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# OFFICIAL TRANSCRIPT OF PROCEEDINGS

# NUCLEAR REGULATORY COMMISSION

## ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

**Title:** MEETING: PLANT LICENSE RENEWAL

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**Work Order No.:** ASB-300-839

07

**LOCATION:** Rockville, MD

**DATE:** Wednesday, June 30, 1999

**PAGES:** 1 - 197

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

JUNE 30, 1999

The contents of this transcript of the proceeding of the United States Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, taken on June 30, 1999, as reported herein, is a record of the discussions recorded at the meeting held on the above date.

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

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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MEETNG: PLANT LICENSE RENEWAL

U.S. NRC

Two White Flint North

Room T2-B3

11545 Rockville Pike

Rockville, MD

Wednesday, June 30, 1999

The subcommittee met, pursuant to notice, at 8:30 a.m.

MEMBERS PRESENT:

MARIO BONACA, Chairman, ACRS

THOMAS S. KRESS, Member, ACRS

ROBERT SEALE, Member, ACRS

WILLIAM SHACK, Member, ACRS

ROBERT UHRIG, Member, ACRS

P R O C E E D I N G S

[8:30 a.m.]

CHAIRMAN BONACA: Good morning.

The meeting will now come to order.

This is a meeting of the ACRS Subcommittee on Plant License Renewal.

I am Mario Bonaca, Chairman of the subcommittee.

ACRS members in attendance are Dr. George Apostolakis -- actually, he's on his way, I guess -- Thomas Kress, Robert Seale, Bill Shack, and Robert Uhrig.

The purpose of this meeting is for the subcommittee to review the NRC staff's Safety Evaluation Report related to the Oconee license renewal application, crediting of existing programs, and related matters.

The subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate for deliberation by the full committee.

Mr. Noel Dudley is the cognizant ACRS staff engineer for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register on June 1, 1999.

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

It is requested that speakers first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We have received no written comments or requests for time to make oral statements from members of the public.

On June 16, 1999, the NRC staff completed the Safety Evaluation Report for the Oconee license application. This is the second Safety Evaluation Report for a license renewal application.

The report identifies only three items that must be resolved for the staff to complete the evaluation. The open items include the basis for excluding specific structures and components from an aging management review, applicability of certain aging effects to structures and components, and the need for additional periodic inspections.

The Safety Evaluation Report also identifies six confirmatory items that involve documentation of certain information or commitments.

The ACRS plans to review and comment on the Safety Evaluation Report at its September 1999 meeting.

On June 3, 1999, the staff issued a Commission paper identifying options for crediting existing programs for license renewal. ACRS plans to review and comment on crediting existing programs at its July 1999 meeting.

This is just one example of the license renewal policy issues that the staff is evaluating and that the subcommittee plans on considering.

We will now proceed with the meeting, and I call upon Mr. Christopher Grimes, Chief of the License Renewal and Standardization Branch, to begin.

MR. GRIMES: Thank you, Dr. Bonaca.

We're very pleased to be here today.

The NRC staff is prepared to respond to the committee's questions concerning both the basis for the staff's review of the Oconee license renewal application, and also, this is different from Calvert Cliffs to the extent that Duke Energy refers to B&W topical reports that establish generic bases for aging management programs, and so, we're also -- we have also arranged on the agenda to speak to the topical report reviews and to discuss the basis for the staff's evaluation of those reports, as well, and as you mentioned, we have designated time on the schedule after we've discussed Oconee to discuss the staff's paper on the generic issue associated with credit for existing programs, and we'll cover that topic when we've finished with the Oconee presentations.

Beyond that, we're here prepared to answer questions, and we've arranged for specific members of the NRC staff to make presentations on the material covered in all three of those areas.

Thank you.

CHAIRMAN BONACA: Thank you.

The Duke staff -- we have a presentation on the topical reports, actually the specific BAW-2251.

MR. ROBINSON: Good morning.

I am Greg Robinson. I am the Project Manager for Oconee license renewal, and on behalf of Duke and our Framatome Technology gentlemen here, I appreciate the opportunity to come share this information with you.

I'm going to take just a few minutes and give you an acclamation and overview of the project and how the topical reports fit into the Duke application. Then I'll turn it over to our Framatome colleagues, who will give you the details of the reactor vessel report.

Also, this afternoon, in your hand-out package that you have in front of you is the remainder of the presentation materials for the afternoon session. It will be a short session that we will cover, and we put all the information in the one hand-out.

This morning, Mark Rinckel from Framatome will take the lead on a bulk of the reactor vessel material. Matt Devan is here, Ken Yoon is here, and Bob Gill will then give us an overview of how the owners group topical information fits into the application itself.

Paul Colaianni will cover the afternoon session for us.

A little background on Oconee: Oconee Nuclear Station began construction in the late 1960s and completed construction in the early 1970s, a three-unit site, 2,538 mega-watts, initial capital cost around $500 million. Commercial operation began in 1973 for units one and two and 1974 for unit three. The initial licenses obviously expire 40 years later, in 2013 and '14, and about 1,300 people are employed on-site.

Here is an aerial of the Oconee site. It's set in northwestern South Carolina in the foothills of the mountains, on a peninsula out in the lake, Lake Keowee. So, you can see the three units there, and you're looking from the discharge out over the plant -- or, excuse me, the intake out over the plant.

Before Mark gets into the details of the owners group work, I thought it would be fair to show you just how long ago we began to work on the technical information that ultimately ended up in the application.

You can see, back in the mid-'80s, there were a number of technical reports, the lead plant work that you're all familiar with, a scientific perspective on aging and aging phenomena, a good bit of research going on both in the industry and by the NRC.

The focus was on aging mechanisms at that point in time.

The time-line here is meant to show you the progression in thought over the last 15 years, where you can see we've evolved from more of a scientific thought process into more of a practical engineering end point that we were ultimately able to use and put in the Oconee application, and I hope you'll be able to see that today.

I won't go through each of the areas. I'm sure you're very familiar with them.

I will point out that the Oconee efforts really began back in the same time period, in the mid-'80s, where we were a participant in the industry efforts and then, later, in the owners group efforts and ultimately got to the 1998 submittal in July of last year.

Current project status, just to acclimate us again here this morning -- I think you hit most of this in your introduction this morning -- safety responses to RAI's were completed, and the safety evaluation was issued just a few weeks ago.

The environmental area, the Draft Oconee Supplemental Environmental Impact Statement was issued the end of May.

There is a public meeting on that in the Clemson-Oconee area here next week, and then, in the hearings area, the NRC commission has affirmed ASOB's decision to deny the petition of our potential intervenor, and that was done in April of this year.

I showed you the time-line, the progression of thought over the last 15 years, and I thought it fair to give you another little rule of thumb as we get into the technical details of the vessel report.

One of the things that we began to notice when we put together the initial B&W's owners group reports was we were beginning to see a pattern emerge, and the pattern ended up fitting into this equation, and the pattern was, if we can define the component and its materials of construction, we understand where it's located in the plant, then we can understand the aging of that component, material, environment, stress conditions.

Then we can look and see if we have programmatic action that can manage that material/aging combination. If, for example, those programs had been in existence for a good long time, there ought to be demonstrable evidence that the programs work or they don't work or they've self-corrected, and all of that taken collectively gives us assurance that we have something that will continue to serve us on into the future.

That was written in many, many words in many, many technical reports.

What we did for ourselves is boil it out into this rule of thumb to give us the confidence that, each time, we could measure back against the standard, making sure that we had covered each of the aspects of this in our integrated plant assessment.

The other area that Mark will touch on with the vessel and that we certainly touched on completely in the application was the time limit that aging analysis, the boundary conditions on the initial design that we had to investigate.

Begin to progress toward the owners group topicals and how they fit into our application.

We divided the application work, the development work, into five areas. We covered the reactor coolant system, which is where the B&W owners group topicals fit, as a separate area, for a couple of reasons.

One, it was an important area of focus for us. It demanded a lot of additional attention, we felt, and also, that is where the owners group work fit back in. So, when the match line between the owners group work and the Duke work -- we wanted to be very clear that we didn't miss something. So, we delineated that area.

The reactor containment was another area that we felt needed special attention.

Radiological line of defense -- we broke it out as a separate area from the remainder of the structures, so that we could study it. Then the other three areas were the classical discipline areas -- mechanical, electrical, and structural.

So, today, we're here to focus on the reactor coolant system and, more specifically, on the reactor vessel.

Here are the reactor coolant system components, just to give you a feel for how they break down. You'll see the piping, pressurizer vessel, and reactor internals and, beside them, some small notation.

Those were the technical -- or, excuse me, topical reports from the owners group that we submitted to the staff for approval over time. They absolutely equal the information that's over in our application for the piping pressurizer vessel and internals.

We also developed through the owners group additional information for the remainder of the components. We did not submit that for approval, but we did use it in the Oconee application.

You'll see there's a safety evaluation for piping, the pressurizer, the reactor vessel, and a draft safety evaluation recently issued for the reactor internals, and today, we're here to give you the details of BAW-2251, and with that, I'll turn it over to Mark Rinckel from Framatome, who will give us the details.

MR. RINCKEL: Good morning.

My name is Mark Rinckel. I'm from Framatome Technologies. I've been the project engineer, generic license renewal project engineer since approximately 1993. I have Matthew Devan here, who is an expert in our materials area, on our surveillance program, and Ken Yoon to assist me in the fracture mechanics area.

So, I will proceed to give you a summary of 2251, and the topics I'd like to go over today are who the participating plants were in the report, the contents of the reactor vessel report, basically how it's divided into the various chapters of our report, the scope, which tells about the component, the aging effects, how we came upon the aging effects for the reactor vessel, the demonstration of aging management, which would be the programs that we credit for those aging effects, and the time-limited aging analyses, which in my mind are really the crux of the reactor vessel report, because it deals with all the reactor vessel embrittlement issues. We'll finish it up with the conclusions.

Now, the participants in our program include ANO unit one, Oconee units one, two, and three, and TMI unit one. Crystal River unit three and Davis-Besse unit one were not participants in our generic program in the reactor vessel report.

All of our plants are similar in design. They're 177 fuel assembly lowered-loop plants, and all of the operating licenses expire somewhere between 2013 and 2016.

So, because of the similarity in the design and construction, it certainly lends itself to generic report treatment.

Now, as Greg had mentioned before, the basic formula that we follow in almost all of our report was establishing an RCS piping report, and I saw Sam Lee here earlier, and he was instrumental in helping in the iteration process in developing how we go about doing these evaluations, and basically, the first thing we do is the first bullet, is we define the intended functions of the component, and for the reactor vessel, there are two intended functions, one of them being maintain RCS pressure boundary and the other one being to support the internals. We find that through going through our design specs, equipment specs as the designer. So, we define those two functions.

The next thing we do is to provide a description of the component, including materials of construction, and this was fun for me, because when most of these components were fabricated, I was in elementary school. So, I had to go back and understand the construction and see how all these things were put together, and the objective there is really to find -- you know, to define the component materials of construction and really go through the fabrication part, and that was a lot of fun for me, because you know, we seem to have lost some of that technology as time has gone on.

DR. KRESS: Did you have sufficient records that you could find the material?

MR. RINCKEL: Yeah, we did. We had -- all the QA data packages were in our records system, and then, when I got stumped, I'd go downstairs to the component engineers, who were in Mt. Vernon when these things were fabricated, and I'd ask them, and I found that they were usually the best source of information.

DR. KRESS: But QA is worth something.

MR. RINCKEL: Yes, it is, even back to the 1968-1970 timeframe.

So, that's really chapter two of our report, is providing the description of a component.

Chapter three of four report is to define the applicable aging effects, and again, we look at material construction, we look at operating environment, and we look at level A and B service conditions. Those are normal and upset conditions.

Those are the normal aging stresses of the component. We did not assume emergency and faulted conditions, as that is not a normal aging stresser.

So, the assessment of aging effects is very much qualitative in this whole process, and again, that whole process was established through our first report, which was the RCS piping report.

Once we've defined the aging effects for the component and the various items, then we look at the programs that manage those aging effects. One of the primary programs is ASME Section XI. There are other programs, forecast and wastage program, Matthew's surveillance program for reduction of fracture toughness and so forth. I will get into that in more detail a little bit later.

The last item is to evaluate the time limited again analyses, includes the upper shelf energy, lots of the reduction of fracture toughness in the belt-line region.

So, that's the basic outline for the report.

DR. KRESS: Was Oconee one of the plants that was used in the original pressurized thermal shocks?

MR. RINCKEL: I believe it was.

MR. YOON: Ken Yoon from Framatome Technologies.

In the initial 1980 period, one of three plants was Oconee 1, Oconee unit one.

MR. RINCKEL: All of the vessels within the scope of this report were designed in accordance with ASME Section III, 1965 edition, 67 addenda.

We have found it very convenient in our report to describe the various components in chapter two of our report, really in accordance with the ASME Section XI examination categories.

For instance, we would divide it into groups.

Examination category BA can include the reactor vessel shell enclosure head.

Reactor vessel nozzles would be examination category BD. That included the inlet-outlet nozzles, core flood nozzles, in-core monitoring system nozzles, and CRDM penetrations at the top of the vessel.

The reactor vessel interior attachments, examination category BN-1 -- those are the core guide lugs, and the last item would be pressure retaining closures, which would be the closure head and the CRDM closure at the top.

Now, the reactor vessel shell and closure head I'll point out here. These are all fabricated from low-alloy steel, either A508 class two forgings or they're A533, was a grade B, plate or a 302 plate. The closure head and the shell are about 14-foot inner diameter, 37-foot high. They're shown here.

These are all clad on the interior surface with Austin stainless steel. They were put in with a weld deposit submerged arc process, usually a two-wire or a six-wire process, which would be a high heat input process was used for cladding the interior surface of the vessel shell and the vessel head. That's the first item.

These shells were usually about six-foot sections that were welded using an automatic submerged arc process, using a Linde 80 flux weld wire that was coated with copper.

At the time of construction back in the late '60s, the copper was put on the weld wire to preclude rusting of the weld wire, and we didn't know at that time that it would result in accelerated reduction of fracture toughness.

So, many of our welds, most of the welds in the belt-line region, are Linde 80 welds that have some copper in them, a little bit more than they probably would without the coating, and therefore was the beginning of our surveillance program that Matthew will talk about later.

So, that's the shell and the closure head.

DR. SHACK: It's clad with stainless steel. There's 82-182 pads underneath the core guide lugs. Is that the only place that you have the 82-182 on the shell?

MR. RINCKEL: Yeah, that's right. These are alloy-600 guide lugs, and they're connected to the cladding with 82-182 weld material. So, that's the only place in the vessel where there's 82-182 weld material.

DR. SHACK: Now, are they welded to the cladding, or there's an 82-182 deposit on the shell and then they're welded to that?

MR. RINCKEL: I believe they're welded to the cladding, to the stainless cladding, but there is a stress evaluation done to show that -- I mean the purpose of those guide logs is to catch the internals, should there be a fracture of the core barrel up near the ledge, and so, it's designed to accept about a quarter-inch drop, and it's blended in with the cladding, but it is a structural weld that's able to withstand that weight.

DR. SHACK: Okay. So, it's not sustaining load most of the time.

MR. RINCKEL: There's nothing on it.

DR. SHACK: It's really just a catcher.

MR. RINCKEL: It's a catcher, and the purpose of that is, if the internals should drop down, is to catch it to prevent the internals from going down to the bottom of the vessel and therefore taking some of the control rods out of the active fuel region. That's the whole purpose of the guide logs.

DR. SHACK: Then the only other alloy-600 and 82-182 weldments would be at the penetrations for the instrumentation and the drives.

MR. RINCKEL: That's right. The nozzles up top, which would be the control rod CRDM nozzles, which would be these up here, are all alloy-600, and then the in-core monitoring system nozzles down at the bottom are alloy-600.

So, that's the inconel or alloy-600 that you have in the vessel.

DR. SHACK: Thank you.

DR. UHRIG: How thick is the wall, pressure vessel?

MR. RINCKEL: The shell region is about 8 1/2 inches in the belt-line, and then it increases to approximately 12 1/2 inches where the nozzles enter the vessel, and the head, the flanges are approximately 24 inches.

The heads -- the bottom head and the top head are, I think, about 4 1/2 to 5 inches thick, and those are made from plate, both the top and the bottom head are plate.

DR. UHRIG: So, the head is about 7 inches.

MR. RINCKEL: About 7 inches, yeah.

We have two outlet nozzles, 36-inch diameter, all clad with stainless steel. Those are forgings, 508 forgings.

We have four inlet nozzles that are 28-inch inner diameter, again 508 forgings clad with stainless steel, two core flood nozzles that are approximately 10-12 inch inner diameter, again 508 forgings clad with stainless steel, and then we just talked about the alloy-600 penetrations which are at the top and bottom of the vessel.

Now, the alloy -- I don't have this in the packet, but since you're interested in the alloy-600, we had problems with the original configuration of the in-core monitoring system pipes down at the bottom of the vessel.

They were three-quarter-inch Schedule 160 pipes that extended through the bottom head and met up with the internals package so that the in-core monitoring system would go up and through there.

In hot functional testing in Oconee unit one, those all broke off, and so, these pipes here, the pipe through the bottom would extend all the way up, and those all broke off right in that vicinity there, and what we had to do was make a reinforcement to increase the strength of this so that it would not break under the flow conditions at the bottom of the vessel, and those were all done after the -- again, after hot functional testing was completed at Oconee unit one.

That made it bigger, made it stronger, but that's really the only major problem that we have had with the vessel to date. We've had, really, very little problems.

DR. SHACK: Do you have cracking in your

instrumentation nozzles?

MR. RINCKEL: Not that we know of, and they do a visual inspection of those, VT-3, every interval, and to my knowledge, they have not seen any, and we have not had any leak at present, and of course, they are at the bottom of the vessel, so they are at about 550 degrees, which is a lower temperature and, therefore, less susceptible to PWSCC than the penetrations, probably, at the top, since temperature does play a factor in that, even though they would be susceptible to cracking by PWSCC.

DR. SHACK: You do a VT-3 on those, but in the license renewal application, you're going to do at least a one-time VT-1 enhanced?

MR. RINCKEL: That was not discussed in there, no. The only thing that we would -- that we committed to in our report was to continue the inspections that we would commit to as part of Generic Letter 97-01, and those included the closure head penetrations and not the bottom head penetrations, but the alloy-600 -- all of the alloy-600 in the loop is within the Oconee alloy-600 program, and that requires some additional looking for the most susceptible components.

So, Oconee took an approach where they looked at all of the alloy-600 items, and they said, okay, let's catalog these and find out which are the most susceptible to

PWSCC and then we will look at the top five locations. To my knowledge, that IMS nozzle did not come up as one of the top five locations.

DR. SHACK: That's the way you do it; you look at the limiting component --

MR. RINCKEL: That's right.

DR. SHACK: -- in the inspection.

MR. RINCKEL: Yes.

DR. SHACK: And what can you actually see with the VT-3?

MR. RINCKEL: Well, you can see if there's cracking there, not fine cracks, obviously, you'd have to have pretty good size cracks. I think you can see if there is any cladding missing, if there's any, perhaps, cracks big enough to extend to the base metal where you can see some rust or something there.

So, that is what you can see, and you only really do a VT-3 of the reactor vessel internals and the interior surfaces of the vessel itself.

Anyway, the other thing that we typically do is, based on the functions -- and I'll put this back -- we identify what items we will subject -- that will be subject to aging management review based on the functions that they -- whether they support an intended function, and there are a couple items that were sent with the vessel to the Oconee units that are not -- do not support an intended function, that are not subject to review.

One of them would be the monitoring pipes, which are there to detect leakage. These items don't support the pressure boundary and are not subject to review.

The other item that's not subject to review is the seal ledge on the outside. It does not support the pressure boundary function.

And the other items that were -- are subject to review that weren't in the scope of the report are the lower CRDM service support structure and the lower portion of the reactor vessel skirt.

Now, those items we simply chose not to include in the scope of the report, because we, in general, were consistent with the IWB inspection boundary. Those aren't inspected in accordance with IWB, so we simply didn't include them, and Oconee would then have to evaluate them in the plant-specific application.

So, that's what's in the report, what's not in the report, what's subject to review, what's not subject to review.

Once we have the component, the materials of construction, we look at the operating environment, the operating stresses, which are service levels A and B, and we determine the applicable aging effects, and again, it was easy for us to group them.

Examination category BA, which are reactor vessel shell enclosure head -- we looked cracking, where would cracking occur at welded joints, why would that be the case, growth of pre-service flaws, fatigue. Fatigue would be time-limited aging analysis.

