop
Mechanical Flow	134,662 gpm/loop
Design Pressure	2535 psig
Operating Pressure	2250 psia
Operating Temperature hot leg	624.6°F
Operating Temperature cold leg	563.4°F
Design Temperature	664°F (pressurizer 684°F)

5.2 - Integrity of the Reactor Coolant Pressure Boundary
5.2.1- Compliance with Codes and Code Cases

- ASME III B&PV Code of record is 2004 edition (no addenda)
- Most RCPB components are designed and fabricated as ASME III Class 1 components
- RCPB components are designed and fabricated as ASME III class 2 components if meet exclusion requirements of 10 CFR 50.55a(c)
- Five ASME code cases used
  - Code cases listed in U.S. FSAR Tier 2 Table 5.2-1
  - A COL applicant may identify additional ASME Code Cases to be used

5.2 - Integrity of the Reactor Coolant Pressure Boundary
5.2.2 - Overpressure Protection

- **Overpressure protection at power**
  - **Provided by PSRVs (spring pilot)**
  - **Keep RCS pressure less than 110 percent design pressure**
- **Overpressure protection at low temperature (LTOP)**
  - **Provided by PSRVs (solenoid pilot)**
  - **Keep RCS pressure less than RPV P-T limit**
  - **LTOP methodology, ANP-10283P, "U.S. EPR Pressure Temperature Limits Methodology for RCS Heatup and Cooldown"**
  - **LTOP design conforms to BTP 5-2 "Overpressurization Protection of Pressurized-Water Reactors While Operating At Low Temperatures"**

5.2 - Integrity of the Reactor Coolant Pressure Boundary
5.2.3 - Reactor Coolant Pressure Boundary Materials

- **RCS pressure boundary materials selection**
  - **Meet ASME code rules for Class 1 components and applicable Regulatory Guides**
  - **RCS components generally fabricated from low alloy steel forgings clad with austenitic stainless steel or Ni-Cr-Fe.**
    - Comply with 10 CFR 50 Appendix G
    - Comply with RG 1.43
    - Sulfur content limited to .008%
    - Additional controls on P, Cu, and Ni for RPV belt line to reduce sensitivity to neutron embrittlement
    - RTndt $\leq$ -4°F
  - **Austenitic Stainless Steel**
    - Comply with RG 1.44
    - Limited cobalt (.05% max) and sulfur (.02%)
  - **Ni-Cr-Fe Alloys (Alloy 690)**
    - Controlled chemistry and mechanical properties
    - Thermally treated after solution annealing
    - Controlled fabrication practices to limit residual cold work
  - **Material selections address experience with operating plants**
    - No Alloy 600 or 82/182 filler material is used
    - No cold worked stainess steel for wetted pressure boundary parts

5.2 - Integrity of the Reactor Coolant Pressure Boundary
5.2.3 - Reactor Coolant Pressure Boundary Materials

Fabrication

- **Complies with ASME Section III**
- **Welding procedures and personnel qualified per Sections III and IX**
  - Welding procedures conform to RG 1.50 and Nonmandatory Appendix D
  - Welders qualified to RG 1.71 when applicable
  - Cladding processes qualified per RG 1.43
- **Weld repairs to wetted surfaces limited**
- **Sensitization of stainless steel prevented by:**
  - Carbon content controls
  - Delta ferrite controls
  - Utilization of solution annealed and rapidly cooled stainless steel
- **Cleanliness maintained during all phases of fabrication**

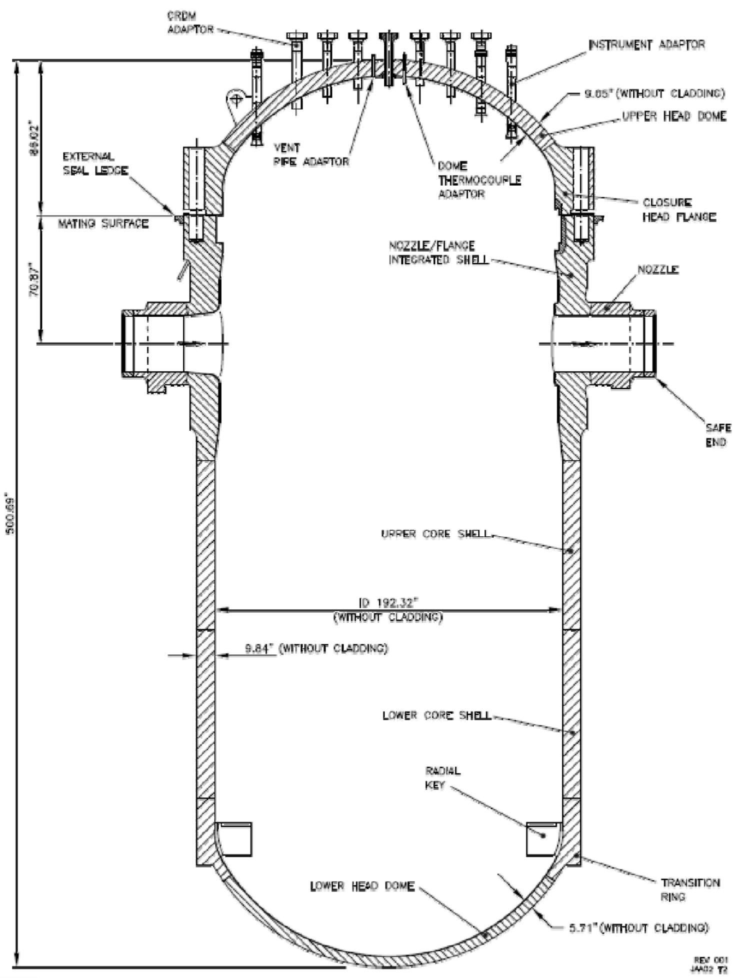
5.2 - Integrity of the Reactor Coolant Pressure Boundary
5.2.4 - Inservice Inspection and Testing of RCBP

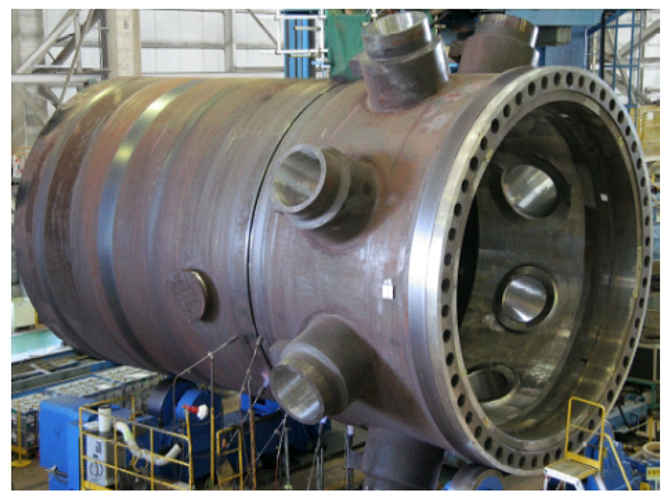
- **Components and piping designed to allow required inservice inspections and examinations**
  - **Section XI**
  - **Code Case N-729-1**
  - **Review of components performed with NDE personnel to assure inspectability using current technology**
- **Number of welds requiring examination minimized by material selection and fabrication techniques**
  - **Reactor vessel inlet/outlet nozzles are integral to the upper shell forging**
  - **Longitudinal seams in vessels and piping are eliminated**
  - **Main coolant piping machined or formed by bending to minimize contruction and installation welds.**
- **A COL applicant that references the U.S. EPR design certification will identify the implementation milestones for the site-specific ASME Section XI preservice and inservice inspection program for the RCPB, consistent with the requirements of 10 CFR 50.55a (g).**

5.2 - Integrity of the Reactor Coolant Pressure Boundary
5.2.5 - RCPB Leakage Detection

- **Methods used to detect and monitor unidentified leakage inside containment :**
  - **Containment sump level and discharge flow monitoring**
  - **Containment atmosphere radiation monitoring**
  - **Containment air cooler condensate monitoring**
- **RCPB leakage detection methods conform to RG 1.45 Rev. 1, "Guidance on Monitoring and Responding to RCS Leakage"**