The external surfaces of the shell enclosure head could be subject to loss of material, boric acid wastage, could have leakage at the closures, bolted closures. So, we looked at loss of material.

Reduction of fracture toughness in the belt-line region.

The last one, growth of inter-granular separations, and I'll get into growth of inter-granular separations with -- the easiest to show is a figure of it here.

That was a time-limited aging analysis. We found this when we went back to the early 1970s, when the components were fabricated and licensed. We found a fracture mechanics analysis that was done for this, and so, we had to evaluate it.

DR. SHACK: There's absolutely no consideration of stress corrosion cracking of the low-alloy steel.

MR. RINCKEL: That's correct. We did not do that, because there was no indication that that's occurred for any of the primary system components, and you also have cladding.

DR. SHACK: Okay. I guess that was my question. Is that because you felt that, in this environment, the material would -- the low-alloy steel, even if exposed, would be resistant --

MR. RINCKEL: Yes.

DR. SHACK: -- or you're simply relying on in the integrity of the cladding, that it will never get exposed?

MR. RINCKEL: I think both.

I mean, if you go back, the only aging effect that we said would crack welded joints would be to the pre-service flaws, and that is why you look at the joints now, is that those things may be there and they may grow over time, and so, we dismissed stress corrosion cracking of the low-alloy steel cladding, and even if it were exposed to borated water in this environment, we do not feel that stress corrosion cracking would be a mechanism, and plus, that's one thing I liked about the rule that changed, is that you talk about cracking and the mechanisms, and you know, we could argue a long time about those, but the fact that we have said we would -- it's possible that we would crack the welded joint, and what do you have there to look for?

So, the aging effects we looked at, again for the shell enclosure head, are listed there.

This figure shows the reactor vessel shell region in the welded joints for Oconee unit two, again cracking at the welded joints, but we had to look at reduction of fracture toughness and where on this shell reduction of fracture toughness would be applicable.

The traditional belt-line region -- and I'll show it to you in just a second -- is primarily the regions of the shell that are adjacent to the active fuel assemblies, and I'll show you right here.

So, that portion of the shell is the traditional belt-line region, and that includes the lower -- the intermediate shell and the lower shell and the welds that connect those shells together, the little portion of the nozzle belt region, which is a forging on the top, and then this region right down at the bottom here with the transition forging.

DR. SEALE: Those are all ring castings?

MR. RINCKEL: Those are all ring forgings, not castings.

DR. SEALE: I mean forgings.

MR. RINCKEL: Yes, sir.

Now, unit one is different from units two and three. Unit one has a plate that's -- two plates to make the cylinders, and that's 302 plate, and units two and three have 508 forgings.

DR. UHRIG: There is some longitudinal welds on unit one.

MR. RINCKEL: Unit one does have some axial welds, yes, sir.

Now, the belt-line region, as I said, is traditionally those regions that are just adjacent to the active fuel assemblies, and then the question for license renewal is would the belt-line region grow and would it, in fact, grow up and include some of the weld that includes -- that connects the nozzles to the nozzle belt region, and the nozzles are subjected to different loads than the shell, because it basically supports the weight of the piping.

So, you have discontinuities where the nozzle comes into the nozzle shell and also piping loads under design basis conditions.

We looked at estimating what the fluence would be up in that region, and it was above 1 times 10 to the 17th, which is the number that says that you need to have that type of material in the surveillance program. At present, we don't have that material, that specific material in our surveillance program.

So, what we did is we had Dr. Yoon do a fracture mechanics analysis for that particular region to see if, in fact, it was more limiting than the shell region of the belt-line. It was not.

Therefore, we could dismiss that region as not being limiting and therefore not within the belt-line region, and the classical belt-line definition for -- that we have used for 32 effective full-power years was also applicable to 48.

So, we were able to narrow the region where a reduction of fracture toughness was applicable to the classical belt-line region, where we are irradiating all of the materials up until fluences, well beyond what we would expect at 48 NPY.

DR. SHACK: Let me just understand that screening analysis.

You do the fracture mechanics analysis from a purely fracture mechanics point of view to get the loads regardless of the presumed toughness of the material, and you're saying that the loads just aren't as high there, or are you really making some assumption about toughnesses, also?

MR. RINCKEL: Well, we included the toughness, because we estimated what a fluence would be. The fluence up in that region was about an order of magnitude lower than the maximum in the belt-line region, and so, we did look at material degradation, reduction of toughness.

DR. SHACK: Okay. So, you don't have that material in your surveillance program, but you then make some reasonably conservative assumption about its loss of toughness.

MR. RINCKEL: Because it's really very similar to the Linde-80 welds and very similar to the weld material that was used in the belt-line region. I believe that's right.

Now, the only portion of the reactor vessel, the base metal, that would be subject to cracking would be the 508 forgings, class two forgings that were clad using the high-heat input process such as the submerged arc two-wire or six-wire, and all of the forgings in the scope of our report were clad using a six-wire process, and what's shown here in this figure are the two beads.

They had bead one, including the six wires, would be the first pass, and this was all clad. They put the forgings on a machine and turned them, and they had an automatic submerged arc welding process where they would lay down the cladding in six wires.

So, they'd roll the thing and make one pass, pick it up and move it, and do another, and that's what these two beads are shown here, bead one and bead two, and at the region where they overlap, in the heat-affected zone underneath, it subjected the forgings to some cracking. This was discovered, I believe, in Germany sometime in the late '60s or early '70s.

What we found at B&W was the largest crack that we had seen when we did NDE.

It was about .1 inches deep and a half-inch long, and what happened back in roughly 1970 was that a fracture mechanics analysis was done to show that that flaw would not grow and the reduction of toughness would be such that it would not be -- it would not jeopardize the integrity of the reactor vessel at the end of the 32 effective full-power years.

So, this became an issue that we had to address for license renewal, because it was an issue that was -- that resolved this at the beginning of operation of our plants.

This will be the subject of Ken Yoon's discussion about Appendix C of our document.

For the other items, we have just covered the aging effects in the last slide on the record for the vessel shell enclosure head.

The other items will be the reactor vessel nozzles -- these are clad low-alloy steel nozzles, again cracking at welded joints, cracking at the inside nozzle radius. There are higher loads on some of our bigger nozzles that could be subjected to stresses at the inside radius, and loss of external material due to boric acid wastage. Again, the closures could leak.

For the alloy-600 nozzles, which would be the CRDM penetrations, the IMS nozzles down at the bottom, we have cracking at or near the heat-affected zone. We have seen cracking not of any of these nozzles but other alloy-600 items.

It typically occurs at or near the heat-affected zone in the base metal, as opposed to the 82-182 weld. So, that has has been our experience, but that would be an applicable aging effect for those nozzles.

The reactor vessel interior attachments are alloy-600. Those are the items that catch the internals should they fall. Cracking at or near the attachment welds. And for the reactor vessel, pressure retaining bolted closures, loss of mechanical closure integrity.

We could have loss of material of the alloy steel studs, cracking, or stress relaxation, but again, the aging effect is loss of mechanical closure integrity that must be managed.

Listed here are the generic aging management programs that are credited for managing the aging effects of the items that we discussed earlier. ASME Section XI, subsection IWB, 1989 edition -- the staff has to have -- NRC staff has to have something to pull off the shelf to look at.

It's the 1989 edition, with appendices seven and eight. Appendix seven and eight deal with qualification of NDE for UT and performance demonstration for UT. These are credited for managing cracking in welded joints, again the fabrication flaws you're looking for.

B&W owners group for reactor vessel integrity program is credited for managing reduction of fracture toughness.

Those are NRC requirements for 10 CFR 50.60, which is fracture toughness, and 50.61, which is pressurized thermal shock, both of which are time-limited aging analyses, and 50.60 gets into the surveillance program.

Technical specifications, the pressure temperature limits, again tied to 50.60, RCS chemistry is credited as an aging management program and RCS leakage limits, primarily for bolted closures.

Commitments to NRC generic communications -- Generic Letter 88-05 is the boric acid wastage generic letter that required all licensees to prepare a program to address boric acid wastage.

Bulletin 82-02 is degradation of threaded fasteners in RCS components, and most recently, Generic Letter 97-01 concerning PWSCC of reactor vessel head penetrations -- we made a commitment in our report that inspections and activities that will be done in the current term of operation will be carried forward to the period of extended operation to manage this aging effect in the next 20 years.

DR. SHACK: The analysis that's used to identify the most limiting components there is based on the EPRI susceptibility model?

MR. RINCKEL: I believe it's -- Matthew, you may be able to answer that.

I believe it is the EPRI susceptibility model that is used to do that, and it considers the material, the stress, the chemistry, and there is a time to crack initiation probability and so forth.

So, I believe that is the EPRI model. Our expert on that is not here today.

I wanted to get into now, really, the time-limited aging analyses associated with the reactor vessel, and the first one that we addressed in our report is thermal fatigue. So, I'll give a summary of that, and then the next item would be compliance with 10 CFR 50.60 and 50.61.

Again, that manages reduction of toughness of the belt-line region. That includes pressurized thermal shock to 480 FPY and the upper shelf energy evaluations.

Growth of inter-granular separations I referred to earlier. We did a fracture mechanics analysis, and Ken Yoon will be discussing that.

The last item would be flaw growth acceptance in accordance with ASME Section XI. When NDE is performed on structural welds in the vessel, if there are any indications that exceed allowable, they become defects, and the options are to repair or to evaluate.

We have found some flaws that have exceeded the acceptance criteria in some of the vessels.

I think Oconee unit two has one. Not many, but they've been evaluated, and there is a fatigue flaw growth evaluation that's done to assess how big the flaw will get at the end of the design life of the component. So, we've had to revisit those.

We did not do that in our generic report. That was a plant-specific evaluation. So, Oconee is handling that through their application.

Our first time-limited aging analysis is thermal fatigue, and when we started into this thermal fatigue area, all of the RCS components have cumulative usage factors calculated for them, and we found that a lot of the transients that go into the calculation of that not only apply to the vessel, they apply to the piping, they apply to every component.

So, you can't really just look at cumulative usage factors for one component; you need to look at all of the components and really get a good basis of what your fatigue design basis is.

So, what we did is we summarized -- Framatome summarized the cumulative usage factors for all the class one components, including the identification of what the transients were that were the controlling factors for those usage factors.

We determined that the current number of design transients would be valid for the period of extended operation, and we also were requested and required to assess the impact of environmental-assisted fatigue. All of that was done in our specimen of fatigue.

And what we started off doing was looking at preparing matrices summarizing the usage factors and the applicable normal and upset transients that contributed to the usage, and for instance, you would have heat-ups and cool-downs from 70 degrees up to 580 degrees. That would be one transient that would have a contribution to usage factor.

The Oconee is designed for 360 such cycles over the 40-year design life. It's stated as such in the FSAR. That's why it became a time-limited aging analysis. There's nothing magical about 40 years; it was just stated that way.

Our job was to look at all of the transients that went into those usage factors, the heat-ups and cool-downs, reactor trips -- there are a number of them that go into the calculation -- and really assess where they are now and where they're going.

Are the original design cycles still okay for 60 years? That was our whole objective of doing this, and we found that, yes, a lot of these plants come up and are base-loaded, and they simply are not accruing cycles such that would put them beyond their cycling at 60 years.

We found the controlling transients for almost all of the RCS components to be listed here -- heat-ups and cool-downs, reactor trips, HPI actuations, EFW, rapid cool-downs, and natural-circulation cool-downs. So, those are the controlling transients for the usage factors for almost all of the RCS components.

For the controlling transients listed on the previous slide, we made an assessment of the number of transients accrued to date for each plant, and I had one for Oconee. Let's see if I can find that. Here we go. This is something that we did.

Oconee unit one is shown here, and these are the heat-ups and cool-downs that they have accrued over time, and you can see, up to 2001, they have accrued about 100. We then did a conservative projection about -- for the next 20 years, up until the end of the period of extended operation or close to it.

The line up above shows the number of design cycles, 360 design cycles.

So, you can see that they are projected to be well below that at the period of extended operation. Therefore, there was no need to increase the number of design cycles for any of the design basis transients.

And we did conclude that, for the reactor vessel and really for all of the RCS components, that the current design cycles are accepted for the period -- acceptable for the period of extended operation.

DR. UHRIG: Do you also add in reactor trips --

MR. RINCKEL: Yes.

DR. UHRIG: -- rapid cool-downs? So, that would make it a higher projection on there.

MR. RINCKEL: Well, each one of those transients would have its own curve.

DR. UHRIG: Oh, okay.

MR. RINCKEL: If you have a usage factor of .9, let's say .5 would be attributed to heat-ups and cool-downs, perhaps .1 to reactor trips, and so forth. It's based on each of those transients you consider, and oftentimes, the heat-ups and cool-downs are bounding. They bound many of the other transients because of the stresses applied and so forth.

So, that's where we had it. We had a separate curve for each one.

In our report, we had demonstrated that the existing usage factors, with the exception of the Oconee ONS reactor vessel studs, remain valid for the period of extended operation, and the reactor vessel studs actually have a usage factor of 1.04 now that I think has since been revised due to -- and recalculated. So, I believe Oconee has taken care of that.

There is a program in place at each of the utilities to monitor these design transients, and we could not go into the detail in our generic report of describing the plant-specific programs.

So, that became a license renewal applicant action item, to describe their thermal fatigue monitoring program. As part of the license renewal application, Oconee has done that, and I think Bob Gill will discuss that a little bit later.

The last thing is, once we had a good handle on the fatigue design basis, understood what the controlling transients were, understood where they were today and where they're going, we had to do an assessment of environmental-assisted fatigue, and we did that for the items, the reactor vessel items evaluated in NUREG-6260.

We used the ANO model described in NUREG-6335, applied environmental factors for the faradic items, and showed that the usage factor would be less than 1. So, we did address environmental-assisted fatigue for the vessel items again.

The factors are not as high for the faradic items as they would be for stainless steel, and I think there is some controversy as to the stainless steel, but we didn't have that to deal with, because we were all faradic in the vessel.

DR. SHACK: On the limiting items, is that on a design basis, or that's actually going back and looking at the actual transients and seeing -- and looking at those usage factors?

MR. RINCKEL: It was a study that was done by, I believe, ANO or the NRC on identifying the limiting items in the vessel, and the items were the nozzles, inlet-outlet nozzles. the core flood nozzle, the weld that connects the lower shelf to the transition forging, I believe were the specific items, and I think I saw John Fair here.

Is that right, John? Okay.

Yeah, John's nodding his head.

So, we looked at those specific items as the items to apply the environmental factors to.

I'm not sure -- I think we also looked at the IMS nozzles at the bottom and I believe the CRDM penetrations at the top, of the alloy-600 items.

So, that was our assessment of thermal fatigue in the reactor vessel report, and the next item is compliance with 10 CFR 50.60 and 50.61, which addresses the reduction of fracture toughness in the vessel, and I thought, really, the best way is to have our expert on our surveillance program give you kind of a history of our reactor vessel integrity program.

It was formed, I think, about 20-some years ago to address the problems with the Linde-80 welds that we have, and it's really an outstanding program, and I was very fortunate to have Matthew help out with this. So, I'm going to turn it over to Matthew here.

I'll turn the slides for you, Matthew.

MR. DEVAN: I'm Matt Devan from Framatome. I'm a metallurgical engineer, and as Mark indicated, I want to give you a brief background of the master integrated program, which I'll refer to as the MIRVP throughout this presentation.

What I would like to do first is pretty much just restate the NRC requirements for fracture toughness requirements and material surveillance requirements.

As Mark indicated, 10 CFR 50.60 requires that all light-water nuclear reactors must meet fracture toughness requirements and material surveillance requirements, as documented in Appendix G and Appendix H of the Code of Federal Regulations.

Also, as part of fracture toughness, we have 10 CFR 50.61, which requires the protection against pressurized thermal shock.

10 CFR Appendix G has requirements, again, for fracture toughness requirements for reactor vessels. One of the requirements is that the upper shelf energy shall not be less than 50 foot-pounds. This was a problem for the Linde-80 welds in that, during the life of the plant, these welds had a low upper shelf energy value and would drop below 50 foot-pounds.

Again, Appendix G allows an equivalent margins analysis per ASME, Section XI, Appendix G, and this has been performed through the end of life with an -- at the NRC with an SER.

Also, in Appendix G, they have requirements for pressure/temperature operating limits, and they utilize the predicted shifts of the reference temperatures, which utilize -- which you can utilize the Reg. Guide 199, Rev. 2, methodology used to calculate the adjusted reference temperature, which is then used to develop these pressure/temperature operating limits.

10 CFR Appendix H is the material surveillance requirements. It utilizes the ASTM E-185 standard, which is basically the standard practice for conducting surveillance tests for light-water nuclear power reactors.

It also states approved withdrawal schedules for capsules for surveillance, for monitoring reactor vessel embrittlement.

It also contains integrated program rules, rules for integrated program, which, again, the MIRVP is an integrated program. So, these are keys that we had to develop when we created this program.

Some keys for the integrated program were for similar design and operating features of reactor vessels, and reactors must have an adequate dosimetry program and also the data-sharing arrangement for these reactor vessels.

For the B&W fabricated reactor vessels, for the PWRs, there were two NSS designers. One was B&W and one was Westinghouse.

The materials that were used to fabricate these vessels, as indicated by Mark earlier, were -- for the plate vessels, they utilized SA-302B, modified, which was modified by a code case, and those were the earlier plants, Oconee one and TMI one.

Also, the later plants or the plants that were fabricated at a later time were -- the SA-508 -- or, excuse me, SA-533, grade B, class one, plate material, and the Oconee three and Oconee two and Davis-Besse plants were forgings, fabricated from A-508, class two.

The welds, again, were utilized for the plate materials. They both contain circumferential welds and also axial welds. For the forgings, they only had circumferential welds, as indicated by the earlier drawing.

All the welds in the belt-line region were automatic submerged arc welds. They utilized the Linde-80 flux, which had a low initial upper shelf energy, and again, as Mark indicated earlier, they were fabricated using a copper-coated wire, which, with the introduction of the copper or the increased amount of copper, can accelerate the reduction of fracture toughness.

For the welds used in the fabrication, each weld wire heat and flux lot had a unique identifier which basically went through a weld qualification for that particular wire heat and flux lot. There were -- welds were qualified both at the Mount Vernon facility and also in the Barberton facility.

The welds, when you see -- for the B&W fabricated vessels, you'll see a WF numeral. That indicates that that weld qualification was performed at Mount Vernon, and SA numerals were basically qualified at Barberton, and all the weld seams in the belt-line region are traceable to either a WF or an SA identifier.

Surrogate welds, just for information, is a -- weld-wire heat can be fabricated from a different flux lot, but when -- as for a surrogate weld, the wire heat is the key, and the flux lot can differ, and it would be a surrogate weld of that.

But the wire-heat has got the unique equivalent copper and -- the copper content, nickel content, and also mechanical properties.

For the surveillance material or surveillance capsules contained in these capsules, again, in accordance with ASTM E-185, contains both base metal and weld metal. The early capsules, which the B&W capsules fall into, they may not have the same WF or SA weld in the vessel belt-line as what's in the capsule.

This requirement was changed in later editions of ASTM E-185, but they do contain both a plate or forging material that is within the belt-line region and a Linde-80 weld associated with that program.

The test specimens that are contained in these capsules -- they are charpy V-notch impact specimens, tension test specimens, and at a later time, compact fracture test specimens were included.

No compacts were included in the very early plant-specific capsules. As time went on, half-T's were included in some of the plant-specific for the B&W reactor vessels, and once the integrated program was developed, supplemental capsules were fabricated using actual 1-T specimens.

Also included, neutron dosimetry wires to calculate fluence and temperature monitors which were low-eutectic alloys which would melt and show the actual radiation temperature exposure that the specimens would receive.

DR. UHRIG: Could you tell me what you mean by compact fracture specimen here?

MR. DEVAN: Ken?

MR. YOON: Compact fracture specimen -- it is a fracture specimen according to the ASME standard. There is various size specimens with two holes in the specimen you can pull under a test machine. You can perform fracture test using these specimens.

DR. UHRIG: It's not impact loaded.

MR. YOON: No. It is just a slow pull.

DR. UHRIG: Okay. It's pre-cracked.

MR. YOON: Yes, pre-cracking is a requirement.

DR. UHRIG: You do mean impact on the tension specimens.

MR. DEVAN: Oh, no. The tension specimens are actual tension tests.

DR. UHRIG: Okay. There are tension impact tests, also.

MR. DEVAN: Right. But what I'm classifying are the slower, actual tension tests.

MR. YOON: In our program, to accommodate the cylindrical shape of the capsule, we used the round compact instead of square, which both are according to ASTM standards.

DR. UHRIG: Okay. Thank you.

DR. SHACK: So, even the half-T are really round geometry?

MR. YOON: No. The 1-T's are round. Actually, it's .9-something.

MR. DEVAN: This is a slide summarizing the reactor vessel integrity program. Again, it was established in the late '70s.

The primary purpose of this program was to resolve fracture toughness concerns with Linde-80 welds because of the low upper shelf energies.

The original participants were the B&W design reactor vessels, which included Arkansas Nuclear one, Crystal River unit three, Davis-Besse, Oconee's unit one, two, and three, Rancho Seco, and TMI one and two.

As time went on, some later participants with B&W-fabricated vessels were included. These were Westinghouse design reactors, which include R.E. Ginna, Point Beach one and two, Surry unit one and two, Turkey Point three and four, and Zion unit one and two.

The reactor vessel integrity program -- the goals were to obtain materials and irradiation effects data, develop test methods and analytical procedures to -- for determination of fracture toughness, and also to provide an effective communication among the owners themselves with these materials, also effective communication with the NRC, and also with the industry.

As I indicated earlier, the plant-specific capsules had deficiencies in that the limiting materials within those plant-specific capsules were not the actual limiting materials within the vessels.

Also, fracture toughness specimens were not included in the plant-specific.

So, the integrated program was developed, in addition to the fact that there were some failures of the capsule holders within the vessels.

So, the B&W owners group developed the integrated program at that point, which established an integrated program for the B&W-design reactors because of the failures of the holders within a few of the reactor vessels.

What they would have would be host reactors, which would host the actual plant-specific capsules themselves, and the other ones would be just basically utilizing -- these host reactors would be obtaining the data because of the similarities of the reactor vessels.

They would be able to pull and test their capsules after being irradiated in these host capsules and provide them the irradiation data that they needed.

The integrated program also added some additional capsules which were classified as supplemental capsules, which were providing additional data for other Linde-80 welds that weren't included in some of the plant-specific.

At the time, the master integrated program has 14 capsules, 14 different individual supplemental capsules, and these were inserted all in power reactors.

DR. SHACK: What are the flux limits on these things, and when you -- presumably, you accelerate these somewhat, but what's the limit on the flux rate acceleration you can give it?

MR. DEVAN: It's all limited on where the location of the capsules themselves are within the reactor vessel. They're based on -- again, their exposure is based on their location, and we project the fluence that's going to be received by these capsules and withdrawn per a withdrawal schedule that is efficient for the participants to obtain the data that's necessary to fill in the data that's necessary for end of life and also for license renewal.

DR. SHACK: But when you add these supplemental capsules, presumably in order for them to catch up, you have to somehow put them in a location with a somewhat higher flux?

MR. DEVAN: These are the same locations of the plant-specific.

The plant specifics are inserted and also withdrawn at different times to -- well, they're inserted, and they get the exposure that is required per ASTM E-185, and then, once they hit that limit or that window, the capsules are actually withdrawn and then stored in our Lynchburg Technology Center, and they are either tested or they're actually stored.

MR. RINCKEL: What's the lead factor?

MR. DEVAN: Well, the lead factor for -- there's two locations within the B&W reactors, and the lead factors for the quarter thickness vessel thickness, which is one quarter of 8 1/2 -- the two locations have lead factors of around 7 and 9. So, they are accelerated.

DR. UHRIG: Do you have any of the weld material among these samples, these capsules?

MR. DEVAN: Yes.

DR. UHRIG: Including the copper that was put into the original welds.

MR. DEVAN: Yes. We have, I believe, eight different weld wire heats, eight or nine, I can't remember, but we have a large number of weld wire heats represented in these capsules, so we have an idea of how each of these weld wire heats is behaving with respect to irradiation and embrittlement.

DR. UHRIG: They use essentially the same amount of copper on the electrodes for the different vessels that -- generally familiar with the Turkey Point situation.

MR. DEVAN: Uh-huh.

DR. UHRIG: Is this comparable to, say, the vessels at Turkey Point?

MR. DEVAN: Yes. Yes.

DR. UHRIG: It was the same procedure, same welding rod or welding wire.

MR. DEVAN: The same process was used to coat the wires, but there was no requirement as to how much copper was going to be put on the wire.

In other words, it went through a copper bath, and then -- so, there are some areas where -- I mean there's no set thickness of the copper coating. So, that's why there are some variations within the copper contents within these Linde-80 welds.

Some welds have, you know, copper contents of .3 weight percent. Others have copper contents of, say, .25 percent. So, there is variation, and again, all these are measured based on a large number -- a very large database of chemistry data that we have at hand right for Linde-80 welds.

DR. SEALE: Let me see if I understand some of the code words you're using here. When you say you have a lead factor of 9, does that mean that I have to essentially put a capsule in for seven years in order to replicate a 60-year anticipated irradiation?

MR. DEVAN: What that means is -- the lead factors that I specified reflect the fluence that has attenuated through the vessel at the quarter-T location, and what that means is that the capsules will lead the vessel wall by seven.

So, if the vessel sees a fluence of 1E18, the capsule exposed at the same period of time would receive, at the quarter-T -- or equivalent to the quarter-T location of 7E18.

DR. SEALE: If that's your lead factor.

MR. DEVAN: Yes.

DR. SEALE: Okay.

MR. YOON: So, your question is correct.

DR. SEALE: My question was correct. Okay.

DR. UHRIG: Do you have thermal seals?

MR. DEVAN: Yes.

DR. UHRIG: So, therefore, there would be a significant reduction.

MR. DEVAN: Yes.

DR. SHACK: Those pressurized thermal shock calculations that you showed -- that was essentially with no additional neutron management or neutron reduction. That's sort of calculated as you're doing it now, so that they have the option of going to a low leakage core or something?

MR. RINCKEL: Yes, that's correct. I think all of our plants have gone to low leakage cores now. They're already there.

DR. SHACK: Okay. So, you can't buy anymore that way.

MR. RINCKEL: No, sir.

MR. DEVAN: B&W, when it generated this integrated program, had a unique situation where they had with I would classify as nozzle drop-outs. These are the areas within -- in the pressure vessel where they cut out to -- for the nozzles themselves, the outlet and the inlet nozzles.

So, what we had was a unique situation in that we had these large disks with an actual Linde-80 production weld within that nozzle drop-out that we could utilize for these supplemental capsules.

This -- again, it added additional data for weld wires heats that were not included in the surveillance program. So, this expanded the database that was necessary to cover some of the belt-line welds that aren't -- were not represented within the plant-specific capsules.

Again, the drop-out -- the welds that were in these drop-outs were utilized in these 14 capsules, supplemental capsules that are part of the master integrated program.

The master integrated program is documented in BAW-1543. The current rev is Rev. 4, and what we have is a supplement document to that which provides the surveillance capsule withdrawal schedules that the plants are scheduled to withdraw the capsules and so whatever is required per E185.

The SER that was issued for Rev. 3 indicated some requests. In particular, we had to do a TMI-2 supplemental capsule re-qualification because of the accident at TMI-2.

They also asked for an analysis of sub-size tensile specimens, because we utilized a smaller specimen than standard tensile specimen themselves, and also, we -- they requested an analysis of our reconstitution process, because one of the capsules we had included reconstituted specimens from previous irradiated capsules, charpy specimens.

These requests were answered and had no further comment from the NRC.

And I would like to conclude with my background by indicating some of the current activities that we're involved with and concluded with as of right now.

We had a post-irradiation testing of a capsule called W-1, which was irradiated in a Westinghouse-design plant, Surry unit two, and what this capsule's purpose was was to document or provide irradiated data from exposure in a Westinghouse reactor vessel, and we had the same material from the same source included in B&W, in capsules that were irradiated in a B&W reactor, and the intention is to compare the irradiated data from a Westinghouse plant to the B&W plants and see what differences, if any, are there, and this is currently -- the evaluation is currently going on and should be completed as part of the 1999 integrated program.

And lastly, the --

DR. UHRIG: What kind of difference would you expect? A spectrum difference?

MR. DEVAN: The spectrum difference I don't think is a problem.

Again, there are some questions of irradiation temperature differences due to the fact that B&W's operate at a higher -- their cold legs are a little bit higher than the Westinghouse folks, and again, I don't know what kind of conclusions we're going to be able to make, because this is such a small database, but it provides a unique situation where we've got the same welds irradiated in two different reactor designs.

The B&W reactors have a cold leg temperature roughly of about 550. The Westinghouse -- there are differences within the Westinghouse. They range from anyway

-- 545 to 540, I believe, somewhere in that range.

Now, there was one reactor -- or reactor vessels, I should say -- Zion unit one and unit two, which operated at a much cooler temperature. They were around 530, but they are no longer part of the program. So, the concern for that is not there.

The last capsule, which was a unique capsule in that it included a previously-irradiated charpy tested charpy specimens that were reconstituted to form new charpy specimens.

So, what this provided was specimens that already had exposure to irradiation and already had the embrittlement, and we could reconstitute those, further irradiate those specimens to get a higher fluence exposure and embrittlement on those specimens, and that testing has just been completed, and the report has just been signed off, and that concludes my presentation.

DR. UHRIG: You alluded earlier to some of the specimens having an impact, a charpy impact value of less than 50. How low was it?

MR. DEVAN: They were --

MR. YOON: Between 40 and 45.

DR. UHRIG: So, it was not a big difference.

MR. DEVAN: No.

DR. UHRIG: Okay.

MR. YOON: Depending on the fluence, but that's about the number we've seen.

DR. UHRIG: Okay.

CHAIRMAN BONACA: Any other questions for the presenter?

[No response.]

CHAIRMAN BONACA: What I would like to do -- we are scheduled for a break, and this seems to be the right time to take it. So, I would like to take a break now and resume the meeting at five after 10.

[Recess.]

MR. RINCKEL: What we're concerned about is the fluence at the inside surface of the vessel. The capsules, the surveillance capsules that Matthew talked about are in Crystal River unit three and Davis-Besse. They are not in ANO and they're not in TMI and they're not in the Oconee units.

Those units all rely on ex-vessel cavity dosimetry, but basically, when we projected out to 48 EFPY, the NRC asked us, well, that's a long ways away, that's about, you know, 30 years from now, and how are you going to ensure that those fluence values that you've used at 48 EFPY are accurate and within uncertainty limits of the correlations that are used for the embrittlement evaluations?

Well, in so doing, that was certainly a valid question.

During the same period of time of the NRC review of our vessel, there was a separate effort going on with the NRC in review of uncertainty and fluence calculations and so forth, and that resulted in the approval of topical report BAW-2241, which addresses the uncertainties of fluence and projection of fluence and so forth, and basically, as a condition of acceptance of the fluence vessels, our fluence values used in our report, our owners have to monitor, using ex-vessel cavity dosimetry, reactor -- the fluence, and using the calculation-based method that's described in BAW-2241, update those calculations on a periodic basis to make sure that the fluence that we have used out to 48 EFPY is still going to be valid.

So, we cannot just put our blinders on and not -- and ignore fluence. We're going to have to continue monitoring.

We'll be using ex-vessel dosimetry to do that, and we will be continually extrapolating out to 48 EFPY to make sure that what we -- the values we've used in our report remain valid.

If, all of a sudden, an extrapolation goes beyond what we used in our calculation -- and our maximum fluence projection was approximately 1.5 times 10 to the 19th -- if, at a later time, it exceeds that, then we would have to update these evaluations.

So, we have committed to a monitoring process that will ensure that these values that we have used in here remain valid.

Now, these values form the basis for the upper shelf energy evaluation that Ken's going to talk about and also the RTPTS evaluations that are performed in accordance with 10 CFR 50.61, and that's the subject of the next slide here.

The next bullet is compliance with 10 CFR 50.60 and 50.61, and the two items I'll talk about would be the RTPTS to 48 EFPY, which is Appendix A of our BAW-2251 report.

Appendix B is a low upper shelf energy. Ken Yoon will be talking about the fracture mechanics evaluation, and then the last bullet is the growth of the inter-granular separations. That's Appendix C of the BAW-2251.

Once we had the end of life or end of 48 EFPY fluence estimates at the inside surface of the vessel for all of the participating plants, we demonstrated that the RTPTS values at 48 EFPY comply with the requirements of 50.61 using Reg. Guide 199, Revision 2.

The results of our calculations, RTPTS welds for all of the participating units were calculated to be below the PTS limits, wich the exception of WF-25 in Oconee unit two reactor vessel, which had a value of 300.1 -- screening criterion is 300, so it was a tenth of a degree above -- and one weld at another plant.

Oconee has subsequently done a plant-specific analysis. That's reported in the application, and the RTPTS value for WF-25 has been reduced to 296.8. They had updated fluence, they had looked at surveillance data, and Bob Gill will get into that a little bit later.

And at this time, I'm going to turn it over to Ken Yoon, who will describe the Appendix B to our report, which is the low upper shelf toughness fracture mechanics analysis, and that's for the limiting belt-line welds that are below 50 foot-pounds, you have to perform equivalent margins analysis in accordance with 10 CFR 50, Appendix G, and that's what Ken is going to describe here.

MR. YOON: Again, my name is Ken Yoon, and I work in the fracture mechanics analysis area.

One of the two fracture mechanics analysis included in the license renewal project -- first one is the low upper shelf toughness issue.

That is really the driving force behind the first creation of B&W owners group, and subsequently, we had all the material testing program was under this program, and for the analysis method and acceptance criteria, we didn't have any in the beginning, but early '80s, NRC wrote a letter to Section XI of ASME pressure vessel pipe boiler and pressure vessel code to provide the acceptance criteria.

So, we started working on it, on this project. It took only 12 years, but we finished it in early 1990s, and the technical basis is well documented in the Welding Research Council Bulletin 413. Additionally, there is a regulatory guide, how to do low upper shelf analysis, is also issued.

For the 40-year design life, all four owners groups, including BWR owners group, completed the evaluation and was approved for their justification for low upper shelf issue.

B&W owners group also performed the analysis for not only B&W-designed plant but our reactor vessel working group members, which is some of the Westinghouse plants having B&W-fabricated the vessels.

So, I'm going to go over the next slide, acceptance criteria. There are three criteria. First one is based on -- all three based on service levels.

First one, for levels A and B, there is a requirement for the crack size, postulated crack size of quarter-T, just like Appendix G, and the safety margin of 1.15, and crack extension of .1 inch was specified.

This J material is the crucial input to this analysis, and B&W owners group performed J resistance curve testing from day one and collected many JR curves for this activity. Some are non-irradiated material and some are irradiated material.

Also, B&W owners group donated weld material specimens to the HSST program, namely 62-W through 67-W series of the both un-irradiated and irradiated specimen testing. That specimens were a B&W contribution to the program.

We collected many specimens, and Ernie Eason took the job of modeling it using the pattern recognition program, and we have a B&W owners group J resistance model as a function of temperature, fluence, copper content, and specimen size. So, that's the basis of this critical evaluation.

For the level C --

DR. SHACK: Now, how does Ernie's curve for the owners group differ from Ernie's curve for the NRC?

MR. YOON: Slightly different, because he exclusively used our database, and he has two or three different wa to look at that data, but ours is exclusively Linde-80 weld data but in a similar format.

For level C, the differences -- the postulated flows depth should be one-tenth of a thickness instead of a quarter-T, and a safety factor of one was given.

In level D, the same as level C, but in this case, the best estimate mean curve was allowed instead of some sort of low bounding materials curve, but those are the only difference.

So, based on this, we used the B&W J material model, and the next one shows some plots of -- because the J material is the key information, I plotted that against the fluence, and you can see that the dotted line is a mean curve and solid line is a mean sigma curve, and this is for high copper Linde-80 W-70 and 209 data points, were plotted as in the illustration.

So, it's trying to show that the model is doing an adequate job, and one thing to notice is B&W plants, design plants, early on, went into low leakage fuel scheme.

So, at their extended life, 48 EFPY, fluence is a lot less than some of the other plants' regular 40-year design life.

So, the results of this evaluation is it is found at all the plants under this program, was found acceptable by the Appendix K. So, that was the conclusion of this program.

DR. SEALE: I must make a comment. This sounds like on-time code development, to mix the jargons of modern management analysis with the codes and so on.

Fifteen years ago, a letter was written to suggest that we needed to look at low upper shelf energies. It was available three years ago, and now you're using it.

It's truly on-time development. I don't think you could cut it any closer. I wrote myself a note here that Demming would be proud.

MR. YOON: It made some of us a career in this business.

[Laughter.]

DR. UHRIG: Could you define what you mean by level A, B, C, D?

MR. YOON: That's the --

MR. RINCKEL: Level A and B are the normal and upset events. Normal events would be like your heat-ups and cool-downs. An upset event would be like a reactor trip.

Level C is an emergency event. For us, that's defined as a stuck-open turbine bypass vale.

Level D is a faulted event, maybe a loss-of-coolant accident, or a safe shutdown earthquake.

So, those are the various loadings that these things are designed for.

DR. UHRIG: Thank you.

MR. RINCKEL: The next item Ken's going to talk about is Appendix C to our report, which is the growth of inter-granular separations. I had talked about those earlier.

Those are the regions on the forgings where the weld overlap is that there are some under-clad cracks, and so, that's what he's going to talk about, is evaluation of those under-clad cracks or inter-granular separations, as we call them.

MR. YOON: Like Mark alluded to earlier, in the early '70s, we found out these under-clad cracks. We have an SER on it. So, that became one of the requirements on this project.

So, we revisited that flow evaluation using modern-day, better solution, as well as a lot more complex loading tables we generated for this project.

So, the cracks we worry about is separation -- I'm a fracture mechanics guy, so it's a flaw. The flaw has maximum depths of .165 inches and lengths of .5 inches. This diagram is not to scale.

So, we assumed, very conservative way, depth of .353, including the cladding, and a length of 2.12. You know, that was the basis for the input flow size, and used this to go through all the particulate growth analysis using all the load tables we created.

So, toughness curve -- so, we used code KONC, KONA method of IW-3600 out of Reg. Guide 199 and particulate growth out of Appendix A of the ASME Section XI code.

DR. SEALE: I'm curious. That looks more like a separation on the lamination --

MR. YOON: Yes.

DR. SEALE: -- between the clad and the base material.

MR. YOON: It's not really truly sharp crack.

DR. SEALE: It's not a crack, and you're not saying that that kind of flaw would propagate in the same way that a crack would propagate.

MR. YOON: It would not.

DR. SEALE: Okay.

MR. YOON: But there's no precise way of predicting that.

DR. SEALE: Okay.

MR. YOON: So, we're attaching it all with very conservative evaluations.

DR. SEALE: So, this is kind of a level of detail in the modeling, even.

MR. YOON: Yeah. It's overkill, but it's a sure way to get rid of the issue.

DR. SEALE: Well, as long as you can say it's a conservative analysis.

MR. YOON: Yes, it is.

So, we had a normal and upset condition, 19 transients.

It's all reported in the Appendix C, and all the design basis transients were used from functional spec, and the stresses from various strategic locations -- we just didn't take one location stress. We just went around, sampled various locations, and we lumped all the fatigue calculations into five groups, and we did thorough job, as much as we can.

DR. SHACK: Five fatigue groups mean you had five contributions to the CUF?

MR. YOON: This is a little different than CUF.

DR. SHACK: You're right. You're crack growth. kay.

MR. YOON: So, the conclusion of this evaluation is, for 48 EFDY, we'll be using all the base transient, design base transient into the cycles. We can show that this crack is no concern. Even though the assumption was very conservative, even that we could show that this was okay.

DR. SHACK: Now, in the crack growth analysis, does it make a difference how you order the groups? Is that what you do? You take the most conservative ordering?

MR. YOON: You mean for the crack stuff?

DR. SHACK: Yeah. When I do the fatigue crack growth, it would be dependent on the order of the cycles, wouldn't it?

MR. YOON: Yeah. So, what you do is you group them and you somehow combine them per yield base, instead of finishing one type of transient all the way through the life, then attacking another one.

DR. SHACK: Okay. So, you bunch them by yield.

MR. YOON: Right. So, you take a portion of that particular duty cycle as part of a per-yield base spectra.

That's my presentation.

MR. RINCKEL: Anymore questions for Ken?

CHAIRMAN BONACA: I have one.