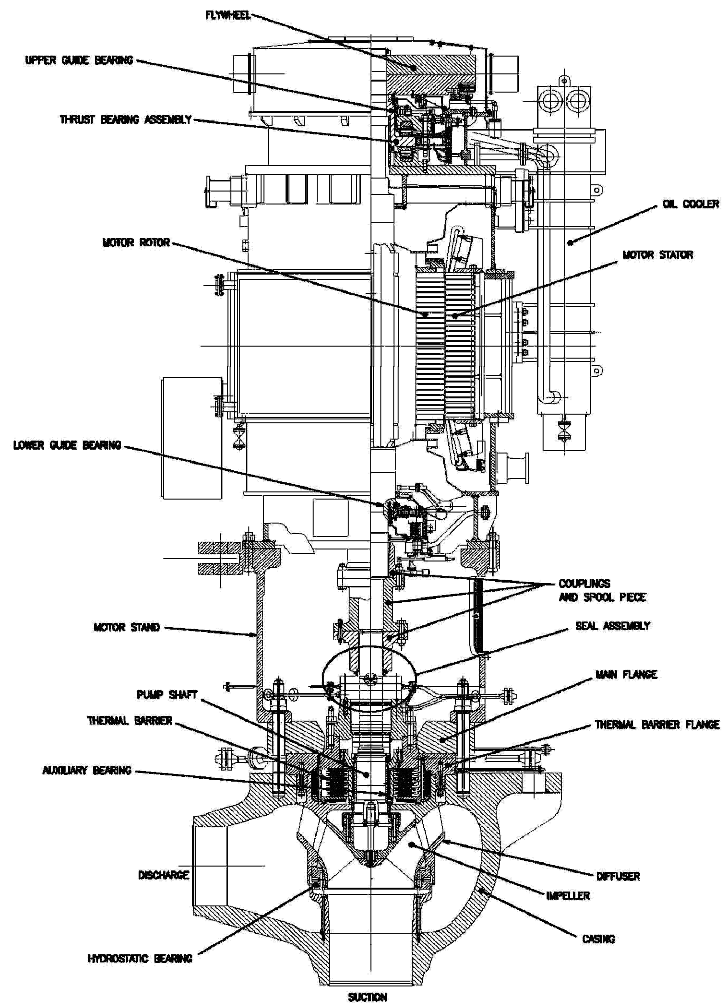
5.3 – Reactor Vessel





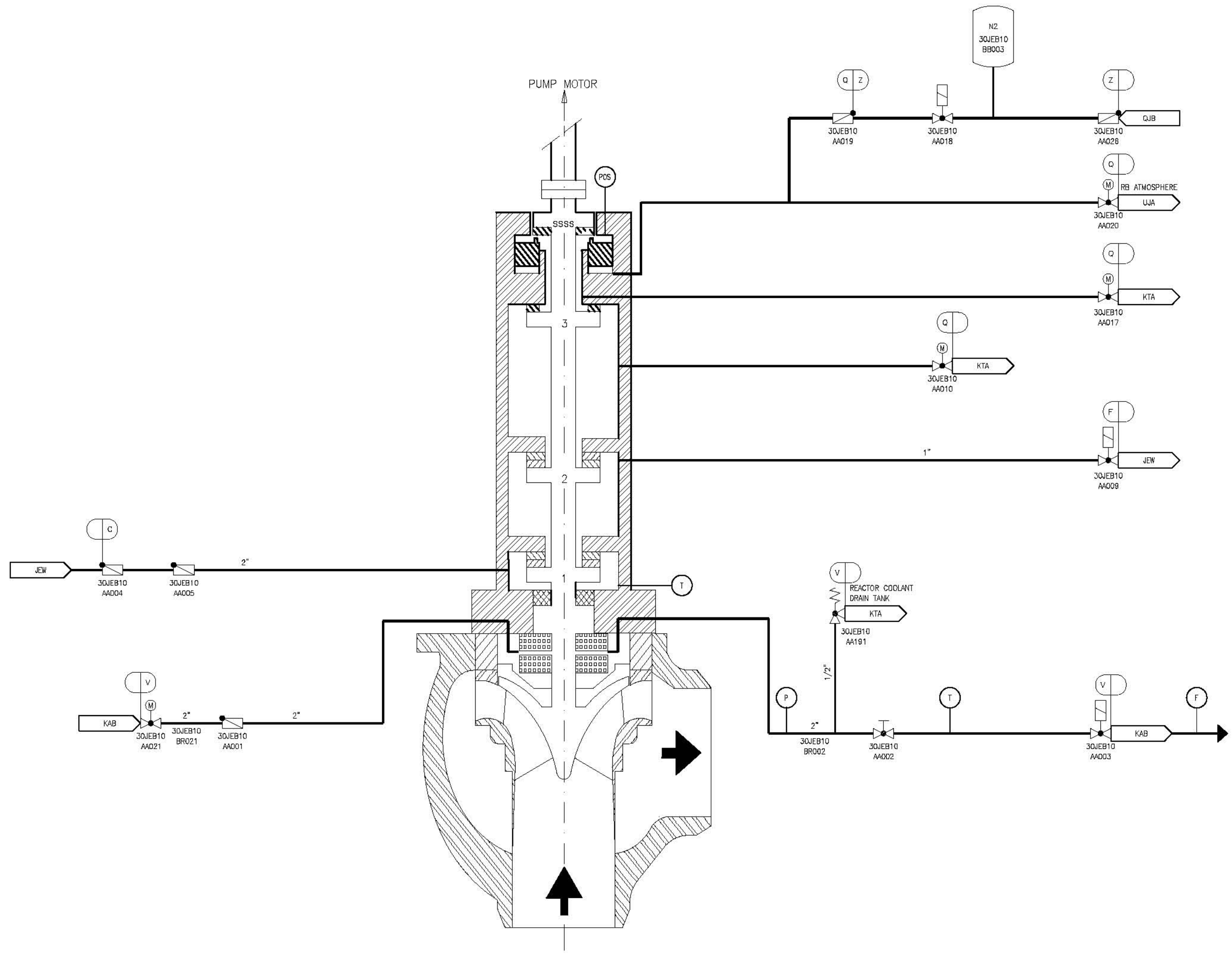
5.4 - Component and Subsystem Design
5.4.1 - Reactor Coolant Pump/Motor

- **Flywheel**
  - **Conforms to RG 1.14 “RCP Flywheel Integrity”**
  - **ANP-10294 “U.S. EPR RCP Motor Flywheel Structural Analysis Report”**
- **Oil collection system**
  - **Consistent with RG 1.189 “Fire Protection of Nuclear Power Plants**

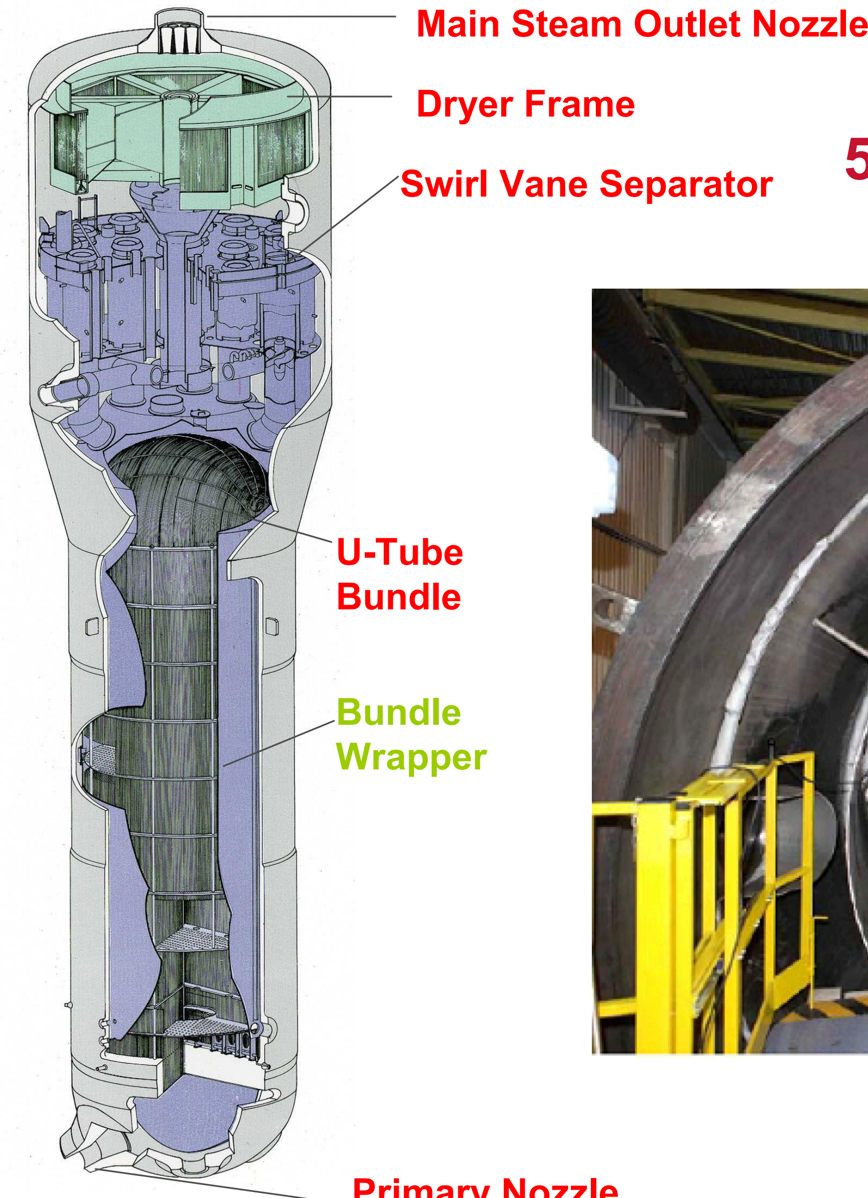


5.4 - Component and Subsystem Design
5.4.1 - Reactor Coolant Pump/Motor

- **Isolate shaft leakage during SBO**
  - **Close Stand Still Seal with gas pressure**
  - **Isolate seal return line**
  - **Isolate seal leak-off lines**
- **Isolate thermal barrier leakage**
  - **Detected by high pressure and flow**
  - **Isolate CCW to thermal barrier**