You say that this crack -- you postulate it's a conservative estimate. Why do you say that is a conservative start for the analysis? Is it because, in inspections, you have never seen a crack --

MR. RINCKEL: Yeah. The original size for the B&W vessels was never bigger than 0.1-inch depth and a half-inch in length, and we've started off with a larger depth than that, which is the biggest that they saw in industry, and so, that -- to start off with, and then I think the methods that we used were just very conservative.

CHAIRMAN BONACA: I'm just trying to understand what the words mean. That's because you have never observations of cracks of that size. In fact, they are much smaller than that.

MR. YOON: But the analysis assumed that that flaw is breaking through the cladding.

CHAIRMAN BONACA: I understand that. I'm trying to understand the context.

MR. RINCKEL: Did you have something, Barry?

MR. ELLIOT: This is Barry Elliot.

We addressed that issue in our SER. These are under-clad cracks. Because they're under-clad, they have a very, very slow growth rate. They're not surface-breaking.

As a result, they grow very, very slow, and the assumption they make is that the clad goes entirely through the clad, which is a very conservative assumption. We addressed that in the SER, that issue.

MR. RINCKEL: Well, that really concludes our presentation of the B&W owners group report, BAW-2251, and then there are -- I think, certainly, you can see that we demonstrate that aging of the reactor vessel will be adequately managed to ensure the component-intended functions during the period of extended operation in both requirements for 5421A3 and 5421C, which are the aging management review, and TLAA portions of the requirements in the license renewal rule.

This report has been built on experience and methodologies developed over the past 20 years and outstanding reactor vessel integrity program, and the work -- the fracture mechanics work are really the keys to show that this vessel, the intended functions will be maintained in the period of extended operation, and at this point, I'm going to turn it over to Bob Gill, who will tell how Oconee has used this report in their license renewal application, unless there are further questions.

DR. SHACK: My question goes probably to Mr. Gill, but when you do the plant-specific analysis for the Oconee weld that didn't make the screening for PTS, the plant-specific uses the surveillance data instead of Reg. Guide 199 to estimate the shift?

MR. RINCKEL: Well, maybe Matthew is probably the one to answer that, because we did re-do the fluence evaluation, and it was somewhat -- a little bit lower than what we had used in 2251.

What we used in 2251 for 48 EFPY was based on a 1994 estimate, and we have since revised all of that. There had been a topical report that had been approved. So, the fluence was a little bit lower, not a lot, I think within about 3 to 5 percent, and there was some other chemistry surveillance data.

MR. DEVAN: The evaluations were in accordance with the regulations, which did -- you did have to consider the surveillance data that was available. That was taken into account in the evaluation. And based on all these additional informations and reduced fluences, we came up with a revised value of 296.8.

MR. GILL: This is Bob Gill.

Just a footnote on that. The original analyses for the topical were done in the '94-'95 time period, and you can see how robust the program is, that over time, as more information is available, we had an even better calculation at the time of application last year of 300.1, and then subsequently we've done even more and gotten it below 300.

So, it's just the evolution, and we'll continue this program, the vessel integrity program, which I'll get into briefly here.

So, it's just a natural process.

My name is Bob Gill. I'm with the Oconee license renewal project. I was one of the members of the B&W owners group vessel materials committee back in '77 at Duke, at the very fledgling committee that we started out, and we had serious concerns at that time of the licenseability of Oconee and all the B&W vessels due to the upper shelf energy concern.

So, a lot of effort was put forth at that time, and thankfully, we've been able to continue that, and I think this committee was also one of the main reasons why the B&W owners group got involved in license renewal some five or six years ago, and I appreciate the opportunity to come back and speak to you all again. I think I was here about 18 months ago.

I am going to talk to you about how Duke has taken the generic owners group report and addressed the plant-specific requirements that are identified there, and in our application, we'll cover the overall Oconee application where the report is covered, briefly go over the process we use to incorporate it by reference in actually all the reports that we are using, all four of them.

We'll address the plant-specific items, renewal applicant action items. This is typical for any owners group topical report that you saw, 95 or 98 percent of the issues, but there are always going to be a handful of items to be done on a plant-specific basis.

We consciously identified some of those. They were just not mature at the time we put this report in in '96.

And then I'll go over the Oconee-specific programs and TLAA's that we addressed inside the application.

We organized the application so that the -- chapter two is primarily the scoping and screening results, and 2.4 is the reactor coolant system, and 2.5 is the vessel, and you'll see a parallel on the numbering scheme that allows easier review.

All the vessel components that are subject to aging management review, all the piece parts are discussed and pointed to in section 2.4.5. All the aging effects associated with the vessel are part of section 3.4.5.

The many programs that we credited are all described in chapter four. We did not keep it similar to the previous chapters because there are many programs that cover components associated with other areas, like boric acid wastage will be used in several areas, not just the reactor coolant system.

Section 5.4 is our plant-specific time-limited aging analyses, and the approved owners group reports are referenced in each of these sections where applicable, if you go through and review that.

We did that by conscious to make sure they are on the public docket, they are on our docket at the time of application.

In addressing the renewal applicant action items, we created an item-by-item, two-column format table to facilitate review by the staff, the public, the ASLB, whoever.

In fact, we had some questions or potential issues regarding that, because this report was still under review. But we do have a two-column format, makes it very simple. Here is the action item that's required; here is the Oconee-specific response.

We provided to the staff in a May 10th letter, and that was less than two weeks after the final SE was issued. We knew pretty much what the issues were going to be, because we had seen a Draft Safety Evaluation Report, and we knew what the open items were. So, we were well prepared to go ahead and address those.

For BAW-2251, there are 13 renewal applicant action items, and we addressed all those in the report.

Just to summarize rather than belabor each one, we had to verify that Oconee was bounded by the topical report, and since we were intimately involved in the creation of the report through the several years leading up to its submittal and in the review, we were real confident about that, but we went through another step to do that.

We actually created a process -- and Mark was involved in that -- of going back and re-reviewing the Oconee-specific information to make sure our chemistry was the same, the materials are right, the Oconee-specific documentation.

We have that in-house to verify that everything that's said in a topical report does, in fact, bound the design of Oconee, all three vessels.

We also verified that the programs and activities that we credit in the topical report are, in fact, in place at Oconee, and I'll go over those in more detail in a moment.

We did have to perform the plant-specific time-limited aging analysis, and we identified the fact that the PTS value on unit two needed to be updated, and so, we've actually done that twice now.

We did it at the time of application and then again earlier this year.

So, we've gotten that down to below the 300 degrees, and another area was to provide summary descriptions of all these programs and time-limited aging analyses in the FSAR, and we, of course, did that as part of the application.

This is a -- I believe a complete list of all the aging management programs that we credit at Oconee, and the number one item is a -- is our version of the reactor vessel integrity program, and you can see here that we credit the master integrated reactor vessel surveillance program that Matthew talked about, the cavity dosimetry program -- we have ex-vessel dosimetry on unit two that we periodically remove.

That gives you a -- you know what the flux is at the core, you have this ex-vessel dosimetry, you can then project to see what the distribution is of the fluence, and that helps validate your models.

We are updating the fluence and uncertainty calculations. We're using the approved topical there, keeping current on that.

We do pressure/temperature limit curves. We currently have a set of curves, I guess, under staff review for going out to 33 EFPY. We've already extrapolated that out to 48 EFPY, so we know we're going to be able to operate at that time.

This is an ongoing program, and another sub-part of this overall program is counting the effect of full-power years.

We have an engineer full-time in Charlotte that monitors this, manages the program, attends the owners group meetings that occur periodically, interfaces with the staff, and this is his program to own and manage it, as well the engineers at the site that actually help implement it.

So, we're pretty well vested in this area, and it's a very important program. It's been around at Oconee in one form or another for over 20 years.

DR. SHACK: Are the pressure/temperature limit curves based on the new code case?

MR. GILL: Yes. Actually, they help give us a lot more relief. There are several code cases, I understand, that give us more relief on the MPSH and the minimum temperatures we've got to have.

That's one of the reasons, even though that's under review by the staff, that gives us confidence we'll be able to have valid curves for 60 years.

Another major program that we have been involved in -- and there's another engineer at Oconee -- at Charlotte and Oconee that's involved -- is the control rod drive mechanism, another vessel closure penetration inspection program.

This is the CRDM vessel head. There's a generic letter several years ago that came out -- I guess there was European experience. We've had several inspections at Oconee. We've been involved in the industry efforts. We credit that as an existing program.

We have one more inspection scheduled this fall on unit two, and we'll determine at that point in time what additional inspections and how often and all. That is really a living program.

That is probably the leading indicator of alloy-600 activity in our alloy-600 program. This is the leading indicator of what's going on due to the geometry, the temperature, that type of situation.

Chemistry control -- our chemistry control program is based on the EPRI water chemistry guidelines. It's an industry standard. We continue to update that as new chemistry guidelines come out. We keep current with it. I don't believe the staff had any real questions or concerns regarding this during the review.

We're real confident in that program in that program, too. We have solid chemists and scientists and engineers involved in monitoring, and this is a well-managed program.

I mentioned the alloy-600 aging management program. Alloy-600 is in several locations. We have identified the most susceptible locations. In addition to the control rod drive mechanisms, there are several locations in the pressurizer which are leading indicators because of the temperature there, and we will be inspecting some of those locations in the future.

The in-service inspection plan is very straightforward. That's your Section XI program. We are currently using the '89 edition. We will continue to update that every 10 years or whatever the regulatory requirements are.

As time goes on, we've built into our commitment either to continue using this or 50.55(a) or whatever version of the code in the future. So, we've addressed that.

That is definitely a living program.

Boric acid wastage surveillance program -- Duke has had one of those for many years. There was a generic letter several years ago.

This is an ongoing program. It covers not only the reactor coolant system, the vessel, but other areas inside containment, other systems, and in some cases, some components in the auxiliary building that may be subject to having boric acid wastage. It's primarily carbon steel-type components.

We have a period monitoring program on that one, also.

RCS operational leakage is a tech spec requirement. It's monitored periodically for the tech specs. This is a backup. We don't want to have leakage, but if we do have it, I think the only place we credit it in the vessel is the leakage between the head and the flange area.

Certainly, we don't have any through-wall leakage at all.

And the thermal fatigue management program, which is becoming more and more formalized at Oconee, we credit that through the reactor coolant system, monitoring fatigue cycles.

We've had a lot of detailed discussion with the staff on that, and we're working on improving the formality of that program.

DR. SHACK: Just a question on your chemistry control program. The units are running at different pH's now, right? Some are higher and lower?

MR. GILL: I don't know off the top of my head on that. They should all be about the same program, because it's all one site.

CHAIRMAN BONACA: On the alloy-600 aging management program, you said that you have the inspection planned for the pressurizer?

MR. GILL: Yes. We have identified several components in the pressurizer that -- pressurizer heater sleeves on unit one, level taps and safe ends, spray nozzle safe ends and the vent nozzles on unit three all seem to be more susceptible than other locations.

CHAIRMAN BONACA: For those leading indicators, what kind of frequency do you have for those inspections in the program?

MR. GILL: We haven't identified a frequency yet. We will be setting that up. We've committed to do at least one inspection during the current 40-year term and also looking at, you know, monitoring industry experience to see what's going on.

We really need to look at the CRDM nozzles to see what's happening there, how fast this is growing, and again, I think it's the third inspection will be this fall, and we'll see and let the materials engineers decide how often is important enough to look at this.

CHAIRMAN BONACA: The question I have, I guess, is regarding the program. Does the program include provisions such that you could have indications --

MR. GILL: Yes, that's right. You'd set up a frequency and come back every cycle, every two cycles, whatever is important.

CHAIRMAN BONACA: So, you already have established some criteria, some time tables and things of that kind.

MR. GILL: That's all described in our proposed program on alloy-600, and that will be carried forth into the FSAR supplement.

So, that commitment, then, becomes visible to the operators to carry forth on. We make changes to it; it's covered by the change process for the FSAR. All these commitments end up being in the FSAR supplement, and that's why that particular plant-specific action item was very important, and it's something we're going to be discussing with the staff over the next several months, is the right level of detail there, make sure the right commitments get carried forward and everybody understands how we go forward here.

It's kind of new ground. We haven't had this kind of detail previously in programs of this sort in the FSAR.

CHAIRMAN BONACA: So, all of these programs essentially contains elements of further inspections and frequency --

MR. GILL: Right.

CHAIRMAN BONACA: -- depending on the indications you have, but what you're telling me is that you really don't have yet experience in many of these programs.

MR. GILL: On the alloy-600, the commitment is to do the inspection and, based on that, determine what additional inspections are needed, does it need to be broadened, do you need to come back a year or two later. Those type of decisions are written into the program.

All of our programs have about 10 or 12 attributes of things we need to do, what the effect is that you're looking for, what the scope is, how often you're going to do it, what's the first one, what's the technique or methodology.

We decided that the best way to measure our programs is to set these attributes up and then match up, make sure all the corrective actions are done in accordance with our existing problem investigation program, they're all done pretty much by administrative controls which are governed by the QA topical, and in some cases, there was a regulatory standard that applies, in some cases not, and we just put that down there.

So, the future [illegible]ks that have to look at this understand that total history.

CHAIRMAN BONACA: Thank you.

DR. SEALE: I'm curious. You said that you had a pressurizer sleeves, I think it was --

MR. GILL: Right.

DR. SEALE: -- that you were monitoring on unit one, and there was something else on unit three and so on. There is discernible differences between those two units that tells you to focus on unit one in one case and unit three in another?

MR. GILL: Actually, during the detailed review that Mark did, we found out that the unit one pressurizer heater bundles are actually different than units two and three, have different design, different welding, and actually have this alloy-182 weld in there, whereas units two and three do not, and also, the design difference -- you'll have these as -- in the overall program, but what we're saying is, even of this set of, say, the pressurizer vent nozzles, the unit three nozzles are most susceptible of all the vent nozzles, so we'll look at those. So, based on the groupings, we'll actually look at the most leading indicators of those.

DR. SEALE: That might suggest down the road that you need to look at the unit two --

MR. GILL: Absolutely. If you start seeing indications, the first thing you do is what about the adjacent units, and you have to go in and look at them perhaps at the next outage.

DR. SEALE: It would also appear that communication between your experience and your cohorts in the users group could very well suggest things both ways.

MR. GILL: And the industry, anybody doing alloy-600 inspection.

DR. SEALE: Well, the users group, in particular.

MR. GILL: Absolutely. The communications is extremely important as we start to see more and more indications, more and more folks inspecting, rolling that into the database, and certainly, the owners group will continue on as long as they're owners, and you know, some of the experiences come from ANO, some from TMI. Roll that in. They talk periodically, make decisions on which ones to inspect.

So, it's very important.

Now we're getting into owners group activities, but yes, it's a very important thing.

That's one reason we're confident. It's not just us working on this. We have this resource of everybody else out there in the industry that's looking at the same thing.

The other owners, EPRI and any work they may be doing to help us, European experience, if that comes into play -- a lot of that helps drive -- that's why it's difficult to say an exact frequency or when you're going to do something, because you have a lot of factors from the outside world that may say you need to do that next outage, not -- you can't wait five or 10 years.

CHAIRMAN BONACA: Just one last question about that overhead, the previous one.

Of these programs that you have, I guess all of them will be still in place if you do not go to life extension.

MR. GILL: That's correct. These are all existing programs. Alloy-600 is -- we proposed as a new one, but in fact, we do have activities underway today in that area. But all these others are, in fact, existing programs that we have in place today.

We're very fortunate that we've had such a robust set-up on the reactor vessel and in the entire reactor coolant system, very few new programs.

DR. SEALE: Are there commitments in these other programs, however, that have been added to those programs as a result of the aging analysis?

MR. RINCKEL: I can answer that, Bob. The CRDM, another vessel closure penetration, is one example of that. That is an ongoing existing program where there's a requirement that they will have to do and continue the inspections through the period of extended operation.

DR. SEALE: I mean have you added things?

MR. GILL: I'm trying to think on the adding. Not on the vessel per se.

We've added some pressurizer -- based on the pressurizer topical report that was reviewed, we've addeu some examinations of the pressurizer, and in the piping, we have added some examinations of small-bore piping, and so, there have been some small areas outside the vessel. The bulk of our new programs and activities have been outside the reactor coolant system completely, and of course, the vessel internals, which we may get to later.

Okay?

I should point out, for each one of these programs, we have some lead engineers at Duke, either at the site or in the corporate office, that monitor -- own up to these programs, not just sitting up there in space.

The time-limited aging analyses for Oconee -- the B&W topical, 2251, was the first topical we had actually on the opportunity to identify what the TLAA's would be and then take time to do the evaluation on a generic basis.

The previous topicals on pressurizer and piping did not -- we had not identified what they would be, so we could not evaluate them.

So, for Oconee, we actually -- you know, for thermal fatigue that Mark talked about earlier -- that's managed by our thermal fatigue management program.

For the flaw growth analyses, we did review all the previous in-service inspections handled on Oconee for the previous 20 years, identified one indication at unit one on the vessel.

We've identified others in other components, but this is the one on the vessel, and that is being addressed by our fatigue program.

For pressurized thermal shock -- and we've talked about this several times now -- we've updated the chemistry, updated the fluence, and now all three units are well within the limits for 60 years.

For upper shelf energy and inter-granular separation, we determined we were bounded by the generic analysis, so no further review was required.

The beauty of these topical reports is, once we work with the staff and work through it, then the subsequent users of it need not go through that. Instead of reviewing a whole document, you're down to 13 applicant action items to look at.

DR. SHACK: That reactor vessel indication -- that's a fabrication flaw?

MR. GILL: I believe it was, yeah.

MR. RINCKEL: Yes.

MR. GILL: It was determined real early and was analyzed and accepted at that time. We just went back and re-looked at the analysis and updated it, and we found, you know, several across the whole reactor coolant system we had to do that, and it was, again, the QA records we had to go back to. We had to go back to the ISI reports.

Duke's practice at the time was to send in the actual calculations to the staff. So, it met all the six criteria for being a time-limited aging analysis.

So, we had the opportunity to go ahead and look at all those, but they all turned out okay.

Okay.

On the conclusions from an Oconee perspective, the vessels are, in fact, bounded by the topical report, and it was a well-worth effort for us to do.

The programs that we currently have will continue to effectively manage all the aging effects of our vessels, and the plant-specific time-limited aging analyses have been evaluated for the 60-year operation, and we feel real comfortable and confident that we know about the vessel. Many of us -- some of us, I guess, have been working on this thing for over 20 years.

Any questions about the Oconee perspective on the vessel? We'll get into more about the application in the review later this afternoon.

[No response.]

MR. GILL: Okay.

CHAIRMAN BONACA: Thank you for that presentation. It was informative.

MR. GILL: We'll turn it over to Barry, I guess, of the staff.

MR. ELLIOT: My name is Barry Elliot. I'm with the Materials and Chemical Engineering Branch of NRR. Today I'm going to give you our perspectives on our review of BAW-2251 and also discuss some of the open issues, how they've been resolved, plant-specific issues and how they've been resolved for Oconee.

I had help on this review from the people over here.

We've completed the review of 2251. There were no open issues; there were no confirmatory issues. There were aging management programs, which was discussed by the -- by Duke and by Framatome. We're not going to repeat all that.

We will, though, tell you that the first three programs are discussed in our SER, and they are common aging programs, so they're discussed in more detail under section 3.2 of our SER.

The bulk of today's discussion will be thermal fatigue and the B&W owners group reactor vessel integrity program. As discussed by Duke, the integrity program consists of surveillance data and analyses, and we'll be discussing that in detail.

There were 13 identified plant-specific renewal action items identified by the staff in its SER. Duke has responded to all 13. At the moment, there is one open item. The 13 items deal with scoping, aging management, and TLAA's, time-limited aging analysis.

The one open item is related to the time-limited aging analysis, and it deals with the question of flaw growth, of the flaw in the unit one reactor vessel. We'd like to look at that in a little more detail to make sure it's being analyzed correctly.

That's the only open item at the moment.

DR. SHACK: That's the existing flaw that they have, the fabrication flaw?

MR. ELLIOT: Yes.

DR. SEALE: That's strictly an analysis?

MR. ELLIOT: At the moment, it's an analysis. We want to make sure that whatever inspections are going to be done in the future, that they're going to be adequate for the life of the plant.

DR. SEALE: Is that flaw of a kind that's susceptible to inspection?

MR. ELLIOT: We haven't seen the analysis yet. We haven't gotten that far. That's the open issue, to look at the analysis, look at the inspection methods, and come to the conclusion, you know, what we have to -- if there's anything more than the ASME code required here.

Right now, they're only limited by the requirements of the ASME code, and we have to decide for ourselves whether additional requirements are necessary.

There are two significant -- very significant license renewal issues. They are the vessel surveillance program and the fatigue of the metal components.

Fatigue of metal components is concerned with the impact of environmental fatigue on the usage factor. The staff has completed its review of this issue. The licensee has done an analysis.

It has looked at the impact of environmental fatigue based on the models described in NUREG-6335, and the staff has determined that the B&W owners group has adequately addressed GSI-190 regarding environmental fatigue of the reactor vessel components, and the fatigue of the Oconee reactor vessel will be managed during the period of extended operation.

Now we get to vessel surveillance, and this is a little broader picture of the vessel surveillance.

Framatome described their program. Oconee is part of an integrated surveillance program.

Participating in that program and having plant-specific capsules in that program are from the three Oconee units, TMI one and two, Crystal River, Arkansas

Nuclear one, Davis-Besse, and Midland, and in addition, it has supplementary capsules.

The advantage of this program is that it provides a vast amount of data, much more than would be normally attributed to an in-vessel surveillance program.

In a normal in-vessel surveillance program, only one heat of weld wire would be part of the program, and it may not even be the limiting weld, and that would be the requirement today for any in-vessel surveillance program.

The Oconee one belt-line, unit one has three circumferential welds and six axial welds. There are six heats of different weld materials in that belt-line.

Oconee unit two has three circumferential welds but only two with significant amount of fluids, and they have two heats of weld material.

Oconee unit three also has three circumferential welds but only two with significant fluence, and they have three heats of weld material in their belt-line.

So, in unit one, there are six heats of weld material. Four of the heats of the weld material have surveillance data, and if it was just a plant-specific evaluation, we were lucky if we got one.

For Oconee unit two, both heats of weld material in the belt-line have surveillance data, and for Oconee unit three, all three heats in the belt-line have surveillance data.

That's the advantage of an integrated surveillance program. The disadvantage is that there's no way to monitor embrittlement if something changes in the reactor vessel design.

That is, if they change some core design significantly or significant changes in the dimensions or something, or cold leg temperature, let's say, we do not have data by which to determine the effect of the embrittlement.

So, what we've had Duke do is establish limits on the critical nuclear environment con ıtions such as gamma heating, radiation temperature, neutron flux, and neutron fluence, and they are to monitor those conditions during the license renewal term, and if they project that they are going to go outside those limits, then they would have to come back to us and propose an additional program.

The current surveillance program only applies as long as they stay within those limits.

There are four TLAA's associated with the reactor vessel. The fatigue of metal components. The staff reviewed the TLAA evaluation, and the staff concluded that the TLAA evaluation performed by the B&W Owners' Group on fatigue of reactor vessel components was acceptable except for the Oconee reactor vessel studs. They became a plant-specific action item. Oconee has reviewed, has reevaluated the studs and found them acceptable, and the staff agrees. So that issue is closed.

There is an open issue on the fatigue part, and I talked to you about that before. That was the floor evaluation. We need to look into that a little more.

Pressurized thermal shock. The neutron fluence -- there are two parts to the pressurized thermal shock. There is a neutron fluence part and the chemistry part, and the surveillance data part. I'm going to talk a little bit more about the chemistry and the surveillance data in a few minutes, but on the next slide. The neutron fluence methodology was reviewed by the staff and found acceptable. There was a charpy upper-shelf energy evaluation --

DR. SEALE: Excuse me.

MR. ELLIOT: Yes.

DR. SEALE: It was a month and a half ago, roughly, or maybe two and a half months ago --

MR. ELLIOT: Yes.

DR. SEALE: Time flies when you're having fun.

We heard from the people in Research about a look at the whole question of pressurized thermal shock, and in particular not only the chemistry that you indicated, but also the distribution of the flaws.

MR. ELLIOT: Yes.

DR. SEALE: And they indicated that a systematic look at that problem or that aspect of the problem was under way. Is that in any way reflected in any of the materials here?

MR. ELLIOT: No, it is not. That's a research program?

DR. SEALE: Yes.

MR. ELLIOT: This is a regulatory program.

DR. SEALE: Okay.

MR. ELLIOT: And it's based -- a regulatory program is based upon the analysis we did early when we developed the PTS Rule, which is SECY-82-465, and the reports that we did for Oconee -- I can't remember the other plant.

DR. SEALE: So --

MR. ELLIOT: They were done in the eighties.

DR. SEALE: Yes.

MR. ELLIOT: And this criterion was developed from those analyses. What Research is doing is they're taking the more -- another look at those type of analyses using --

DR. SEALE: With hopefully a more realistic flaw distribution.

MR. ELLIOT: With a more realistic flaw -- what they say is more -- what we say is a more realistic flaw distribution, and seeing what the impact is on the screening criteria of the PTS rule. It may be that it goes up. In that case, you know, maybe no one has a problem. Or, you know, it might go down, depending on -- there are a whole bunch of issues here that have to be evaluated, not only --

DR. SEALE: But the expectation is after that you'll be able to say and how.

MR. ELLIOT: Right. And -- so there's more than just --

DR. SEALE: Yes.

MR. ELLIOT: Flaw distribution here that's at issue. There's --

DR. SEALE: Chemistry and --

MR. ELLIOT: A whole bunch of things. But this rule -- what we're talking about today is what we developed --

DR. SEALE: I got you.

MR. ELLIOT: More than 15 years ago. Okay?

The B&W Owners' Group did a charpy upper-shelf evaluation, an upper-shelf energy evaluation, and it's contained in a topical report. We reviewed the topical report, and we concluded that it provided sufficient fracture toughness data and analysis to demonstrate that all the member plants could meet the requirements of Appendix G, 10 CFR 50, and the ASME code at the end of the license extension period.

The upper-shelf energy evaluation was just an extension of the previous evaluation. The previous evaluation, which had been done in the mid-nineties, was for 40 years, and this evaluation just extended it to 60 years.

The next -- we also reviewed a Topical Report 2274 which had to deal with growth of intergranular separation and low elasticity of forgings in the heat-affected zone of stainless steel weld deposit cladding. Duke went into a lot of detail on that. I would just like to add that the previous analyses were done in the seventies. Since then there has been a lot of changes in the fracture mechanics analyses. This new submittal contains all those changes. It is the most up-to-date analysis. It evaluates fatigue, the growth of cracks, as well as embrittlement. And it incorporates the latest technology we have in those areas.

We concluded the analysis demonstrates that the underclad cracking will not be a problem. It will meet the ASME Code fracture toughness requirements for fracture at the end of 60 years.

There are two things I think are very significant that I thought were of interest, and that was the integrated surveillance program and the PTS analysis. I discussed the integrated surveillance program. I'd like to discuss now the PTS analysis in a little more detail as is written here.

Our original estimate when the B&W report -- BAW

2251 was given to us, we determined that Oconee Unit 2 upper shelves and lower shelves circ weld would be over the screening criteria at the prior to 60 years. So we made this a plant-specific action item. Duke has responded and they've revised the fluence, and in addition I just want to say it revised the chemistry. This is an active program that had been going on since 1992. It had nothing to do with this submittal.

We had during the Palisades review discovered that plants were not sharing data sometimes and they weren't reviewing all the data, so we put out a generic letter in which we requested everybody to evaluate their chemistry data relative to all the other data existing in the industry as well as the surveillance data. And it went on for about three or four years, and as a result, there are some new chemistries. In this case the chemistry went down slightly, and that impacted their evaluation, where instead of having a PTS -- RT PTS value of 304, it went down to 297.

We compared -- the methodology was the discussed earlier, was that they used the chemistry to determine the amount of embrittlement. We looked at that. We compared it to the surveillance data that was available for this heat of material, and that assumption is conservative for this heat. So we feel that the value of 297 is applicable.

DR. KRESS: Would you have reached that same conclusion if the value had stayed at 304?

MR. ELLIOT: It would have been even more conservative. I mean --

DR. KRESS: Yes, it would have. That's right.

MR. ELLIOT: What we look at when we make the judgment is the RT PTS value is the sum of three quantities. It's the sum of the initial value --

DR. KRESS: Shift.

MR. ELLIOT: The shift, and margin. And we look at what -- the surveillance data shifted. Is it accounted for in the shift plus the margin? And in this case it was accounted. The surveillance data is less -- the shift in the surveillance data could be accounted for by those quantities. Or actually those quantities were more than the shift in the surveillance data, so they consider it's conservative.

DR. SHACK: When those chemistries change, is that because somebody else brought in -- I mean obviously the chemistry changes from point to point in the weld when you take the sample.

MR. ELLIOT: Yes.

DR. SHACK: You just have more data and you do a statistical analysis and that gives you slightly different numbers when you look at larger data sets?

MR. ELLIOT: Yes, that's what's happening. In the past we had plants that had their own little data sets, and no one ever put them -- no one had put them all together. B&W had done a little bit of that, but it wasn't all together. And when we put out the generic letter, different owners' groups started putting all the data together. Don't forget, B&W fabricated vessels for Westinghouse in themselves, so we had to get all the Westinghouse data together with the B&W data and put it all together to get the most accurate values of chemistries.

DR. UHRIG: So what you're saying is that you have more confidence in the large sample of data as opposed to the individual plant --

MR. ELLIOT: Right. It's a more robust data base now than we've ever had.

DR. SHACK: You mean nobody actually went off and did more chemistry analyses. They basically just looked at all the data that was really around and looked at it in toto.

MR. ELLIOT: That's true.

That's all I have to say today. Thank you.

CHAIRMAN BONACA: Okay.

MR. GRIMES: If there are no other questions on the staff's review of the B&W vessel topical and the related topicals, the staff will proceed with a presentation on the status of the license renewal activities. We would begin by presenting a general picture of where we stand generically, license renewal issues, and the overall program attributes. And that's going to be presented by the license renewal project manager for Oconee, Joe Sebrosky, who is being ably assisted by Steve Hoffman, who's a senior project manager in the License Renewal and Standardization Branch.

CHAIRMAN BONACA: Yes. And the fact, you know, we are running ahead of time, and I think it would be appropriate to continue with the presentation and maybe a second one we have scheduled for the afternoon so we can gain some time. So with that, let's proceed.

MR. SEBROSKY: Good morning. As Chris said, my name is Joe Sebrosky. I'm project manager for Oconee license renewal. And to my left is Steve Hoffman.

What I'd like to go over is in general the status of license renewal activities, and also a broad overview of the SER related to Oconee license renewal.

The way that we're going to present this material -- you've already seen a foreshadowing of this this morning -- we have lead presenters for each section that are going to do the presenting, but we will also have the principal reviewers in a panel-type discussion up here at the front. And for the most part, that's what you'll see. In some selected cases, you will see just one individual up here giving the presentation.

That's the first couple of slides, just to let you know for the particular sections who the lead presenters are.

I guess I'd like to go to the status of license renewal issues, which is slide number 5 in the package. And for the first section, for license renewal issues, as the subcommittee is aware, there's 108 license renewal issues that the staff is currently tracking. Most of these issues were given to us by NEI in the form of comments on the draft standard review plan that we had issued. Out of these 108 issues, we've binned them into Priority 1, Priority 2, and Priority 3.

Priority 1 items mean that the resolution of those particular issues are needed or the staff felt it was needed in order to resolve issues associated with either Calvert Cliffs or the Oconee license renewal application. license renewal which

DR. SEALE: Okay.

Priority 2 items are less important items but are of a general nature and then Priority 3 are lower priority than that. Out of the 108 issues, the Staff has written proposed resolutions for nine, and the process in general for resolving 108 issues is that the Staff after some dialogue with NEI, the Staff writes a generic position for that particular issue and that is what we have done in the case of nine issues.

We expect then that NEI would write back to us and either agree with the disposition or take some exceptions to it. They have only written back to us on one issue, so out of the nine issues that we have sent letters to NEI on, we have gotten a response to one.

Down the line, once we have that response, once the issue is settled, then the only activity that is left is we have to determine the appropriate disposition for that resolution, be it NEI 95-010, which is the industry guidance or SRP or the draft Regulatory Guide.

DR. SEALE: You've got 108 initially and it looks like you have got a pretty tall hill to climb, but I need a little bit more information to decide how tall.

Of that nine, you have had comments on one. Have you received any indication that you are going to get comments on the other eight or that the other eight are satisfactorily resolved?

MR. SEBROSKY: I will turn it over to Mr. Grimes.

MR. GRIMES: We have gotten some indications that whenever we agree with NEI, we've gotten an indication they are going to be satisfied with the answer.

DR. SEALE: Yes, but that still evades my question.

MR. GRIMES: We are going to talk to you tomorrow about this issue associated with credit for existing programs, which really gets -- I think that is going to be the watershed event that is going to help us start dealing with these issues in a more expeditious way.

There were 17, I believe -- Steve, 17 Priority 1 issues?

MR. SEBROSKY: That is correct.

MR. GRIMES: And we have addressed all of those in the safety evaluations for Calvert Cliffs and Oconee. We have dealt with those issues in some shape or form. As a matter of fact, the issue of credit for existing programs, we have also addressed in the reviews for Calvert Cliffs and Oconee because we have reviewed all the programs. We didn't make any distinction about whether they existed or not, and that formed the basis for our safety evaluation, but at this point I think that once we get over a Commission decision associated with the scope and depth of the Staff's review, then the NEI Task Force and we will have a clearer understanding of the expectation about the depth of the safety evaluation basis for these issues, and so I think that then we will start to see the dialogue pick up quicker on these others, but at this point the indications are that NEI is relatively satisfied. We haven't heard any significant complaints.

DR. SEALE: Well, you basically have 99 or 100 rather than 108.

MR. HOFFMAN: And another point, too, is remember, these came in as comments on the Standard Review Plan. They are not all major issues.

DR. SEALE: I appreciate that.

MR. HOFFMAN: Some of them are just improvements, comments where we can revise and make the SRP a little more efficient.

DR. SEALE: Sure.

CHAIRMAN BONACA: You said that there were 17 Priority 1 issues, and also you said that they were addressed in terms of Oconee and the BG&E application. Okay. How come you only have nine proposed resolutions? You seem to have 17 resolutions.

MR. GRIMES: Well, we just addressed the other eight issues directly in the review, but we haven't got a safety evaluation that addresses how we would propose to deal with it on a generic basis like we do for these nine issues.

CHAIRMAN BONACA: Okay.

MR. GRIMES: And so we just incorporated it into the Staff's review. We dealt with the issue as it was presented to us in these first two applications but there is a lot more work that goes into developing a generic safety evaluation that explains what the expectation is for all plants.

DR. SHACK: But it is kind of a misnomer to say the Priority 1s are the ones that have to be resolved in order to do these, because you have essentially done that part for the plant-specific.

MR. GRIMES: That is correct, but remember we described these things as Priority 1 before we began the review for the first two plants, and so you are correct, to continue to call them Priority 1 must be resolved for the first two applications is misleading to that extent.

We would have hoped that we would have had generic resolutions on these issues but that process hasn't gone as fast as we would have liked. As a matter of fact, it got substantially derailed with this credit for existing programs issue because almost all of our attention has been devoted to developing the underlying policy issues to present to the Commission.

CHAIRMAN BONACA: You seem to characterize the 108 issues as really centering regarding the depth of NRC review. Is that a pretty good characterization of the thrust of the dialogue you are having with NEI?

MR. GRIMES: Yes, because as Steve pointed out, the vast majority of those came from specific comments that we got from NEI on the Standard Review Plan.

Since the Standard Review Plan represents the tool by which the Staff is directed to perform a review of scoping, screening and aging management, and time-limited aging analysis, it is fair to characterize those issues as scope and depth of the Staff review.

MR. SEBROSKY: If there's no more questions about the license renewal issues, I will go on to the standard format for the application.

DR. SHACK: Let me just ask one question. Obviously you are getting generic solutions. I mean you are not going to be going over the pressure vessel report for ANO-1. You have reviewed that. Do you have any feel for what fraction of the work is being done generically, you know, for the next B&W license renewal? Are you going to say 15 percent of the effort, 20 percent?

MR. GRIMES: That is the second time that question has come up. The CFO always asks that question when they look at the budget numbers.

It is difficult to say because, for example, Barry Elliot pointed out in his presentation that there is a broader generic issue associated with how to treat the vessel for all plants.

The B&W owners have a program, but then the CE owners have a different program. The BWR Owners Groups have two or three programs. Westinghouse has 51 varied units and I don't know that I could find the Westinghouse program simply because of its diversity, but at the same time we need to put some clear guidance in the Standard Review Plan that talks about treatment of the vessel program and so we have got one piece of that generic answer with this B&W evaluation, but that does not necessarily mean that what we worked out in terms of the safety evaluation basis for BAW-2251 constitutes "the answer" -- the generic answer that could apply to all of the owners' groups.

Looking at it from that perspective, I think that we made a substantial gain. Whether it is 15 percent or 20 percent is very difficult for us to measure. It will vary according to the issue. It will vary according to the extent to which there are generic features of these issues that cut across all plants.

I think I could contrast the reactor vessel issue with the containment issue. What is the appropriate standard for maintenance and surveillance requirements for containments? There are three different BWR containment designs. There are dry -- there are three different kinds of dry containments. There's subatmospheric containments. Yet the industry's simple view is why don't you just say the maintenance rule and IWL is satisfactory and leave me alone? Maybe I said that in too pejorative a way, but it is difficult for us to say that there's a simple explanation of what constitutes the containment program that will manage aging effects that are applicable to all containments for a 20-year period of extended operation that begins about 14-15 years from now, and then we'll extend 20 years beyond that.

I am not going to be here to make sure that I did it right, even if I live that long, so that is a long-winded answer to say no, I don't have a number for you.

MR. SEBROSKY: Continuing on, for the standard format for applications, Steve is actually the lead for this, but I'll go ahead and give you the highlights.

Back in March we transmitted to NEI, we transmitted the formats for both the Calvert Cliffs and the high level format for what the Oconee SER was going to look like. That was given to NEI with the thought that when a high level look at what we did for operating reactors the SRP and the SERs along with the applications are one and the same, as far as what is discussed in what chapters.

In order to try to come to a convergence on what an application should look life, that is the main reason that we transmitted the SER formats for Calvert and for Oconee to NEI.

There was a public meeting on April 13th and NEI has responded in a June 17th letter -- we just got the response -- where they essentially provide us two different formats. One format looks like the SERs. If you look at the SER for Calvert and you look at the SER for Oconee you will see on a high level that they are very similar. There's of course some differences in the details, but as far as what is discussed in what section, the SERs are pretty close.

The one format that NEI provided in the June 17th letter looks similar to that SER format. They also provided us a format that is different, that's based on a commodity group approach and Steve is trying to set up a meeting in mid-July to discuss the two different formats with NEI. The hope is that we will converge to one format and come to an agreement.

MR. SEBROSKY: That's where we stand on the standard format for the applications.

DR. SEALE: Does everybody know what the commodity group program is?

MR. SEBROSKY: I have to admit to you that that's one of the reasons for the meeting, is to try to understand the commodity group approach and why it was chosen.

Going on to the next slide, the status of the standard review plan and reg guide and NEI 95-010, I think the subcommittee is aware that these documents are in a draft form, the draft SRP and the draft reg guide.

As far as NEI 95-010 goes, that was issued in March '96, and the draft reg guide proposes to endorse it. As far as the SRP and reg guide update plan goes, Chris alluded to the credit for existing program issue that goes to the heart of the scope of the staff's review and also the depth of the staff's review. And you'll hear some more discussion about that tomorrow, but obviously we have to figure out where we're going in those two broad areas before we can come up with an update plan as to how that'll affect the SRP and the reg guide.

That is basically the high-level status of the license renewal activities. I guess I'd like to move on and give you a broad overview of the Oconee license renewal application.

This slide basically has the same information that Greg Robinson provided earlier. I'll just touch on a high level on some notes.

If you look at the schedule in general, we've met all the milestone schedules. Both the staff and Duke have met all the milestone schedules. The SER was actually issued a day ahead of schedule. It was scheduled to be issued June 17, and we issued that on June 16.

As far as the hearing status goes, Greg mentioned that there was a potential intervenor, the Chattooga River Watershed Coalition, and the only thing that I would have to add to Greg's discussion was the Commission did affirm the ASLB's decision to deny the petition in April. The deadline to file an appeal by Chattooga has just recently passed, and the staff has not seen any appeal filed by Chattooga River Watershed Coalition.

As far as the comparison between the Oconee and Calvert Cliffs license renewal reviews, there are some obvious differences. One is a CE plant; the other's a B&W plant. But if you look at the applications, you'll note that Calvert was pretty much based on a vertical approach, in that they for a particular system would list how they did the scoping and screening process for that system, how they identified the aging effects, the aging-management programs and TLAA's. It was based on a system approach.