5.4 - Component and Subsystem Design
5.4.2 - Steam Generators



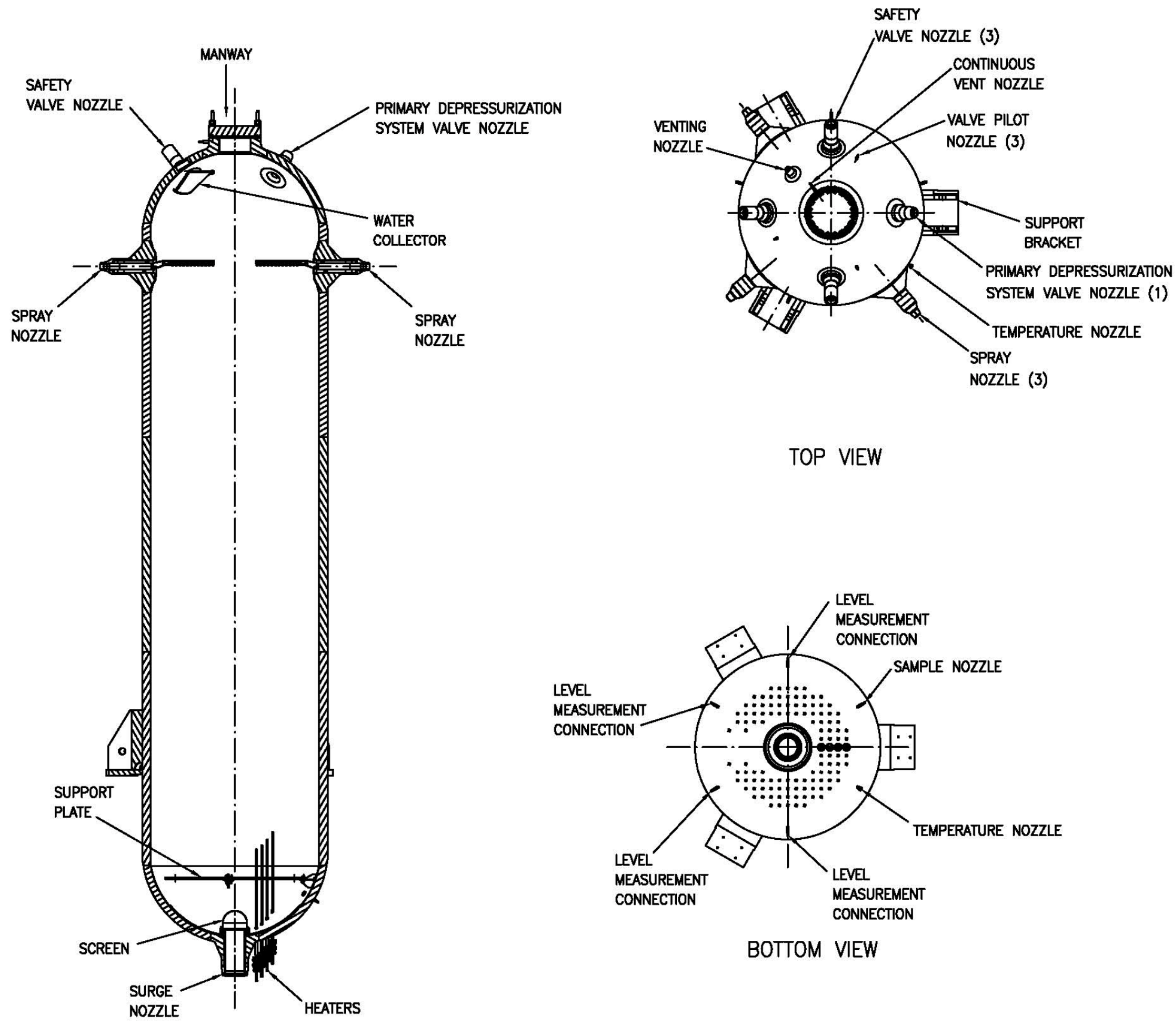
5.4 - Component and Subsystem Design
5.4.3 - Reactor Coolant Piping

- Forged austenitic stainless steel
- Bend pipes to reduce welds
- Large Nozzles forged with pipe to reduce welds
- Small nozzles have machined internal orifices

5.4 - Component and Subsystem Design
5.4.7 - Residual Heat Removal System

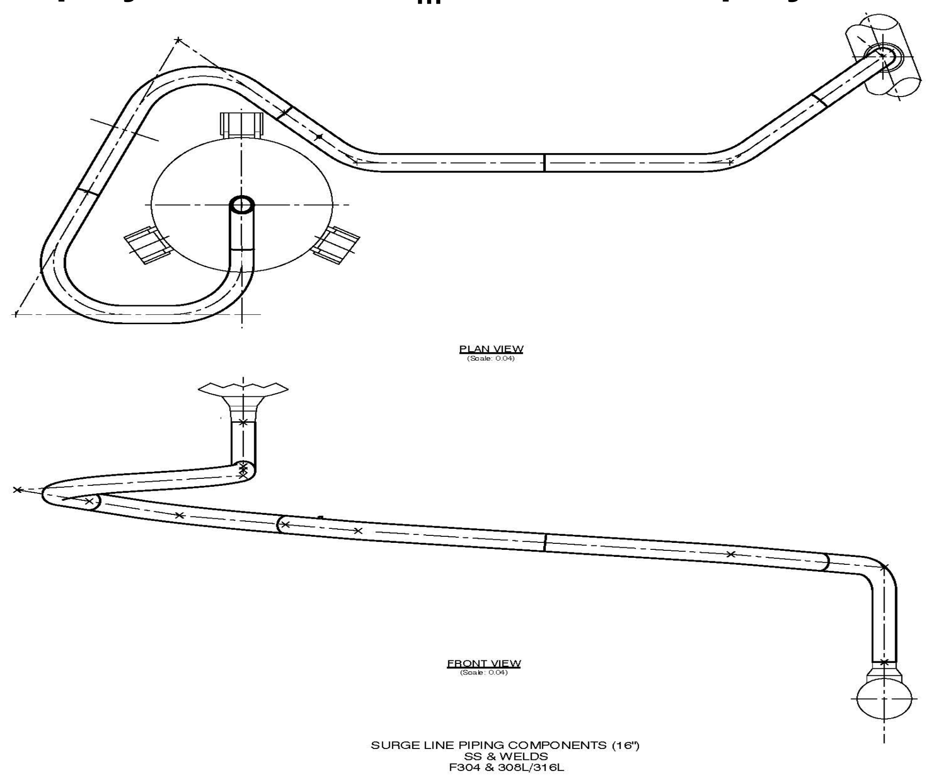
- Residual Heat Removal System has 4 physically separated and independently powered RHR trains
- Conforms to BTP 5-4 “Design Requirements of the Residual Heat Removal System”
- Self venting to prevent gas voids in piping
- Performance
  - Normal cooldown 250°F to 131°F (refueling operations) 15 hours
  - Safety grade 2 train cooldown to cold shutdown, 250°F to 200°F in 2.7 hours

5.4 - Component and Subsystem Design
5.4.10 – Pressurizer Arrangement



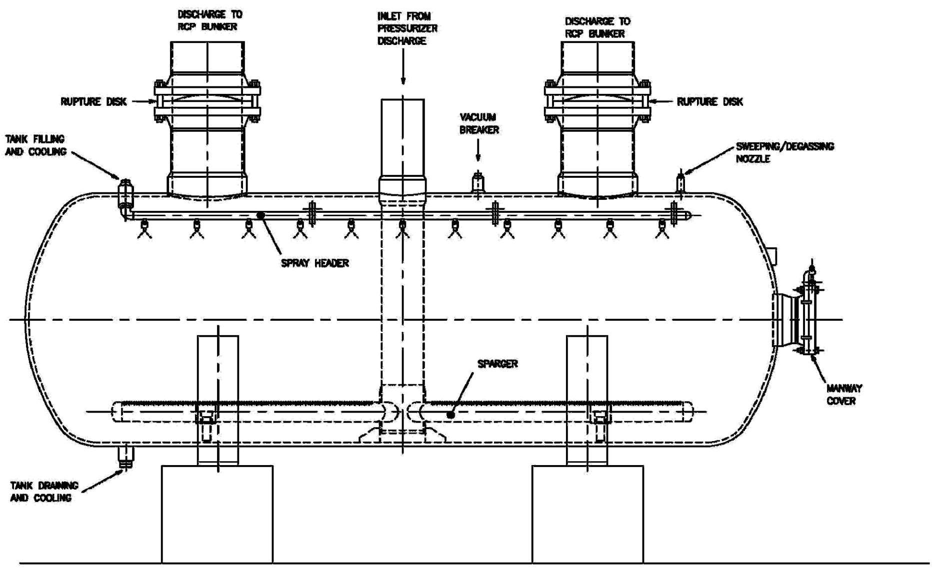
5.4 - Component and Subsystem Design
5.4.10 – Pressurizer Surge Line