If you look at Duke's license renewal application, it's more on a horizontal approach. Chapter 1 is the introduction. Chapter 2 on a high level is how they did the scoping and screening process broadly. Chapter 3 is the aging effects. Chapter 4 is the aging management programs. And Chapter 5 is the TLAA's.

When you look at the SER's for Calvert and for Oconee, you'll note that the SER's actually look more like the Oconee approach, in that chapter 1 is an introduction, chapter 2 discusses the scoping and screening process for particular systems, chapter 3 is actually a combination of the aging effects and aging management programs, and then chapter 4 is a discussion about the TLAA's. So although the applications differ in their approach, the SER's look similar.

Also, we've -- as you know, Duke relied on several topical reports, and that's been discussed. Specifically they relied on topical reports for RCS piping, pressurizer reactor vessel internals, and fluence methodology. So that's a difference between Calvert and the Oconee approach.

The only other thing that I'd like to mention as far as differences go is when you actually look at the plants themselves there are some differences, although we're trying to come to convergence. With 103 plants out there, you're going to come across unique differences. And when you look at Oconee, some of the differences are that they have a hydroelectric plant as an emergency power supply. They also have a building called the standby shutdown facility that doesn't exist at Calvert.

Anyway, I just give you that as a note that generic resolution can only go so far.

Continuing on with the license renewal application, regarding the license renewal inspections, there's actually two inspections that are scheduled, and there's a third inspection that's optional. The first inspection was on the scoping and screening process, and that was done in April. The finding out of the inspection report or actually I'll read you a sentence from the beginning of the inspection report.

It basically says with the exception of the items identified in this report, your scoping and screening process was generally successful in identifying those systems, structures, and commodity groups required to be considered for aging management. The issues that are detailed in that inspection report you will also see crop up this afternoon in Bob Latta's discussion. We do have an open item in that area that Bob Latta will talk about. And the inspection report alluded to that open item.

The second set of inspections are on the aging-management review, and that's actually broken up. It's a two-week inspection. The first part happens July -- is scheduled for July 12, and the second portion of that is scheduled for July 26. The staff has actually -- because one of the units will not be in an outage during that time frame, the staff has already gone down there when Unit 1 was in an outage to take a look at areas that are not going to be accessible when they go down there in July.

And then the last inspection is a final verification which is at the region's discretion, and that's an optional inspection.

As far as the future for the Oconee schedule, if you go back to the schedule dates, the next target date is Duke is to respond to the open items by October 15. The staff is scheduled to issue the SER in February. The ACRS final meeting is scheduled for May. And then the license renewal is scheduled for August 2000.

I'd like to move on on a high level and let you know how we handled the Priority 1 license renewal issues for Duke. The next two slides in the package basically tell you what the issue is, a brief description, and then where it's dispositioned in our SER. And the lead presenters will talk about the issues when they come up here. There are, however, four issues that will not have further discussion, so I'd like to touch on those quickly.

The first of those would be 98-0003, which is operating experience, and the note that we have is Duke provided the information. If you look in their application and also in our SER you'll see references to operating experience. And in general the Priority 1 issue is how are you going to use operating experience and to what extent are you going to use it for your SER. We've done that. It's not contained in one section in our SER, it's spread throughout the SER.

The second issue I'd like to talk about just briefly is 98-0009, which is the FSAR content. We have an open item in our SER. The open item number is 3.0-1. We have not settled with Duke what the FSAR supplement should look like. In part of their application they gave us what they believe is the necessary changes to their FSAR. The staff has reviewed that as part of their application, but there are several things that are intertwined with that issue, and if you look at the open item in the SER, I'll just read a quick sentence from it, it says therefore, the resolution and the information that needs to be added to the FSAR will be addressed after other open and confirmatory items are resolved prior to issuance of a renewed license. That will be one of the last open items that we'll take care of.

Another issue that I just note is consumables. It's 98-0012. It is actually not a Priority 1. The reason that I mention it, though, is that there are several open items that are in Section 2.2 of our SER that touch -- that refer back to this consumable position that we just recently issued. So that's just a piece of information for the subcommittee.

And the last issue that will not be touched on by a specific reviewer is 98-0068, which is the coded additions. The note that we have on the slide is Duke provided the information. The concern with this open item was to what extent -- or the concern with this Priority 1 issue was to what extent are code additions going to be used and how is the staff going to judge them to be acceptable or not. And basically what you'll see throughout the SER and the guidance that was provided to the staff is if Duke references a code addition, you have to make sure that they -- a code, they have to reference the addition. And the staff has to agree with that addition. So you'll see that throughout the SER. That's not contained in a particular section.

And as far as the status of the Priority 1 issues, those are the things that I wanted to note.

Unless there's any questions, I'm done with the presentation.

CHAIRMAN BONACA: Okay. I think the next presentation we have on the schedule is Duke's presentation.

So we adjourn now and then resume at one o'clock, and give time also to the subcommittee to participate in that meeting with BG&E.

MR. GRIMES: We're going to hold our monthly management meeting at noon with BG&E and Duke and talk about the status of both reviews, and we'd be pleased to have the ACRS subcommittee join us.

CHAIRMAN BONACA: Okay. With that then we adjourn this subcommittee meeting for the morning, and we'll resume the formal presentations from Duke Engineering at 1 p.m.

[Whereupon, at 11:36 a.m., the meeting was recessed, to reconvene at 1:00 p.m., this same day.]

AFTERNOON SESSION

[1:01 p.m.]

CHAIRMAN BONACA: We are going to resume the meeting of the subcommittee, and I believe we are about 40 minutes ahead of time in our schedule and we have now the Duke Energy Corporation presentation. Hopefully, we will complete the scheduled SCR reviews on time. If we are ahead of time, I would like to possibly advance some SCRs from tomorrow morning into today.

MR. GRIMES: We will attempt to accommodate that. We will keep an eye on the clock and see where you are going and then we'll see whether or not the Staff that had planned on coming tomorrow is available.

CHAIRMAN BONACA: Okay, and in case we can, we will adjourn the schedule at 4:00 p.m. for the discussion of the ACRS, so with that in mind, let's start now with Duke Energy Corporation's presentation.

MR. COLAIANNI: I am Paul Colaianni. I will be doing the presentation this afternoon. Also I do have Mike Sumner up here, who is the mechanical lead. I am the electrical lead for the project, so if you have any questions, which I encourage, do ask as I go along.

First, I would like to put up the slide -- photograph again, and of course being mechanical Greg Robinson forgot to point out the most important feature of the slide, which is the switchyards, which as an electrical --

[Laughter.]

MR. COLAIANNI: -- engineer, I just wanted to point that out. Electrical seems to be forgotten in many things in license renewals.

VOICE: That is our loading dock.

MR. COLAIANNI: For the disciplines, as Greg explained this morning, we split it up to the engineering disciplines basically, and the basic rule of thumb we used is that if it carries current it is electrical, if it supports, protects or restrains the movement of a component, it is civil structural, and pretty much everything else is mechanical. There are maybe a few exceptions to that, but that is kind of the basic rule that we took the whole plant and split it along those lines to begin our reviews.

For the scoping of components, each discipline used a slightly different approach. Structural relies on a CLB definition that appears in the UFSAR, Mechanical went straight from a functional review process and Electrical uses an encompassing approach, so you will see these differences play out as I describe them in the presentation.

All the reviews that are taking place, as Greg described this morning, we had a separate review for the reactor coolant system, separate review for the containment structures, and then we had a systems, structures and components review, mechanical components, and electrical components. These three are what I will be covering this afternoon, the three discipline reviews.

This gives a layout of the topics I will be touching on during the presentation. The first one, the IPA, Integrated Plant Assessment, scoping and screening for each of the three disciplines, and then the aging management review for all three disciplines, TLAA reviews for all three disciplines, and the programs and activities that are credited for license renewal. So that lays it out. The first topic will be the scoping and screening.

We will take up each engineering discipline separately. Next slide.

Now an overall look at the scoping and screening for the three disciplines, the structural and mechanical component methodologies are consistent with NEI 95-10. We use that as the basic guide for going through the reviews. The electrical component methodology follows the requirements of Part 54 and also uses guidance in the statement of considerations that was published with the rule and is generally consistent with the guidance provided in 95-10, although there are some differences and exceptions to that guidance.

The structural review, scoping and screening methodology, the basic methodology was to identify the structures and the structural components within the scope of the rule, and their intended functions, and then from that list to identify the structures and structural components subject to an aging management review, there again applying all the scoping/screening criteria.

This is laid out in a simple flow chart where all the structures are identified, then the structures are scoped and intended functions are identified for the scoping process, then these in scope structures are broken down into the structural components that make up those structures, and the intended functions of those structural components are scoped, so within each structure the structural components are determined whether they meet an intended function or not. From all that you get the structures and structural components subject to an aging management review.

Here is an example of the scoping summary. Basically all the structures are listed in the left-hand column, and this would continue on for all the structures. This is just a sampling of the first few. The classification of structure is here, and that is either Class 1, 2, or 3 as defined in the Oconee SR.

This defines whether it is within license renewal or not, yes/no, and the function. Basically on these I think there's 12 criteria that define all the intended functions that a structure could have, so basically if it meets any of these functions then it is within the scope of license renewal.

Then the break-out of the detail for safety-related and nonsafety-related and the regulated events, those are broken out separately and each of those criteria is answered yes or no as to whether parts of the structure meet the license renewal intended functions, and in documentation information on the right, so that is like a first page.

I will give you an example of that process. The results shown on a global scale are these are the structures that were found to be within the scope of license renewal. There are several structures that are outside normal structures and equipment pads, and those are grouped down in the last one called Yard Structures, which includes a lot of outside things such as trenches and towers and elevated tanks and transformer pads.

This is a complete list of all the structures that were included in the scope of review.

Going from those, basically this shows a matrix that was used for each of the structures that was listed in the last table. We have got them listed here and then what was broken out here was all the possible components, structural components, that might be in any of these structures. The list would go down further than this for a complete list, and then for each of the structures an "x" would indicate that there are some anchors, anchorages or embedments in the auxiliary building, and the same determination would be made about all the structures, basically outlining all of the structural components, all the piece parts within that structure that would pertain to it.

That pretty much ends the structure scoping and screening. The mechanical component scoping methodology basically looks at -- splits it based on systems, splits up the plant, and the systems are scoped using the criteria and rule along with determining the intended functions to see what intended functions they serve, and then the identification of components within these in-scope systems are determined along with their intended functions, so it is broken down first into system, and then looking in those systems for what components in those systems are in scope.

The mechanical scoping process for each criteria look like this for 54.4(a)(1) and (a)(2), the safety and nonsafety, a functional flow path identification using event, mitigation and calculations. At the start of the process, fluid pressure boundary determinations were made, physical interface identification was made, and then other designated item identifications, anything else that should be included within the review, and this information was documented not only in calculation but also onto mechanical system flow diagrams where the diagrams were highlighted to show the portions of those systems that were in the scope of license renewal for any of the criteria.

This slide shows scoping events that were used scoping calculations that were done in the mechanical scoping process. This is basically all of the events that we used to determine what components in the mechanical systems need to be part of the scope of review.

CHAIRMAN BONACA: I understand there is some difference with the Staff or some questions to resolve, and I am trying to understand what is it. Let me ask a question. For example, you have loss of main feedwater in the scoping. Why didn't you have feedwater line break? I am trying to understand what the issue is, okay?

MR. COLAIANNI: I know the Staff is going to go into some of that explanation of the issue also, but Mike, do you want to get that or --

MR. SUMNER: I think I need to refer that to Greg.

MR. COLAIANNI: Okay.

MR. ROBINSON: In that particular example, we did not exclude looking at things like feedwater line break. What we did is we focused on including the things that have traditionally been part of the design and licencing basis of the plant and make sure that we clearly defined what that set of events was, and then focused on that.

We recognized that over the course of the last 25 years many other events and topics have come along and we have looked at them and addressed them and made sure that we understood their applicability to the plant, but we did not see them as design basis events or events that we would use for scoping.

CHAIRMAN BONACA: Okay. I tried to go through a little exercise to see if I understood this issue, because I think it's one that keeps going back and forth.

For example, I made the example that if you go to line break because if you had to go to line break, you have to have certain equipment to deal with it. In that case you would want to have isolation of the lines.

My understanding is that Oconee doesn't have main feedwater isolation valves but it has control valves used for that function. Therefore the expectation from the rule would be that the control valves, at least the passive portions of that, would be addressed in the rule. Now if you told me they are addressed in the rule because we are including them by some other means, that would be satisfactory to me, but I would like to know what the answer is to that question.

MR. COLAIANNI: And there again we have tried to go strictly from Oconee's definition of what we have traditionally had as our design basis events, and from that, that is basically where we got this list.

CHAIRMAN BONACA: So going back to the question, then there would be passive components in the feedwater control system or the piping. Are they included in the scope of the application?

MR. SUMNER: My name is Mike Sumner. Yes, they are. They are included.

CHAIRMAN BONACA: So they are by some other means. Okay.

MR. ROBINSON: This is Greg Robinson. They were classified on our documents set as being safety-related and how they got to be tagged as safety-related we can debate forever, but in that particular example the piping and the valve bodies and things that were already identified, and Paul mentioned on the highlighted flow diagrams we did go through and highlight the schematics to point out the areas that we have traditionally had labelled as safety-related in the plant, and it does include those.

CHAIRMAN BONACA: All right.

DR. SHACK: But the answer is the process was essentially done by tradition then?

MR. COLAIANNI: The process was to go by what we understand, what Oconee understands as Oconee's design-basis events, and that is the starting process.

DR. SHACK: What are the infamous additional 32 events for possible inclusion? Somehow you did seem to sort through these things in some way. What was that process? Was it again tradition? The 32 weren't traditionally considered safety?

MR. ROBINSON: This is Greg Robinson. I will try to answer that.

MR. COLAIANNI: I'll refer to Greg, yes.

MR. ROBINSON: What we did was somewhere around the late '80s or early '90s with our design-basis documentation program, we realized we needed to write down some of our tribal knowledge in a history.

We had longstanding licensing and design engineers who knew how the plant design evolved over time, but we did not have that written down. In the process of writing that down through the course of the 1990s, we got to a point where we said it would be nice to step back from the particulars of writing down each item as we think it applies to Oconee and take a more global look.

When we did that, we said let's go look around the industry, everything everyone has considered, and we came up with about 58 -- I believe that was the number -- 58 different events that had been considered, some of which were never considered on Oconee, but we wanted to include them in the mix. From that, we sorted through the licensing basis, essentially compiled the licensing basis of the plant to find these numbers of events that you see up here on the screen, and the other number, the 32, were the ones we found not to be applicable, but the broad view didn't occur until the early to mid-'90s.

We backed away from the problem and said let's take a broad view of this and make sure we are in the right ball park.

DR. KRESS: Did the PRA play any role in this at all?

MR. ROBINSON: Not directly in establishing these event sets, no, sir.

CHAIRMAN BONACA: I understood the licensing basis for the plant, but we are looking here for aging of components which play a significant safety role, so to me it doesn't matter if they are safety-grade or control-grade at this stage -- that was part of the original license and we are not questioning that. We are questioning whether or not we are capturing them in aging programs, and you gave me an answer for the feedwater system that said yes, we do. Well, that specific one.

The question is broader in general. It is are you capturing them in any case?

MR. GRIMES: Dr. Bonaca, this is Chris Grimes. I would like to clarify that we have concentrated on applying the scoping criteria and 54.4 and the Staff will explain in its presentation the open item, but when we apply those criteria we apply them to identify those intended functions that are associated with the design basis and so if we find, as you pointed out before, if we find that we can think of an event that they didn't include in their methodology, the first thing we are going to do is go see whether or not it matters in terms of whether or not that excludes an intended event -- or system, structure or component.

But in the event that we find that they did not consider an event and they don't have, as you pointed out, they have got some design differences, if it ends up excluding some system, structure or component, the first question we have to ask is is that a deficiency in the licensing basis that should be treated under Part 50 today, rather than trying to solve it as part of license renewal, so we are trying not to backfit the design basis in license renewal.

We tried to be very careful about that in order to make sure that license renewal wasn't doing something it was not intended to do.

MR. ROBINSON: May I add, Chris -- Greg Robinson -- in addition to the focus that both Duke and the NRC had on meeting the regulations or working to meet the regulations, we did on a plant-specific basis take a look a the risk-significant results from the maintenance rule efforts, and the results from the license renewal efforts, and when you overlay them we have found that the risk-significant mechanical systems that were determined through other risk processes are included in the license renewal scope and do receive aging management review, so I can answer that part of your question.

CHAIRMAN BONACA: So also indirectly you are answering Dr. Kress's question?

DR. KRESS: Yes, indirect answer to mine, too.

MR. NEWBERRY: Scott Newberry, Staff. Just by way of example -- that is a good question and I remember back in rulemaking we talked about risk significance, and one of the reason the scope -- because of those questions, the scope was expanded to explicitly include ATWS, station blackout, and fire protection equipment and they are listed explicitly in the scope.

CHAIRMAN BONACA: Okay.

MR. COLAIANNI: So using also in addition 54.4(a)(3), basically the mechanical systems that satisfy the regulated event criteria were picked out of the licensing commitments and design documents. They related to those for each of those four -- those five events -- and those components were pulled out and made sure that they were included in the scope of review.

Now for the screening -- that was the scoping -- for the screening basically Mechanical used, put a menu up of active versus passive components, and the mechanical groups that were highlighted were run through that menu to determine the passive components in the systems in the in-scope systems that needed to be reviewed, and that is basically what this slide is identifying.

The list of mechanical components subject to aging management review, a list was provided in the application to identify those components.

Here are the results of the scoping, all the systems at Oconee that were included or that meet the scoping criteria, and you have got Oconee systems that are with the plant proper, the safe shutdown facility -- or standby shutdown facility systems, and then the systems at Keowee, which is the hydroelectric plant supplying emergency power, but this is a list of all the systems that meet the scoping criteria.

An example here is given next on how the components were screened. You have got the systems listed here and these are the different materials that might be part of the system and this gives remarks on the materials and where the information came from. This is what shows up in the station calculation that identifies the components within a system, the materials.

Now we are on to electrical. The electrical scoping and screening methodology as basically laid out is a little bit different from the mechanical approach. Except for specific components that are scoped out or screened out, all plant electrical components are included in the aging management review. To explain that a different way and contrast it to the mechanical approach, for mechanical the systems were defined and everything was scoped to determine exactly what was in-scope. In electrical it turned out to be more efficient for Oconee to start with the whole plant and only screen or scope out a few pieces of equipment for particular reasons but leaving the rest of the components in, thereby having an encompassing review of components that are both in-scope and some that are not within scope but not trying to differentiate exactly which ones meet which criteria.

It does include everything that is in-scope but it does also include components that do not really meet the criteria.

So the way that breaks out in the scoping and screening criteria, 54.4(a), the scoping, basically everything is scoped in but a few specific commodity groups of electrical components are scoped out. Evaluation is done to scope them out. The screening criteria for the active-passive components was actually applied to all electrical component commodity groups, so this was done for all of them.

The screening criteria for the replacement criteria was only applied to a few groups of components but was not applied to everything. The basic evaluations for electrical did not break it down int systems to start out with, it broke all the electrical components into components in commodity groups to start out the review.

This chart shows the basic process -- identify electrical component commodity groups installed at Oconee along with their intended functions, and then applying the scoping or screening criteria. These were not done as it shows here really in a sequence. They were all done sort of as independent steps, and then what came out of the scoping and screening were a list of electrical components that were included in the review.

Here we have a table that shows all the electrical component commodity groups. This basically describes all the electrical commodity groups that are installed at Oconee.

Some of the commodity groups are broader than others, but basically that includes everything in the plant.

This table gives the results of the application of the screening criteria, the passive-active screening of components. It was done to all the components. Most of the determinations were made elsewhere or documented elsewhere to start out with in the rule. You have the reference documents. The rule says that these particular components are subject to or do meet the criteria, the passive criteria, and these particular components do not meet the criteria.

The working draft of the SRP in NEI 95-10, which has the same tables, say that these particular components do not meet the criteria. There is a September 19 letter from the NRC to NEI that particularly speaks to these particular components as not meeting the criteria, and at the time this table was made, what Oconee did in the application was address these particular sets of components, some of which did meet the criteria and some of which didn't.

Since then in particular there's been an NRC letter which addressed fuses that probably should be added to this table, but for Oconee really it's just these determinations that really should be of discussion in the application.