- **Surge line has continuous sloping and vertical section**
  - **Designed to minimize thermal stratification**
  - **Continuous spray flow of .77 lb_m/sec in each spray line**



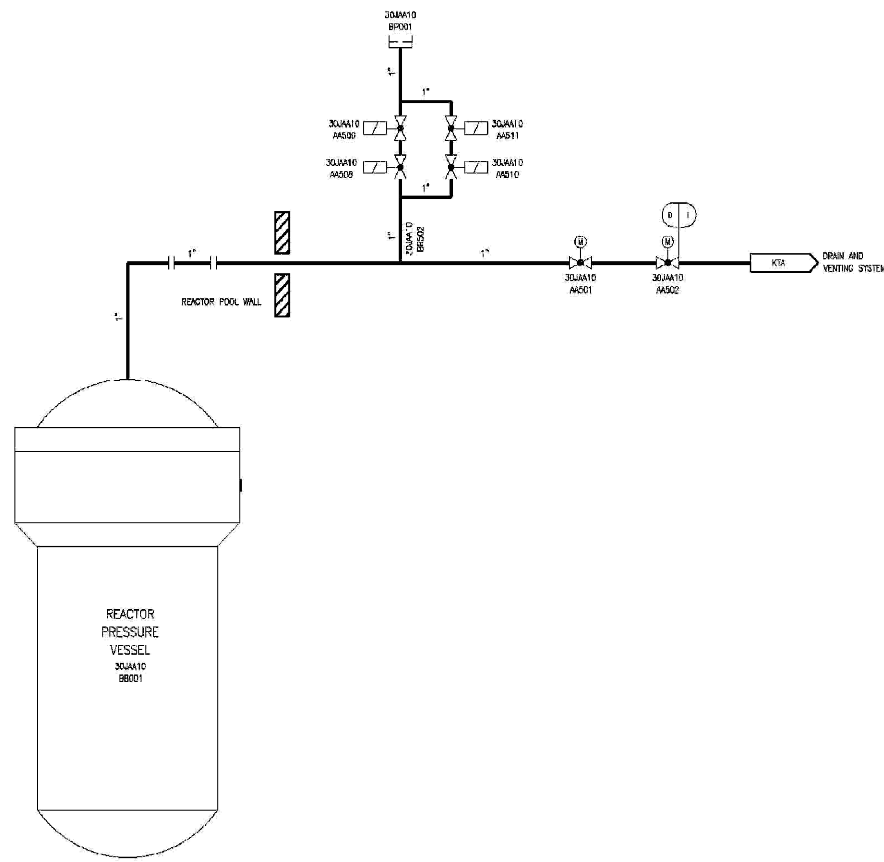
5.4 - Component and Subsystem Design
5.4.11 - Pressurizer Relief Tank

- **Collects, monitors and condenses steam discharged from the pressurizer**
- **Accommodate worst case anticipated operational occurrence - pressurizer discharge (turbine trip)**
- **Rupture disks routed to RCP room floor**
- **Allows for testing of PSRVs**



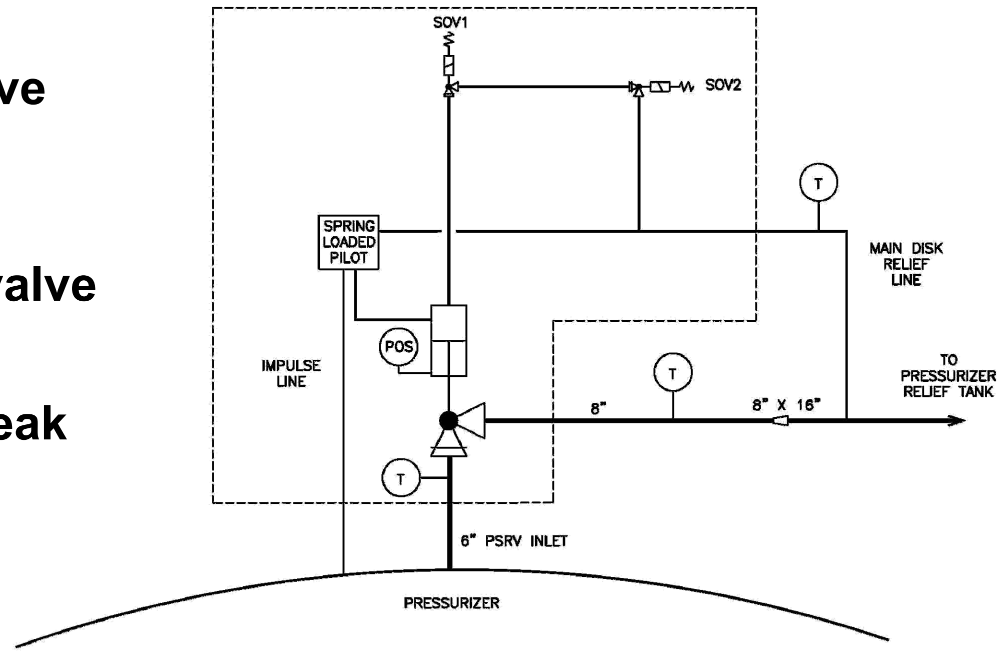
5.4 - Component and Subsystem Design
5.4.12 - RCS High Point Vents

- Remove non-condensable gases from RPV
- Conforms to 10 CFR 50.46
- Restricting orifice on discharge to limit flow to one CVCS pump
- Two parallel flow with two valve in series
  - Spurious opening does not cause leak path
  - Single failure does not prevent opening a path
- Positive valve position indication
- Separate power sources



5.4 - Component and Subsystem Design
5.4.13 - Pressurizer Safety Relief Valves

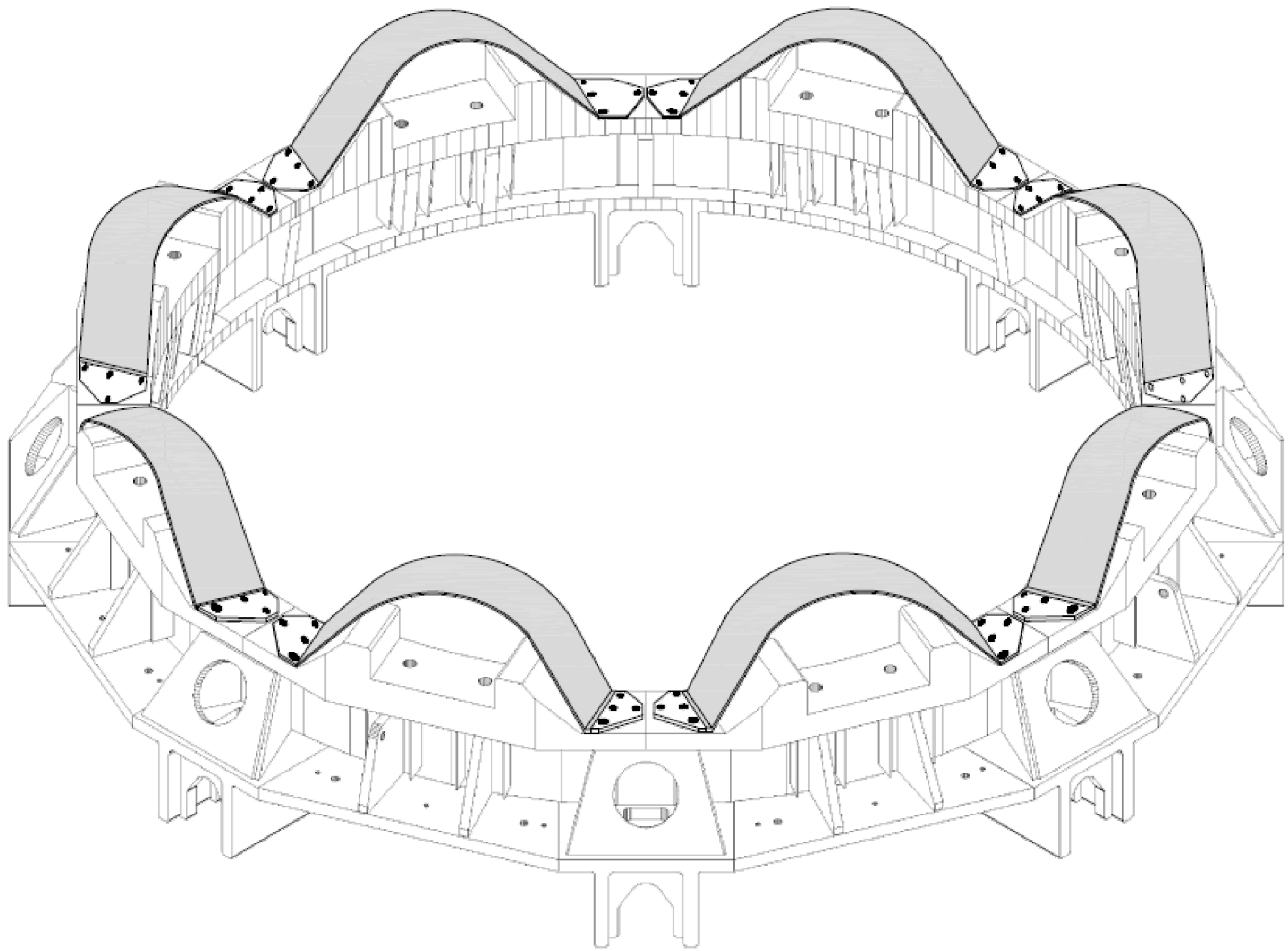
- Main relief disk with position indicator
- 1 spring operated pilot valve for at-power overpressure protection
- 2 solenoid operated pilot valve in series for LTOP
- Temperature sensors for leak monitoring



5.4 - Component and Subsystem Design
5.4.14 - Component Supports

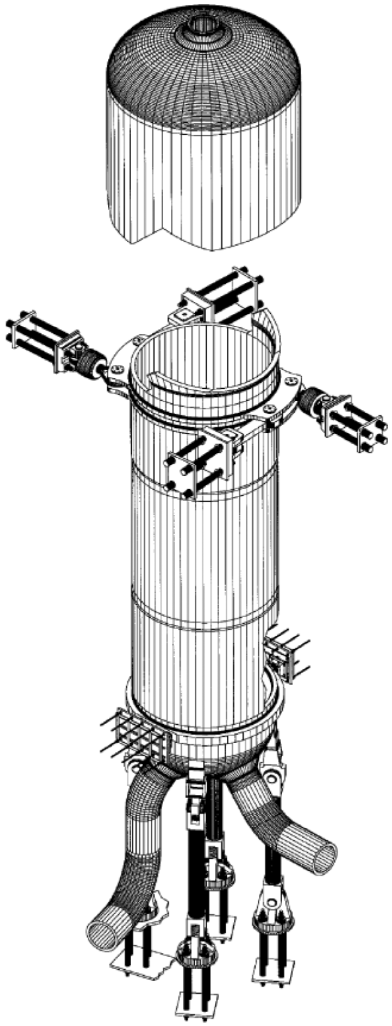
- **Designed to ASME Subsection NF**
- **Provide support and restraint of the primary components**
- **Maintain ability of the components to perform their safety related functions**
- **Affixed to Reactor Building with embedded bolts**

5.4 - Component and Subsystem Design
5.4.14 - Component Supports

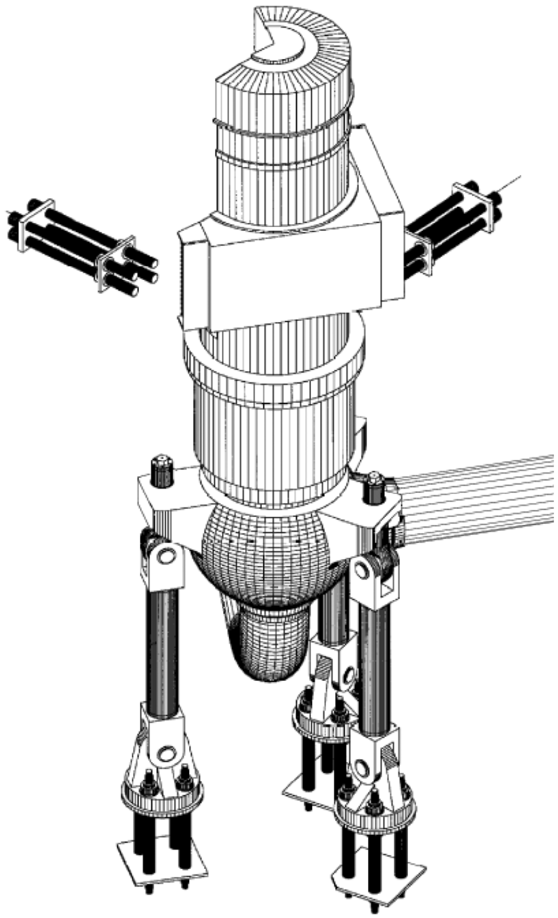


Reactor Vessel Support Ring

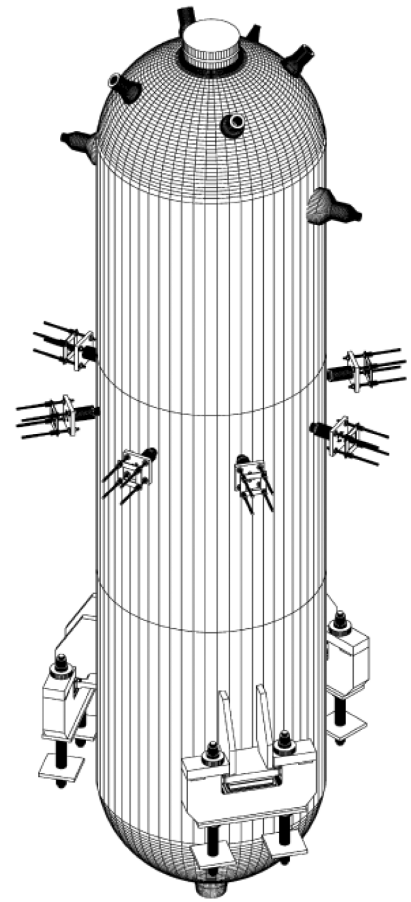
5.4 - Component and Subsystem Design
5.4.14 - Component Supports



Steam Generator



Reactor Coolant Pump



Pressurizer

Abbreviations and Acronyms

LTOP	Protection at Low Temperature
PSRV	Pressurizer Safety Relief Valves
RCP	Reactor Coolant Pump
RCS	Reactor System
RPV	Reactor Pressure Vessel
SBO	Station Blackout

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Presentation to the ACRS Subcommittee

AREVA EPR Design Certification Application Review

Safety Evaluation Report with Open Items

Chapter 5: Reactor Coolant System and Connected Systems

March 3, 2010

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Overview of DCA

SRP Section/Application Section		No. of Questions	Status Number of OI
5.2	Section Title Integrity of the Reactor Coolant Pressure Boundary	51	12
5.3	Section Title Reactor Vessel	27	8
5.4	Section Title Component and Subsystem Design	49	5
Totals		127	25

Description of Open Items

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- **RAI 365, Question 05.02.01.01-5**: Provide the technical basis of how 10 CFR 50.55a(b)(1)(ii) is addressed while using the 2004 Edition and 1993 Addenda
- **RAI 332, Question 05.02.02-11**: Perform LTOP analyses for the U.S. EPR design
- **RAI 332, Question 05.02.02-12**: Provide a reference or discussion on the dynamic loading analysis of the loop seals during the discharge of the safety valves
- **RAI 88, Question 05.02.03-1**: Will remain an open item until ASME issues the applicant’s Code Case, and the staff reviews its acceptability
- **RAI 278, Question 05.02.03-20**: FSAR Tier 2, Table 5.2-2 does not list weld filler material for RCPB piping, steam generators or pressurizer.
- **RAI 278, Question 05.02.03-21:** Modify the FSAR to provide limit on ferrite in Mo bearing cast austenitic stainless steels to address thermal aging embrittlement

Description of Open Items

- RAI 278, Question 05.02.03-23: Modify the FSAR by: (1) Discussing those components used in the solution annealed and rapidly cooled condition, and (2) discussing those components that will be used in the solution annealed and rapidly cooled condition followed by post weld heat treatment after welding. In addition, state in the FSAR that non-sensitization of safe-ends will be verified in accordance with RG 1.44
- RAI 278, Question 05.02.03-22: Address the inconsistency between the FSAR, which references stabilizing heat treatments, and the applicant's response to RAI 199, Question 05.02.03-17, which indicates that Grade 347 material used to fabricate the CRDM will not receive a stabilizing heat treatment, and modify the FSAR accordingly.
- RAI 88, Question 05.02.03-14: Tracks the open review of the applicant's responses to the RAI and other staff RAIs pertaining to ITAAC
- RAI 365, Question 05.02.05-9: Identify a COL information item specifically for the procedures relating to the conversion of instrument indicators and alarm setpoint