This gives the results of all the electrical components that are included in the aging management review. Here you have the component commodity groups that had components that met the scoping and screening criteria, and this describes in words the groups of components that meet the criteria, giving exclusions where necessary, and this lists the intended functions that were used for the components, but that is the complete set.

Now I will go on to the integrated plant assessment aging management review for structural, mechanical and electrical components.

Although the scoping and screening was done slightly differently for each discipline, when you get to the aging management review it is done, really addressed the same for all three disciplines.

The reason is that at a high level you have got component materials and you add in component environments or stressors that could affect those materials and also you look at potential aging effects -- what sort of aging effects can happen to those materials, and then basically you are looking at determining whether those aging effects are applicable to those materials in those environments, and applicable also meaning having a time limit is going to cause the loss of intended function if unmanaged for the period of extended operation.

The TLAA reviews that were performed I will discuss next. The TLAAs involved plant-specific design analyses, focused on boundary conditions or assumptions based on the 40-year operating term, and the action is to assure that the analyses are valid for the extended period of operation or that the effects of aging will be adequately managed for 60 years.

Oconee-specific time limit aging analyses have been identified by reviewing the Oconee UFSAR documented correspondence and other topical reports. The resultant list includes EQ fatigue, tension loss and pre-stress, reactor vessel embrittlement, just as some examples, and no Oconee exemptions were based on a time limit aging analysis.

The TLAA process is consistent with the guidance provided in NEI 95-10, the process that was used by Duke, and provides reasonable assurance that we found them all and evaluated them.

The last area that I will cover is programs and activities credit for license renewal.

This chart gives an overview of all the programs that are credited for license renewal, a total of 50; 28 are existing programs or activities that are going to require no change at all. There are 11 existing programs or activities that need to be enhanced in some way or other, and then there are 11 new programs or activities that need to be instituted at the station.

This is a list of the 28 existing programs that do not require any changes for license renewal, this is the list of the existing programs to be enhanced, programs and activities, and a listing of the new programs and activities. Most of these new programs, from here on down, are inspections, one-time inspections.

CHAIRMAN BONACA: Before you change that --

MR. COLAIANNI: Yes?

CHAIRMAN BONACA: -- just could you give me an example of an enhancement in one of them, just to get a feeling for --

MR. COLAIANNI: Okay. Mike, can you give us an example? Maybe the Keowee oil sampling?

MR. SUMNER: My name is Mike Sumner. The Keowee oil sampling program at the hydro station has been there since 1970 and they take oil samples on a periodic basis for years and have them analyzed, but it wasn't formalized. The results were very hard to come by. They just did it. It was done by the fossil hydro department.

We enhanced that program by making it very formalized and having a bonafide frequency and documenting results and keeping track of stuff like that. That is a particular enhancement.

CHAIRMAN BONACA: Okay, thank you.

MR. COLAIANNI: All right. Talking about the safety evaluation report, it was recently issued. There are 43 open items and six confirmatory items; 28 of the 49 are relatively straightforward to address. We don't see any real complications coming in for those. Three of the 49 items are related to the UFSAR supplement. Eighteen of the 49 in five different topic areas will require meetings. The topic areas are scoping process and results, complex assembly boundaries, consumables, CASS components, and reactor vessel internals.

To end up -- some observations on implementing the license renewal rule. These are rather broad but basically we saw a need to develop clear definitions of terms so that we as an industry and the NRC can be always talking from the same page. That would help streamline the process.

Document scoping and screening processes -- in a lot of respects that is talking about the electrical process which wasn't represented in the NRC guidance and/or in the inspection plans, and just basically it would make the process easier if that were included, to broaden the ability of the utilities to use efficient means to get things done.

Also, develop a technically sound process for handing emerging issues -- GSIs is an example. That sort of thing. But these are just some broad topics, observations that we have had that would help the process.

That ends my presentation, unless there are any questions on any parts?

CHAIRMAN BONACA: Just looking at the programs you had to institute, you have the 11 new programs and activities. This seems to be a significant fraction of the overall programs that you are talking about. I mean you had 28, 11 are enhanced and 11 are new, but --

MR. ROBINSON: This is Greg Robinson, if I may, a little explanation of the programs.

If you will note, there are two new programs, and both of them were discussed or mentioned, at least the Alloy 600 was mentioned in this morning's discussions. Those are really the only true new programs. A point on the inspections below that, the nine inspections below that. In many cases when we could not characterize an aging phenomena, we just did not feel comfortable technically to say a phenomena was not occurring. We said why don't we go look, and so the one-time inspections are aimed at doing that aging characterization.

If there is aging present, we will continue on and the process will allow us to implement some programmatic action. If there is not any aging present or we cannot determine that there is any, we will then be able to form a better technical conclusion and those will drop off, and so there's only two that we plan to carry forward, so the percentages change when you look at it from that perspective.

MR. TUCKMAN: Mike Tuckman. If you look at it from the perspective of work hours expended in the year, there is not a relationship to the number of programs. Most of the programs that are in place are very heavy usage programs. These one-time inspections are relatively small in comparison.

DR. KRESS: When you look at the fact that you did add a couple of new programs, and you may add more depending on these inspections, is that a lessons learned for operating plants that aren't yet thinking about license renewal? Should those programs be there?

MR. GRIMES: This is Chris Grimes. I will tackle that one.

We would not expect individual license renewal applicants to reflect on the generic implications of these findings. We have identified half a dozen to a dozen issues that have come up as the guidance for license renewal has formulated. We refer to the panel that reviews events and determines what things warrant further action. Some of these things have evolved in bizarre and unusual ways, but they do get fed back into the operating reactor program.

DR. KRESS: That was basically my question.

MR. GRIMES: We feed back this experience into the normal, into the regulatory process because license renewal is predicated on the regulatory process carries forward through the period of extended operation in order for us to focus on just this small set, and I would like to provide a different perspective on the statistics.

That is, irrespective of whether or not the inspections are done every day or they are done once in a 60-year period, they lend to the public credibility of our knowledge, understanding and ability to address whether or not those aging effects will have an impact on the intended function in the unlikely event that a design-basis accident should ever occur, and so from our perspective on the statistics, we essentially weigh things that occur routinely almost the same way that things that we want to just verify we never need to anything more about it.

CHAIRMAN BONACA: I had a question. These are inspections -- the ones on the right side, my right side. I see that pressurizer examinations, but this morning I asked a question because I saw a program that covered that and I was told that the program provides for examinations to be stepped up in case you have in fact findings from those inspections.

MR. GRIMES: Right.

CHAIRMAN BONACA: So these are not just one-time inspections outside of some kind of programmatic requirement. You have some programs under which you are going to put those?

MR. GILL: This is Bob Gill. Let me clarify. These are different pressurizer components than the ones we talked about this morning. This is in fact the heater bundle and actually the interior cladding and spray nozzle, not the Alloy 600 parts.

CHAIRMAN BONACA: Okay.

MR. GILL: And in fact the pressurizer cladding was a concern from an operating experience event about 10 years ago. We are going to go in and look to see if there is any indication of iron oxides on the cladding and then go further, the pressurizer heater bundles, the stress corrosion cracking of a weld which might lead, and so that is different than the Alloy 600 big program we were talking about, which is up there at the top. There is overlap on some of these. It's just the way they got binned and when they were born and that type of thing.

DR. SHACK: A similar question on the small bore piping in the sense that you have had problems with small bore piping --

MR. GILL: Right.

DR. SHACK: -- and you are looking for now a particular in this inspection.

MR. GILL: This would be different piping. If you flip up the other overhead with the existing programs, let's touch on that briefly.

The existing programs that we have had operating experience on is the program to inspect the high pressure injection connections throughout the cooling system. We had the event a couple of years ago. Many years ago the B&W Owners had an event, created a program. There was a generic letter and all that. We had some problems on implementing that program, but that is an existing program that covers that specific location, its unique thermal phenomena, its situation there. We do RT/UT, all kinds of examinations on that. Many inspections are done on those nozzles. Those are HPI makeup nozzles.

To flip back to the small bore piping, that is different. That's events in drains and impulse lines and other things that are less than four inches, not the HPI nozzles per se.

CHAIRMAN BONACA: All right. We would like to get a sense from the Staff when we have a presentation of how this compares with the BG&E application.

MR. GRIMES: We are going to cover that during the separate discussion tomorrow on the credit for existing programs.

CHAIRMAN BONACA: Okay.

MR. GRIMES: In a general way -- and we will try to show you the contrast.

CHAIRMAN BONACA: From our perspective it is very hard to compare. It seems almost apples and oranges in that BG&E have approximately 400 programs and here we are talking about 50. They are different things, I understand that, but

I would like to put them in the same context so we understand.

MR. GRIMES: We are going to cover that during tomorrow's session.

MR. TUCKMAN: Dr. Bonaca and Dr. Shack, this is Mike Tuckman.

It is interesting to note, since you asked the question about small bore piping, that is not a program that we had identified as something needed to be done. We believe the ASME code was sufficient to and does require various visuals, et cetera, of small bore piping. This was something that came out in the NRC's SER on reactor coolant system piping and when you talk later about credit for existing programs, one of the concerns of the industry has been the accretion of requirements from existing programs, and that would be an example of one that we added as a result of the review process from the NRC but I don't know that we would necessarily agree it should have been added.

Did I do that right, Chris?

MR. GRIMES: Yes, sir. We twisted his arm. It's just a question of whether we twisted it fairly.

[Laughter.]

DR. SHACK: Would that be true also of the reactor vessel internals aging management program?

MR. TUCKMAN: I don't think so.

DR. KRESS: Does Oconee deal with design basis hydrogen by using recombiners?

MR. TUCKMAN: Yes.

DR. KRESS: Is there not a program associated with those that -- I didn't see it on your existing programs. It seems like it is a component needed to mitigate the design-basis event.

MR. COLAIANNI: And it is included in the review.

DR. KRESS: It is part of the review?

MR. TUCKMAN: Yes.

DR. KRESS: Gets screened out for some reason?

MR. SUMNER: This is Mike Sumner again. We evaluated and it had no aging effects because it is stored in the warehouse. It is a portable piece of equipment.

DR. KRESS: Oh, I see. It's not in there --

MR. SUMNER: It is not in the reactor building, no, sir.

DR. KRESS: I see, I see, so there wouldn't be any aging --

MR. SUMNER: Right. We keep it out in the warehouse and keep it warm.

DR. KRESS: -- because it is in a controlled environment.

MR. SUMNER: Yes, sir.

MR. COLAIANNI: It has a heater that we keep it up at about 200 degrees to keep it warm and dry.

DR. KRESS: Okay.

MR. COLAIANNI: Any other questions on any aspect of it?

[No response.]

CHAIRMAN BONACA: If none, I want to thank you for a really informative presentation. Thank you.

MR. COLAIANNI: Okay. Thanks for the chance.

CHAIRMAN BONACA: Now we are moving to Staff presentations.

MR. LATTA: Good afternoon, gentlemen. My name is Robert Latta. I work in the Quality Assurance, Vendor Inspection, and Maintenance Branch within NRR. My function was to review the aspects -- the application related to scoping and screening.

Section 2.2 of Exhibit A of the application described the methodology used by Duke to identify the mechanical systems and components to meet the requirements of 54.4(a)(1) and (a)(2), that being safety-related and non-safety-related components. These requirements state in part that the plant systems -- excuse me, the plant -- I'd better have that light on, I can't read -- these requirements state in part that the plant systems, structures, and components that are within the scope of this part are safety-related SSE's that are those relied upon to maintain functional during and following design basis events as described in 50.49(b)(1).

However, as described in the application, the design criteria to which Oconee Nuclear Station was originally built did not include all of the systems, structures, or components that needed to be included under the safety-related criteria defined under 54.4(a)(1) or the non-safety-related criteria defined under 54.4(a)(2). Therefore, Duke relied on the results of a design study that identified the systems and components that are needed to fulfill the safety-related criteria defined in 54.4(a)(1).

Since the design study conducted by Duke only validated those functions required for the successful mitigation of Oconee design basis events identified in chapter 15 of the FSAR, it was unclear to us whether or not all of the functions required for the successful mitigation of these DBE's set forth in the Oconee current licensing basis have been identified as required under the rule. Further, since Duke's methodology had not identified all of the SSC's required under 54.4(a)(1), the potential existed that these conditions also existed for components addressed under 54.4(a)(2), non-safety-related SSC's.

During the staff's most recent meeting with Duke representatives on May 11, 1999, involving Oconee's license renewal application scoping issues, RAI 2.2-6, the staff identified two action items that needed to be resolved within the confines of the SER. And those are described on my first slide here, that is, that the applicant is to review their response to RAI 2.2-6 to include a description of the processes used to identify the events for Oconee Nuclear Station's license renewal scoping and expansion as to how these -- and an explanation as to how these 26 events identified during the May 11 meeting are sufficient to satisfy the requirements of 54.4(a)(1) and (a)(2).

Subsequent to the development of these slides, we did receive the letter from Duke that was dated June 22 which provided their revised response to the RAI. This included a description of the 26 events used for mechanical license renewal scoping relative to the second bullet there where we were evaluating subsequent to the receipt the need for future inspection efforts that is an ongoing effort within our organization.

Questions related to the open item or --

MR. GRIMES: Is that all for 2.1, Bob?

MR. LATTA: Yes, sir.

MR. GRIMES: This is the way that the scoping issue that you referred to, Mr. Bonaca, this is the way it's characterized in the safety evaluation, and we have received a response from Duke concerning how they identified the 26 events, and as I mentioned before, our objective in this review is to make sure that we're satisfied that all of the intended functions associated with the current licensing basis have been identified and that the associated systems, structures, and components that are relied upon to perform those functions have been properly screened or have been subject to an aging-management review. And so we're going to, as Bob mentioned, we're going to proceed to pursue the information supplied in the letter from Duke. Copies of that should have been provided to the ACRS, but I'll make sure that Noel --

DR. SHACK: We have it.

MR. GRIMES: Okay.

CHAIRMAN BONACA: Okay. So right now this remains an open item.

MR. GRIMES: Yes, that's correct.

DR. KRESS: Does the staff have any plans to use something like a risk-importance measure, components that end up -- to see if the design basis actually captures all of the ones that you might -- risk important?

MR. GRIMES: We used risk-importance measures in order to focus the scope of our inspection activities. As I mentioned before, we're consciously avoiding trying to challenge the adequacy of the current licensing basis to --

DR. KRESS: I recognized that was your marching orders.

MR. GRIMES: But that doesn't -- as I also mentioned before also, the process provides that we try and do smart samples that we look at things that have risk importance in terms of the processes, the methodology, and the aging-management program. So we're going to look in areas that if there are questions, you know, concerning whether or not the intended functions are really doing the right things relative to plant risk, we find something and we'll pass it on back to folks to think about in terms of the current licensing basis.

DR. KRESS: So you really don't -- oh, you think that might ought to be incorporated into the licensing basis?

MR. GRIMES: If we find something that's risk-significant for which there is some question about whether or not the current licensing basis is the right current licensing basis.

DR. KRESS: Would that have to be subject to a backfit?

MR. GRIMES: Yes, it would. We put it into the appropriate process for making decisions about changing the current licensing basis.

MR. NEWBERRY: Dr. Kress, let me follow up. A week before last we got our staff requirements memo from the Commission on risk-informing Part 50, and in that SRM they tasked us to go look at the definitions of "safety related" and "important to safety." And, you know, Bob just mentioned (a)(1) of the rule in terms of scope of license renewal used the term "safety related."

So even though the SRM is directed at Part 50, we've talked about it with the industry at our first kickoff meeting, and I think we're trying to figure out to what extent that project is going to draw in Part 54. And I think we're going to end up tackling that issue that you just raised in the context of that effort.

DR. KRESS: Once you approve a license renewal like this, though, what's going to come, particularly for Oconee, before you ever get to that.

MR. NEWBERRY: Yes, that's true.

DR. KRESS: You wouldn't go back and grandfather.

MR. NEWBERRY: Mike's going to shake his head no on that. I don't think so.

MR. GRIMES: Like I said, we're trying to proceed along this, you know, walking that very careful line, recognizing that the state of the art will continue to evolve, and we don't necessarily like being on the cutting edge of technology in terms of fixing the regulatory process, but we keep being driven there for, you know, a variety of things.

But in this case, we're just going to try and --

we need to satisfy ourselves that the suite of events -- as a matter of fact, we ought to define the term, what is a design basis event. We'll clarify what we understand the current licensing basis to be, and then we'll proceed from there to make sure that we've got all the functions.

DR. KRESS: The nature of my question was that clearly design basis events incorporate risk-significant or else we wouldn't do them. But my question was are you really limiting yourself to that or are you making some other sort of overall risk evaluation so that you assure yourself that you're not really missing something that might be risk-significant, and in my mind even if it were not captured by the design basis event, you might want to make it part of the license renewal. If it's really risk-significant you want to capture it in the scope of an aging-management program, only to be sure you weren't limiting yourself.

MR. GRIMES: On the 14th when you talk about policy issues you can consider that, because we specifically took the language in the statements of consideration to be an admonition that we should -- the current licensing basis carries over. But, like I said, if we find something we think is important, whether it's a plant-specific question related to the current licensing basis and the state of the current licensing basis, or whether there's a generic implication, you know, we'll refer those to the appropriate processes.

MR. TUCKMAN: Dr. Kress, this is Mike Tuckman. I'm from Duke.

One of the interesting things about this rule is that it is not a risk-informed rule, it is a very prescriptive rule. I think in reality we are covering everything, and if an improvement were to be made in the rule sometime in the future, the scope of things that you look at in the license renewal would be greatly reduced.

DR. KRESS: You're probably right. I looked at a lot of things in here that might not have to be in the scope if you did it really risk-informed. It would probably go that direction.

MR. TUCKMAN: As Greg talked about, we use the maintenance rule as a kind of a tool to look and see how it matched up with license renewal, and of course in the maintenance rule we do look at risk-significant systems and pay more attention to those as we do in license renewal. But as far as the actual rule went thus far, it was very prescriptive and you treat everything the same.

DR. KRESS: I recognize it's strictly a design basis concept.

MR. TUCKMAN: Yes, sir. I think it's very unlikely that we will have missed it.

CHAIRMAN BONACA: Although everything is captured in the discussion, but does it mean that you have to do something about it? All you have to do is to address the need. And so even within the context of a prescriptive rule, I think that it is a way to soften the blow, I mean, you can say hey, this is justification for not doing further inspection. And I think that it's only fair to say that that should be allowed by the rule.

But I think the only place where it is important is where, you know, you may have a component out there because of some insights and it may be from looking at a broader set of initiators, either through PRA or through deterministic approach, or by one component there is important and we may have missed it. And that was the thrust of I guess my question, and I'm sure that that's really what the staff is doing and will bring to closure. And I don't expect to see surprises, I mean, to the question I had on the feedwater system, I got an answer that said we already included it. That's the answer. So -- okay, with that, any other questions?

DR. SHACK: Well, just on a general, I mean, does the license renewal give you a way around backfitting in the sense that you get a chance to look at the degradation of a passive component and have it addressed whether it's safety-significant or not?

MR. GRIMES: We had one example that was just mentioned in terms of twisting the applicant's arm to address the lack of an inspection activity associated with small bore piping, and, you know, we get into at least do one inspection, check and see whether or not the QA program needs to pick something up. And in the course of resolving questions and comments on the standard review plan, there are other areas like that that have come up where the utilities have said they don't think that it's worth it, and we've said prove it, and we've, you know, we'll march through those. Those are the 108, you know, things to do when we have spare time.

If we can embarrass the industry enough into going out and checking some of these gaps, then eventually we will have a full program, but even then by the time we get that cleaned up then operating experience will say there is something else we ought to go check -- you just have to follow up there, getting back to Dr. Shack's comment. These inspections and programs are solely focused on 40 to 60 years, so the licensee is not obligated unless there is a relationship there to go look today. The focus is on aging from 40 to 60, so we are not talking -- remember Chris's comment, to feed it back in -- that is another process.

DR. KRESS: After 60 years has gone by, can we expect license renewal renewal?

MR. GRIMES: I'm sorry, what was the question?

DR. KRESS: Can we expect a license renewal renewal after 60 years?

MR. GRIMES: I would want to answer that question in two ways. The first way is that in this question of credit for existing programs, we were reminded to point out that we have not even completed the first two applications yet, and so it may be expected that the license renewal rule needs to be renewed here in the very near future based on the experience from the first few applications, and we'll constantly revisit whether or not this is the right thing at the right job associated with license renewal, but then the second part of the answer is we specifically point out that 20 years from now Oconee can come back and apply to renew the renewed license for another 20 years on a presumption that the maintenance activities are going to take sufficient care of the plant so that they could justify continuing operation, so long as it is economically viable.