Description of Open Items

- RAI 365, Question 05.02.05-10: Identify a COL information item specifically for the procedures relating to operator actions to manage the long-term low-level RCS leakage
- RAI 365, Question 05.02.05-11: The verification of the RCPB leakage detection sensitivity, response time, and alarm limits for the RCPB leakage detection instrument was not included in the proposed ITAAC
- RAI 365, Question 05.03.01-14: Specify the minimum thickness of the cladding when qualified as weld buttering
- RAI 365, Question 05.03.01-15: Confirm that a low-heat-input weld process will be used for sequences where the radial-key attachment welds are made without subsequent post-weld heat treatment
- RAI 357, Question 05.03.01-13: Discuss operating experience with the use of bolted connections that attach the RVSP specimen guide baskets to the outside of the core barrel and how these bolted connections will maintain their structural integrity for the life of the plant for anticipated degradation mechanisms. ANP-10283, Revision 1

Description of Open Items

- RAI 64, Question 05.03.02-5: Tracks the open review ANP-10283P, Revision 1
- RAI 278, Question 05.03.02-7: Provide a table of the data points (reactor coolant temperature vs. pressure) for each P-T curve displayed in Technical Report ANP-10283, Revision 1
- RAI 278, Question 05.03.02-8: Clarify the thickness value (including vessel thickness and cladding thickness) used to calculate the fluence at the 1/4t and 3/4t locations for all materials provided in Technical Report ANP-10283, Revision 1
- RAI 278, Question 05.03.02-9: Provide all values (i.e., chemistry factors, fluence factors, margins, ΔRTNDT, etc.) used to calculate the ART at the 1/4t and 3/4t locations for all applicable materials provided in Technical Report ANP-10283, Revision 1
- RAI 278, Question 05.03.02-10: Address PTLR Criterion 4 (GL 96-03) and clearly identify both the limiting adjusted reference temperature (ART) values and limiting materials at the 1/4t and 3/4 t locations (t= vessel thickness) used in the development of the P-T limits

Description of Open Items

- **RAI 341, Question 05.04.01.01-2: Requests that FSAR Tier 2, Section 5.4.1.6.5, “Preservice Inspection,” specify that the surface and volumetric examinations will be performed after the spin test so that any flaws that have initiated or grown during the spin test can be detected.**
- **RAI 341, Question 05.04.01.01-3: The material specification should be included in FSAR Tier 2, Section 5.4.1.6 to ensure that the analysis in AREVA Report No. ANP-10294P, Revision 1, bounds the material that will be used for the thrust runner**
- **RAI 341, Question 05.04.01.01-4: Specify ITAAC in FSAR Tier 1, Table 2.2.1-5, Chapter 2, for performing this test to ensure that the flywheel assembly can withstand a design overspeed condition and preclude the generation of missiles**
- **RAI 364, Question 05.04.02.02-16: Revise the TS to clarify the requirement to inspect 100 percent of the tubes in newly installed original and replacement steam generators during the first refueling outage following installation.**
- **RAI 342, Question 05.04.12-5: Address the deficiencies in the FSAR and RAI responses related to the unidentified makeup systems and/or the CVCS makeup in the event the high point vent system failed to open.**

Technical Topics of Interest Section 5.2.1.1 – Compliance with Codes and Standards

- U.S. EPR code of record for design certification is 2004 Edition (no addenda) of ASME BPV Code with exception that the 1993 Addenda to 1992 Edition is used for seismic design of piping in compliance with 10 CFR 50.55a(b)(1)(iii).

- If a different Code edition or addenda is planned, the COL applicant must identify the edition and addenda in its COL application for NRC staff review and approval.

- **RAI 365, Question 05.02.01.01-5**: Provide technical basis of how 10 CFR 50.55a(b)(1)(ii) is addressed while using 2004 Edition and 1993 Addenda to 1992 Edition since these Code editions and addenda are not accepted by 50.55a(b)(1)(ii) for the weld leg dimensions.

Technical Topics of Interest
Section 5.2.2 – Overpressure Protection

<u>Summary of Open Items</u>

- RAI 332, Question 05.02.02-3
  - applicability to the U.S. EPR of the methodology, identified in BAW-10169P-A, “B&W Safety Analysis Methodology for Recirculating Steam Generator Plants,” B&W Fuel Company, October 1989, for LTOP analyses

- RAI 332, Question 05.02.02-12
  - Inlets to the PRSVs and PDS valves have water loop seals and analysis of the dynamic loads developed by the loop seals is required. The staff evaluated loop seal dynamic loading extensively in NUREG-0737 and found that the dynamic loads caused by loop seals were very significant. The RAI requests that the applicant provide a reference or discussion on the dynamic loading analysis of the loop seals during the discharge of the safety valves

Technical Topics of Interest
Section 5.2.2 – Overpressure Protection (Continued)

- There are no COL Action Items
- Design confirmed by initial test program and Technical Specifications

Technical Topics of Interest
Section 5.2.3 – Reactor Coolant Pressure Boundary Materials

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Materials specifications:

- With the exception of SA-479 (UNS S41500) used in the CRDMs, all RCPB materials specifications meet ASME Section III requirements
- Ferritic materials meet fracture toughness requirements of ASME Code, Section III

Technical Topics of Interest
Section 5.2.3 – Reactor Coolant Pressure Boundary Materials (Con'd)

Compatibility of Materials with Reactor Coolant:

- All RCPB materials exposed to reactor coolant are stainless steel, nickel base alloys, or ferritic materials clad with stainless steel and nickel based alloys.
- For PWSCC resistance, Alloy 690 is used in lieu of Alloy 600. Dissimilar metal welds use Alloys 52/52M/152
- Unstabilized austenitic stainless steels are low carbon (≤0.03% C). Materials and processing conform to the guidance in RG 1.44.
- RCS Chemistry evaluated under SER section 9.3.4
- Guidance in RG 1.36 is followed to control leachable contaminants in thermal insulation

Dissimilar Metal Welds (DMWs)

- Primary DMWs
  - RPV, pressurizer and steam generator nozzle to safe-end welds. Low-alloy steel to austenitic stainless steel.
  - Alloy 690 CRDM adapter tubes to RPV Closure Head
  - Alloy 52/52M/152 weld metal.

Technical Topics of Interest Section 5.2.3 – Reactor Coolant Pressure Boundary Materials (Con’d)

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Dissimilar Metal Welds (DMWs) (cont)

- Safe-end welding
  - No buttering
  - GTAW Welded in rotating fixture (flat position welding)
  - Narrow groove joint design
- CRDM Tube to RPV Closure Head
  - Partial penetration J-groove weld
  - Supplemental testing for weld materials

Technical Topics of Interest
Section 5.2.3 – Reactor Coolant Pressure Boundary Materials (Con'd)

Open Items

- 05.02.03-1 Code Case to include SA-479 (UNS S41500)
- 05.02.03-14 ITAAC
- 05.02.03-20 Table 5.2-4 does not list weld filler materials for RCPB piping, steam generators or pressurizer.
- 05.02.03-21 Provide ferrite limits for Mo bearing cast austenitic stainless steels
- 05.02.03-22 Address inconsistency between RAI response and FSAR regarding stabilizing heat treatment of Grade 347 stainless steel
- 05.02.03-23 Discuss PWHT of austenitic stainless steel safe-ends and corrosion testing.

Technical Topics of Interest Section 5.2.4 – Inservice Inspection of the Reactor Coolant Pressure Boundary

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- The staff found that the operational program complies with the requirements of ASME Code Section XI and 10 CFR 50.55a for inservice inspection of the reactor coolant pressure boundary:
  - Design enables the performance of inservice examinations by eliminating interferences due to design, geometry, and materials of construction
  - Personnel, equipment and procedures used to perform examinations are qualified in accordance with the requirements of IWA-2300, and mandatory ASME Appendices VII and VIII
- The staff found that the US EPR DCD met the acceptance criteria adequate to meet the above requirements which include :
  - Regulatory Guide (RG) 1.26, as it relates to the quality group classification of components
  - RG 1.147, as it relates to ASME Section XI Code Cases acceptable for use
  - ASME Code Case N-729-1, as modified by the Final Rule under 10 CFR 50.55a(g)(6)(ii)(D) for reactor vessel head inspection guidance
  - NRC Bulletin 88-05, as it relates to the establishment of a program to detect and correct potential RCPB corrosion caused by boric acid
- No open items related to the Inservice Inspection operational program.

GDC 30, "Quality of Reactor Coolant Pressure Boundary"

- Open Items
  - RAI 365, Question 05.02.05-9: FSAR should include a new COL information item to instruct the COL applicant to develop procedures to convert instrument indicators to a common leakage equivalent and to provide alarm setpoints for low-level leakage operator actions.
  - RAI 365, Question 05.02.05-10: FSAR should include a new COL information item to instruct the COL applicant to develop procedures relating to operator actions to manage the long-term low-level RCS leakage.
  - RAI 365, Question 05.02.05-11: ITAAC should be revised to include verification of the RCPB leakage detection sensitivity, response time, and alarm limits.

Technical Topics of Interest Section 5.3.1 – Reactor Vessel Materials

- Materials meet NRC requirements.
  - RV is alloy steel (SA-508).
  - RV cladding is stainless steel (308L/309L) or nickel alloy (52/52M/152).
- RV Welding Issues (Open Items)
  - RAI 365 05.03.01-14: Specify minimum thickness of nickel alloy buttering.
  - RAI 365 05.03.01-15: Confirm that low-heat-input weld processes are used for attachment welds without subsequent post weld heat treatment.
- RVSP meets NRC requirements. But guide baskets are bolted to core barrel which is not a standard attachment method.
  - RAI 357 05.03.01-13: Discuss operating experience and structural integrity issues with bolted connections in this application (Open Item).

Technical Topics of Interest
Section 5.3.2 – P-T Limits, Upper-Shelf Energy, and PTS

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Pressure -Temperature Limits

- AREVA addressed submittal of P-T limits by providing a Pressure-Temperature Limits Report (PTLR)
  - ♦ PTLR
    - Follows guidelines of GL 96-03
    - Contains bounding P-T limits and complete methodology
  - ♦ COL Information Item 5.3-2
    - A COL applicant that references the U.S. EPR design certification will provide plant-specific pressure and temperature limits using approved PTLR methodology

OPEN ITEM: All open items in this section are associated with review and approval of U.S EPR PTLR

- OPEN ITEM 05.03.02-1: Tracks the open review of the U.S EPR PTLR (ANP-10283P, Revision 1)
- RAI 278, Question 05.03.02-7
- RAI 278, Question 05.03.02-8
- RAI 278, Question 05.03.02-9
- RAI 278, Question 05.03.02-10

Inspection

- FSAR
  - ♦ Pre-service inspection is performed prior to being placed into service.

- Open Items RAI 341, Questions 05.04.01.01-2 and 4
  - ♦ FSAR should provide:
    - An ITAAC in FSAR Tier 1, Table 2.2.1-5, Chapter 2, for performing a spin test to ensure that the flywheel assembly can withstand a design overspeed (125 percent of normal operating speed) condition.
    - Specify that the surface and volumetric examinations will be performed after the spin test so that any flaws that have initiated or grown during the spin test can be detected.

Flywheel Design

- Flywheel Analysis (AREVA Report No. ANP-10294P, Revision 1)
  - Evaluates the critical speed due to excessive deformation for the flywheel based on the collar and thrust runner.
- Open Items RAI 341, Questions 05.04.01.01-3
  - FSAR should provide:
    - specify the material used for the thrust runner.
      - ensures the analysis in the AREVA report bounds the material to be used for the U.S. EPR thrust runner.

Technical Topics of Interest Section 5.4.2.1 – Steam Generator Materials

- Materials and design features meet NRC requirements
- Essentially the same as replacement SGs at U.S. plants
  - Thermally treated Alloy 690 tubes (typical diam., wall thickness)
  - Stainless steel support materials
  - Tube support plate holes designed for good flow
  - Alloy 690 divider plate and tubesheet cladding
- Full-depth hydraulic expansion of tubes in tubesheet
- Access for cleaning, inspection, foreign object search/removal
- Primary and secondary water chemistry conform to the EPRI guidelines
- COL Item 5.4-1: ASME Code edition for SG inspection
- No open items

Technical Topics of Interest Section 5.4.2.2 – Steam Generator Program

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- SG Program meets NRC requirements
- Based on NEI 97-06 for structural and leakage integrity through EPRI guidelines
- Tube repair criterion conforms to the guidance in RG 1.121
- Conforms to the standard technical specifications (STS) where applicable (STS do not address initial inspection for new plants)
- One open item (RAI 364, Question 05.04.02.02-16) : revise technical specifications to clarify the requirement to inspect 100% of the tubes during the first refueling outage following initial SG installation and replacement

Technical Topics of Interest Section 5.4.7 – Residual Heat Removal (RHR) System

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- RAI 212,Question 06.03-6
  - Protection of the LHSI pumps against cavitation damage is provided by an automatic stop upon detection of low loop water level or low Δp_{sat} (difference between RCS hot leg temperature and hot leg saturation temperature). An evaluation of LHSI net positive suction head (NPSH) during DBAs is provided in FSAR Tier 2, Section 6.3.3, "Performance Evaluation." The evaluation includes consideration of IRWST water temperature, suction sump screen blockage, and uncertainty in hydraulic resistances, and concludes there is sufficient NPSH during DBAs. The staff has not completed review of the applicant's response to a request for additional information issued under Section 6.3 of this report (RAI 212, Question 06.03-6) regarding this issue. This issue will be addressed in Section 6.3 of this report.

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Technical Topics of Interest Section 5.4.11 – Pressurizer Relief Tank

- RAIs have been resolved
- There are no Open Items
- There are no COL Action Items
- The applicant's design meets the requirements of GDC 2 as it relates to protection against the effects of earthquakes. Failure of non-safety-related systems does not have any adverse effects on safety-related systems.
- The applicant's design meets the requirements of GDC 4 as it relates to the protection of safety related equipment from adverse environmental effects and from missiles generated by rupture disk failure. This criterion is met, because the system design prevents steam or water release to containment under any normal operating conditions or anticipated operational occurrences. In addition, the tank is orientated such that the rupture disks do not pose a missile hazard to safety-related equipment.

Technical Topics of Interest Section 5.4.12 – Reactor Coolant System High Point Vents

Open Item

- RAI 342, Question 05.04.12-5.
  - Additional information is needed to confirm that the CVCS can provide adequate makeup if the high point vent system fails open and to confirm that this failure would not be classified as a LOCA.
  - "The CVCS is not a safety system and is not required to supply reactor coolant makeup to the RCS in the event of small breaks or leaks in the RCPB. Also, the CVCS is not designed to perform the safety function of the ECCS during a DBA. Therefore, GDC 33 and GDC 35 are not applicable to the CVCS."
  - GDC 33 requires that a reactor coolant makeup system be provided for protection against small breaks in the RCPB. The applicant has not identified such a makeup system that complies with GDC 33. Until the applicant can adequately explain why GDC 33 is not applicable to the CVCS system and then relies on the CVCS charging pumps for protection against small breaks in the RCPB, the staff considers this an open item pending further information from the applicant.

Technical Topics of Interest Section 5.4.12 – Reactor Coolant System High Point Vents (Continued)

<u>Conclusions</u>

- Except for the open item discussed above, the staff concludes that the design of the RCS high point vents is acceptable and satisfies the guidance of SRP 5.4.12.

- This conclusion is based on the staff's determination that RCS high point vents include components and piping to remotely exhaust non-condensable gases from the primary coolant system and vent the gases to the containment atmosphere. The review included the applicant's proposed design criteria and design bases, and the results of the applicant's analyses of the vent system design. In addition, the basis for acceptance in the staff review is conformance of the applicant's designs, design criteria, and design bases, including resolution of the open items, for the RCS vents and supporting systems to applicable regulatory guides, branch technical positions, and industry standards.

Acronyms

ART- Adjusted Reference Temperature ASME – American Society of Mechanical Engineers
ASME-American Society of Mechanical Engineers
COL – Combined license
CRDM – Control rod drive Mechanism
CVCS- Chemical and volume control system
DBA- Design-basis accidents
DBE- Design-basis events
DMW- Dissimilar metal welds
ECCS- Emergency Core Cooling System
EPRI – Electric Power Research Institute
FMEA- Failure modes and effects analysis
FSAR – Final Safety Analysis Report
GTAW- Gas tungsten arc weld
ITAAC- Inspections, tests, analyses, and acceptance criteria
LOCA – Loss of coolant accident
LTOP- Low-temperature overpressure protection
MSRT-Main steam relief trains

Acronyms

MSSV-Main steam safety valves
NPSH-Net positive suction head
PDS-PZR depressurization system
PTLR-Pressure-temperature limits report
PWR-Pressurized water reactors
PZR-Pressurizer
PWSCC-Primary water stress-corrosion cracking
PWHT-Post-weld heat treatment
RAI – Request for additional information
RPV-Reactor Pressure Vessel
RVSP- Reactor vessel surveillance capsule program
RV- Reactor Vessel
RHRS- Residual Heat Removal System
RCPB- Reactor coolant pressure boundary
SER – Safety Evaluation Report
STS- storage tank systemtrains

Technical Topics of Interest
Section 5.2.3 – Reactor Coolant Pressure Boundary Materials DMWs

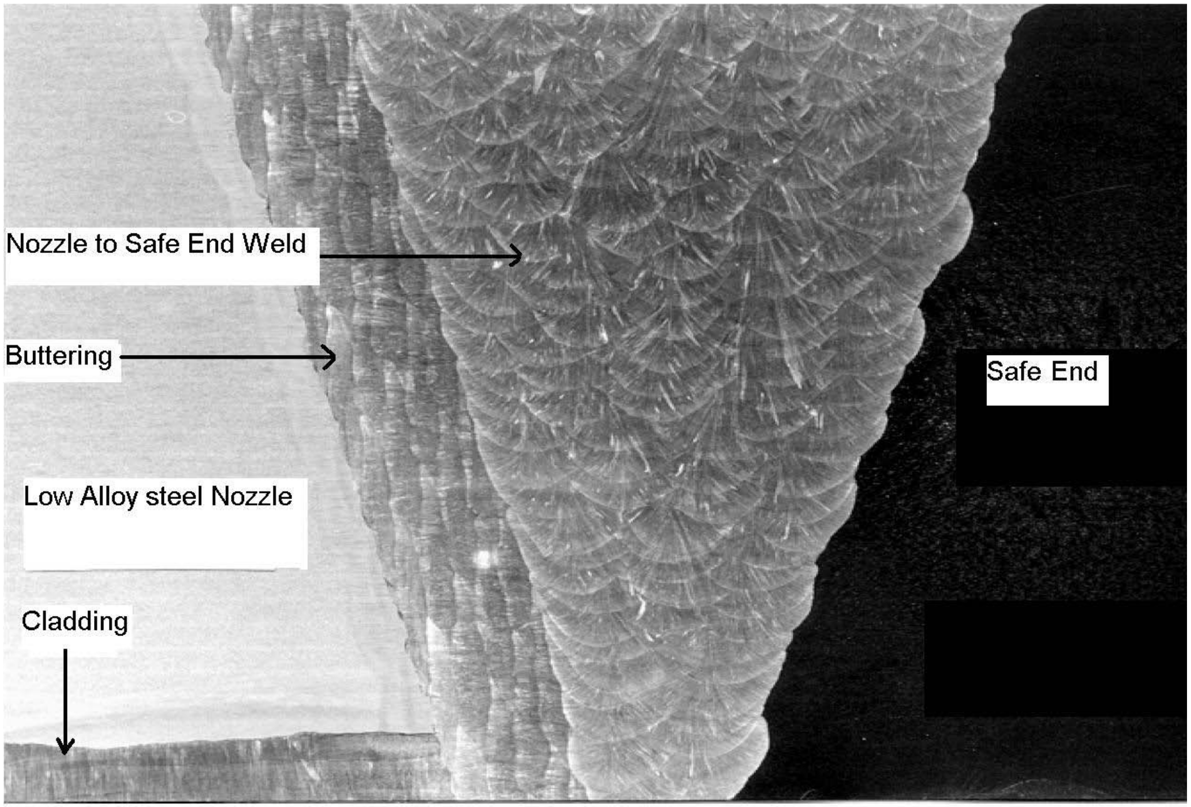

Figure 2. Typical DMW Weld Configuration With V-Groove Joint Design Using Buttering

Technical Topics of Interest
Section 5.2.3 – Reactor Coolant Pressure Boundary Materials DMWs

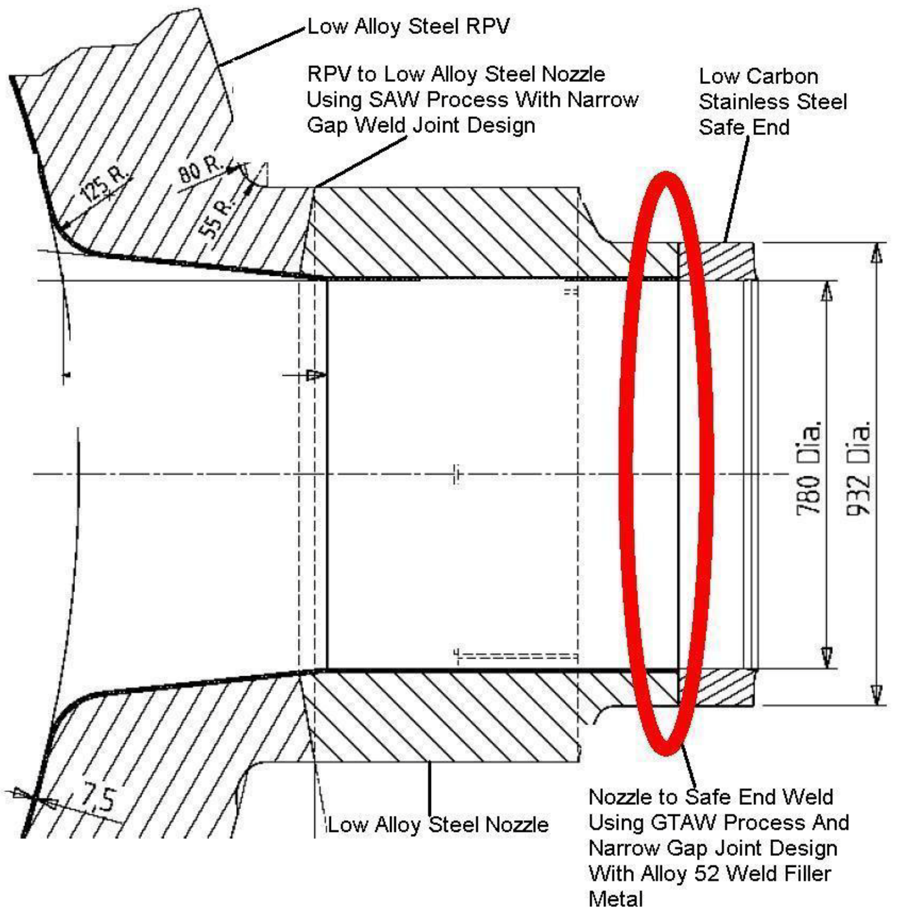

Figure 1. RPV Nozzle to Safe End Weld Configuration

Technical Topics of Interest
Section 5.2.3 – Reactor Coolant Pressure Boundary Materials DMWs

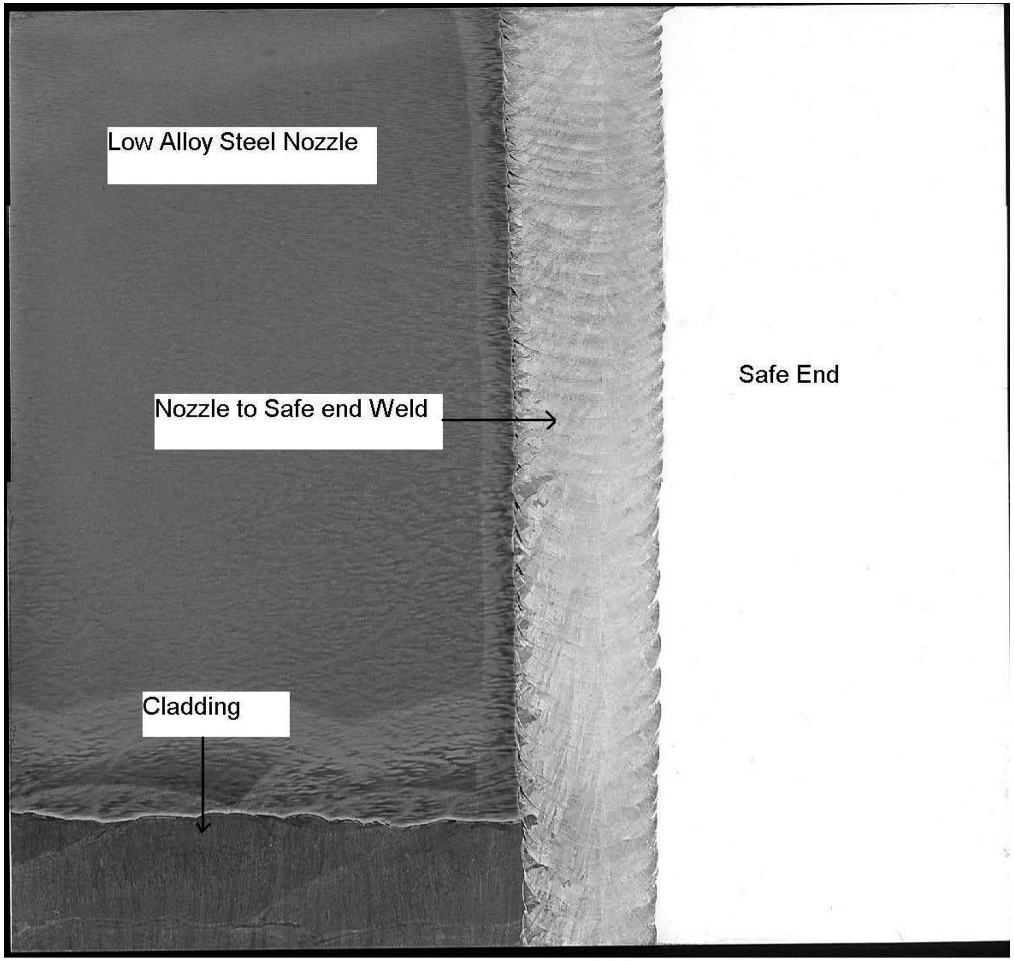

Figure 3. GTAW Narrow Gap DMW