MR. TUCKMAN: This is Mike Tuckman. I am going to let somebody else worry about that problem.

[Laughter.]

DR. KRESS: You are going to be retired by then.

MR. GRIMES: Thank you, Bob. Mr. Gratton and Mr. Shemanski are going to start with the next section, but I do want to ask this. The Staff's presentation was set up on a streamlined format based on some feedback we got from you on the Calvert Cliffs presentation. We are going through each chapter and we are going talk about the open items, the confirmatory items and anything the Staff thought constituted a particularly important or noteworthy thing to pass on to the Committee, but at the same time, we are here to answer any questions that you might have about the Staff's evaluation basis, so don't hesitate to take us in a different direction if you need to.

CHAIRMAN BONACA: Before we start the presentation, the discussion we just had, one thing that I would like to state is that it is impressive to see how many programs exist already in the units and also the insights provided by the maintenance rule, and the existence of corrective action programs.

I mean on the positive side of it, there has been tremendous progress in the industry in the past 10 years and it is pretty impressive to see how ready the industry is to move to license renewal. I mean there is a lot of stuff in place there that is pretty impressive.

MR. GRIMES: Mr. Bonaca, I would like to emphasize that although we talk about disputes over what is necessary for license renewal, we agree that almost all or nearly all of the existing programs deserve the credit that we are going to give them for managing aging effects and the area where these disputes probably should continue is a struggle between the plant operators and the regulators to constantly challenge these areas where there are programs that don't get a whole lot of visibility, that don't get challenged often enough in order to be readily understood as being an effective aging management program.

But we are only talking about out of those 49 open and confirmatory items there are like five areas of controversy, and regardless of whether or not you count individual procedures and come up with a number between 400 and 500, or whole programs and come up with a number that is like 50, having five things to argue about in order to come to a conclusion about granting a 20-year license I think is remarkable in terms of credit to the regulatory process that we have set up over the last, what? -- four years.

CHAIRMAN BONACA: With that, are we ready?

MR. GRATTON: Thank you very much and good afternoon. My name is Chris Gratton and I am the Divisional Coordinator from the Division of System Safety Analysis for the License Renewal Activities, and what I am going to be discussing today are the scoping and screening activities performed by the Staff.

Since Chris already took my thunder, this is a streamlined presentation that will not focus on the process that we use but rather the results of those activities.

What I am going to cover today are the open items that the Staff identified during the review, the confirmatory items -- which is actually only one, so confirmatory item. We will discuss how the Staff addressed license renewal issues, the Priority 1 issues that were in our area of concern, and I will discuss one item of interest, the difference between the BG&E review and the Oconee review that we have just completed.

As you can see on the slide, the first two open items have to do with systems that the Staff considered were within the scope of license renewal yet the licensee did not determine to be within the scope of license renewal so questions were asked of the licensee to justify the exclusion of the recirculated cooling water system and the chilled water system.

The RCW system is a closed cooling water system that removes decay heat from the spent fuel pool cooling system and transfers it to the CCW system. The chilled water system provides air conditioning or cooling air to the control room. Both of these systems we felt met the regulations to be within the scope of license renewal and we are pursuing justification for their exclusion.

The third open item is identified here as skid-mounted equipment. The real clarification is that for an emergency diesel generator supporting the SSF, the licensee identified the skid as being within the scope of license renewal but excluded the components on the skid as being subject to an aging management review under a provision in Part 54.21 and the Staff believes that it was inappropriately applied in that certain components on the skid were excluded without consideration for aging management review. It was just sort of a blank exclusion, so there are clarifications in NEI 95-10 that we believe they should have applied and reviewed those components on the skid and we are pursuing that also.

DR. KRESS: What sort of components are they? Starting the diesel or --

MR. GRATTON: There are some components associated with the fuel oil system. This is piping up to the diesel generator, cooling water to the cylinder cooling jackets, and portions of the air starting system.

The fourth open item has to do with structural sealants in general. The issue came about in questions regarding water stops that were cast in place for the auxiliary building. The Staff identified them as not being identified as within scope of license renewal at all. When the question was brought up the Applicant stated that they did not meet any intended functions that would require them per 54.4.

When that was brought up there is a Staff position on consumables. We consider these to be consumables, and they should have been addressed as such, but because they are passive and long-lived, we felt that they did meet some scoping requirements to maintain the integrity of the auxiliary building, protecting safety-related equipment that may be in the spaces from either flooding or intrusion of water, so we believe they should be within scope and we are still discussing their inclusion within the scope of license renewal.

The fifth open item has to do with staged equipment, and this is Appendix R equipment. It includes items such as pumps and switchgear and cables that would normally not be considered within the scope of license renewal but because this equipment is staged in a warehouse and not continually in operation, the Staff believes that some provision should be made to monitor its again because the rule assumes that active equipment such as pumps and switchgear are continually in use, monitored and tested, and this equipment is available in the event that there is a design basis fire and as such it is not being used.

The last three open items are similar in nature in that the Applicant identified them as being within the scope of license renewal but did not provide justification for its exclusion from an aging management review. They provided a condition monitoring or performance monitoring as the reason why they are not subject to aging management review, but the rule requires that -- or maybe the statements of consideration identify that a site-specific justification that the condition monitoring program that they are referring to should be described in adequate detail, and those provisions were not included.

The three systems that we are talking about or the three components that we are talking about are the Keowee and turbine building roof structures. These were identified as being monitored by the Applicant and replaced based on their condition. Ventilation sealants, which includes sealing material like tapes for the ventilation ducting for the control room pressurization and filtration system, that again would be monitored but replaced on the basis of their condition.

The final one was some fire detection cabling. The Staff feels that more information about how the performance or condition monitoring is taking place is needed so that we will be assured that the components will be replaced prior to their failure.

Those were the open items. The one confirmatory item that we had involved piping segments that provide structural support in particular for boundary points. In the BG&E review, there was a specific section that was written on the identification of these piping segments and the anchors that are included with them.

There was some confusion over the Oconee application identifying these segments and the Staff got together with the licensee and the issue was resolved. We are just waiting for written confirmation that -- and docketed information that will close this item.

As far as license renewal issues, three issues came within the area of the DSSA review for scoping and screening. I have mentioned some of the consumables that we had issue with, specifically the structural sealants. There was a letter issued by the Staff that identified our position on structural sealants, and they mostly include areas such as packing and O-rings, which are excluded from license renewal, structural sealants -- which the Staff considers included because they are long-lived and passive, oils and greases were excluded and filters, fire extinguishers and hoses and other fire protection equipment were excluded but they are subject to certain justifications that are required by the licensee to ensure that their exclusion is appropriate, and the Staff addressed those as they were performing their review.

Cascading -- there were a few issues identified for cascading in the BG&E review, but they did not seem to carry forward in the Oconee. There were not as many instances where hypothetical failures had brought systems that we felt or components that we felt should be within the scope, and it just was a matter of site-specific, so there were no good cascading examples that we could think of for the Oconee review.

As far as the heat exchanger function, the heat exchangers that perform safety-related intended functions for license renewal were identified as having an intended function that included the transfer of heat. I believe in the BG&E review that was not identified, so the industry picked up on that and they did include that intended function for this review.

As far as items of interest, the one that I did want to identify was the difference in the methodology that the Applicant used for identifying the systems, structures and components that were in the scope of license renewal. Two different methods were used -- two different approaches, I should say, were used.

The BG&E used simplified diagrams to identify the bounds of the review, where Oconee provided a voluminous number of flow diagrams that were computer-generated and they were very robust with respect to identifying the ends of -- the boundaries of license renewal. They were very helpful to the Staff because they provided a lot of information about and beyond what BG&E had provided, but BG&E, because they provided simplified diagrams, they tended to have more emphasis on the written text, so there was a lot more description of where the bounds were and tables and charts associated with a number of components, which was helpful in their review.

In the Oconee review, they were sort of sparse, and they provided a lot of flow diagrams and then the final result of the components and structures that were subject to aging management review, but our review is a two-step process, which structures and components are within the scope of license renewal, and of those which ones are subject to aging management review.

We didn't have that first group. We had to pull them off of sometimes as many as 28 diagrams to try and figure out which ones were subject to aging management review.

Neither of them were wrong or in my own personal opinion, since I did a number of these, I felt the Oconee was easier because I could read the diagrams and see them, but that was a large difference, and I believe that I had more success reviewing the latter ones, so I just wanted to highlight that and maybe give Oconee some kudos in choosing that methodology for the Staff to review.

That is the end of my presentation. Paul Shemanski is sitting next to me. The majority of the review was done in DSSA but the electrical portion was done in the

Division of Engineering and Paul up here is from DE, and if there were any questions on how that was done, he could also answer those questions.

CHAIRMAN BONACA: Questions?

DR. KRESS: Would you prefer a combination of the Oconee and the BG&E?

MR. GRATTON: I would prefer a combination of the two. The written text tended to eliminate some of the questions. I had an example of areas where we became sort of tube-locked in our trying to find answers. The seismic anchoring I think was one of the areas of concern. If there was a written text on how they approached that, the questions wouldn't have come out when you looked at the boundaries. They went up to the safety-related, nonsafety-related interface and stopped.

They called that a boundary position when in fact the boundary went beyond that and included the pipe segment and a seismic anchor. That was one example.

Another one was the method that they used to identify components. They used almost like commodity groups to say, for an example, in the intake structure, they used steel beams, columns, plates and supports, and since there weren't any that really fit in there, the trash rack rails, the rails that the trash rack rolls on, was considered a steel column.

DR. KRESS: That is confusing.

MR. GRATTON: It is. When you read that, you are -- you know, there's no steel column there, but when in fact it was made of the same material, it ages the same way, you know, it really belongs in there, but there was no explanation that went with that. It was just, you know, say a diagram and a table, so maybe a little bit more text would have helped.

DR. KRESS: Is this a message that might be transmitted to, say, ANO?

MR. GRATTON: Well, to tell you the truth, Mr. Grimes -- we are currently working that.

MR. GRIMES: We are trying to gather this experience and then fold it back through the process as we settle on a standard content.

DR. KRESS: That makes your job easier.

MR. GRIMES: Yes, and the delicate balance is we want to make our job easier but one of our four principles is that we also want to reduce unnecessary burdens, so we are going to try and find a nice middle ground and then fold that back into either a revision to 95.10 on the contents of the application, or review guidance in the Standard Review Plan.

DR. KRESS: Is 95-10 still a living document that is going to be changed?

MR. GRIMES: Yes. They haven't talked about changing it, but we have said that some of this experience is more appropriate there.

MR. TUCKMAN: Mr. Kress, this is Mike Tuckman. I have the distinction of chairing the NEI Working Group on this also, and our intent is to take the lessons learned from both BG&E as well as Duke, as well as the various issues that we are getting resolution papers from the NRC on, fold those into a revised NEI 95-10.

DR. KRESS: That would be a great way to handle it.

MR. TUCKMAN: Yes, and also just to provide further assurance to you that the industry is working together, Gary Young is here from Entergy and they have been very actively involved in this process too, so they are getting lessons learned to make theirs work a little better than ours.

DR. KRESS: Great. I am glad to hear that.

DR. SHACK: Since they are going to get charged for it.

[Laughter.]

CHAIRMAN BONACA: With that, let's take a 15-plus minute break. We'll start again at a quarter of three.

[Recess.]

CHAIRMAN BONACA: Okay. Let's resume the meeting with presentations by the NRC staff.

MR. GEORGIEV: Good afternoon. My name is George Georgiev. I am with materials engineering in Chemical Engineering Branch, Division of Engineering.

I am here to make a presentation on aging effects. What is different for this application as compared to the BG&E application is that Duke has grouped various systems and identified common mechanisms which cause certain aging effects and have evaluated them in section 3.52 of the application. And section 3.1 is the result of our review of this application.

Basically the section involves only identification of aging effects, and our review consisted of what are the identified materials in the applications, what are the aging effects, and what is the environment. And we tried to find out whether we can identify over and above what the applicant has done. And with all fairness, after we did our review, we didn't find anything different than what they have found. So consequently Duke has done a good job about it.

However, we do have two open items, and all this question is, if Duke did such a good job, how do you have two open items? And I will attempt to answer that.

[Laughter.]

Okay. The answer is that those are imported open items from other sections of our review, because when you do this general lumped together review that is not so much system-specific. It is conceivable that something could be missed when you do a system review. And we ran it through our individual system reviewers, and doing so we came up with the first open item, which is the aging effects discussed and accepted by the staff are not consistently applied by the applicant of the system, specific discussion of aging effects. And we list in section 3.1 which systems these open items relate to.

The second open items relate to buried components, and basically in our review we couldn't get a feel how much buried piping is involved in these facilities. And with all fairness again, the applicant has provided flow diagrams. We went and reviewed the flow diagrams, we took out what was buried, identified, and we more or less we can say what is involved. But then one of that piping is a very large diameter piping. It's 137 inches. And the aging program that they proposed to manage the effects on this buried pipe is such that it is really responsive to this large pipe. It doesn't address the small pipe because the examination would be done from the ID of the pipe. And if it is a smaller-diameter pipe, it cannot do it. So that is the background information of this open item.

DR. KRESS: Are these pipes buried in concrete?

MR. GEORGIEV: They're buried in soil, in soil, yes. They too address buried in concrete. Today is no problem with the buried in concrete.

DR. UHRIG: What are they, discharge pipes, 137 inches?

MR. GEORGIEV: I think intake pipes.

MR. ROBINSON: This is Greg Robinson. The intake piping is 11-foot diameter coming in from the lake, and I believe it's a 9-foot diameter discharge piping. And the piping is coated and wrapped, and was buried at the initial construction.

MR. GEORGIEV: Basically so what these open items is intended to do is seek information, find out what other piping is involved. And another problem was like they do have notes in the flow diagrams which allow that anything, treat waters of an inch up to 6 inch, you can put -- you could use stainless steel. But it is a maze, not shell. So we really don't know what is stainless, what is carbon, what is cast iron. And that is the purpose of this open item, so when we get this information, we can evaluate it and determine, you know, what are the problems.

DR. SEALE: You say that the QA records were not such that you could ascertain what these particular pipes were?

MR. GEORGIEV: We don't have the QA records.

Those are side documents. But we do have with the application the flow diagrams, and with all fairness, those are very good flow diagrams. They have more information than typical flow diagrams, construction flow diagrams, because they were made for license renewal. But it is not that detailed for us to really determine how much is exempt, so to speak, by what they are proposing to do. Maybe very small, but maybe miles-long piping. We don't know. And that is the purpose of that.

Other than that, we have an item of interest. The applicant has performed an extensive review of aging effects and an exhaustive identification of aging effects, which is a compliment that they did a good job identifying the aging effects.

That concludes my presentation and that of panel member Miss Stephanie Coffin, and she did detailed system reviews. So if you have some question concerning the open items, she will be more than happy to give you the specifics.

CHAIRMAN BONACA: Any questions?

DR. SEALE: This is an item that's still to be resolved in the --

MR. GEORGIEV: Yes, sir. We'll receive the information, and when we receive it, we'll have a fuller picture as to determining whether what has been proposed is adequate.

CHAIRMAN BONACA: Okay.

MR. ROBINSON: This is Greg Robinson. If I might add just a thought here, I think one of the things that you're seeing is not availability of QA records, but it's the level of detail we provided in the application to address the problem at hand versus say the piping drawings or construction drawings that we certainly have on site, but we can go and measure off the miles of piping or what not, and we will be providing that information. So I think we're just seeing the comparing and contrasting of the available information set with that provided in the application.

DR. SEALE: Sometimes they're curioser than they are other times.

MR. HOU: My name is Shou-Nien Hou, Material and Chemical Engineering Branch, Division of Engineering.

Now, the common aging management program consists of 13 individual programs. So to make a long story short, I will just go to the open items, unless you have specific questions so I can explain to you.

Now the first open item is related to the Duke quality control program. That program set the requirements of the corrective actions, document controls, confirmation process. Especially it sets the control process and responsibility and activities for initiating the corrective actions for responding to the nonconforming conditions. And the program generally is in conformance with 10 CFR Part 50, Appendix B. So it is quite acceptable, except that it's only covered safety-related components. But we know in the license renewal review it covers safety and nonsafety both.

So after interaction with the licensee, we come out with an agreement that they're going to expand a similar requirement to the nonsafety-related components. But we do need an official commitment either in the updated FSAR or toward the quality assurance program, like documents of Duke-18. And that's our open item.

Any questions?

[No response.]

The next open item relates to the exchanger performance testing activities. You know, the heat exchanger contains a lot of tubings and small pipes, and it's subject to the corrosions that reduce the efficiency of the heat transfer.

So performance testing activity actually is performed, periodic testing of its heat-transfer capability by measuring the flow rate and also the temperature difference across the heat exchanger. But in the standby shutdown facility heat exchangers they only do the flow rate measurement. You know, that's not enough, because that heat exchanger includes air-cooled coiling, also water-cooled condenser.

And they also have the fins. If the fins fail, now even the flow rate's maintained the same, but because of fin failure the flow patterns change and the heat transfer function changes, and that is going to degrade heat-transfer capability. So we feel that across the heat exchanger the temperature difference measurement is essential. And that's our question.

Another is about the decay heat removal coolers, building cooling units, and standby shutdown facility heat exchangers. We would like to know what is acceptance criteria for the performance testing and what are the bases. And also we know that heat transfer function is needed for the normal operating conditions and also for the accident conditions. Can they do that? And we'd like to know the story.

Also, in what condition do we consider that we should initiate corrective action? On some occasions you mentioned that it's 4 percent above the previous performance testing results, or it's below certain limits. But we'd like to know the limits for all these three, the heat exchangers. And this is another open item.

The third one is about surface model piping corrosion programs. Now we know that certain piping consists of a lot of tubings and pipes. The material is made out of copper, brass, and cast iron, and also the carbon steels. And also the environment, it's raw waters. Now, oh, the makeup of others, degradation mechanism. What, your inspection, what you did, what the licensee did, is try to use the carbon steel components sort of as leading indicators.

Now, that immediately raised some questions. How do you justify to use the carbon steel as a leading indicator and as a result will bounding or the material conditions and/or the degradation mechanisms.

And also one question we'd like to ask is another technical use. It's ultrasonic testing. Now, ultrasonic testing may not be suitable for tests as you test the localized corrosion and microbiologically induced corrosions, and those things may happen to some standard steel, and you may not be able to detect that from the carbon steel testing results.

And most of all, that program does not cover the inspection of the Keowee systems. It's only to the Oconee plants, but not Keowee. Now how to program up all the results from Oconee can bond the Keowee condition. So that's our question.

We have no license renewal issues.

That concludes my presentation.

CHAIRMAN BONACA: Thank you. Any questions?

DR. SHACK: What's different about the treated water system stainless steel inspection and this MIC question you're asking. Are they the same?

MS. COFFIN: That's talking to the service water inspection, which is raw water environment.

DR. KRESS: Okay.

MS. COFFIN: And the treated water is, well, treated water.

DR. SEALE: In these systems that have mixed piping, have you had problems with them so far with corrosion?

MR. ROBINSON: This is Greg Robinson. When you say mixed piping, are you talking about stainless and carbon and --

DR. SEALE: Yes, stainless and carbon and copper especially, or brass.

MR. ROBINSON: The short answer is no. We've had carbon steel issues and we've used stainless steel as a replacement material, and the periodicity of corrosion problems on that stainless is going to be decades --

DR. SEALE: Yes.

MR. ROBINSON: And so we don't expect to see anything for a long, long time, if ever.

DR. SEALE: Yes.

MR. ROBINSON: But, no, we've had no problems --

we have had carbon steel problems, but no problems with the other materials.

MR. GRIMES: This is Chris Grimes. The nature of the open issue isn't -- wasn't driven so much by a question but that we know there have been problems with corrosion of copper or cast iron. It's a reliance on an indicator from findings of carbon steel inspections as --

DR. SEALE: Yes.

MR. GRIMES: Since you're not going to see anything for a decade, are you going to remember that that's the only thing you're measuring in order to make sure that you take appropriate corrective action for these other materials?

DR. SEALE: If it's going to leak, you want to know by how much.

MR. GRIMES: Right. And where.

DR. SEALE: Yes.

MR. GRIMES: And where.

DR. SHACK: Well, I thought it was more the fact that, you know, I mean if it's general corrosion, it's true, I mean stainless and carbon steel are, you know, grossly, but there's nothing that says you can't pit or have MIC attack on the carbon steel, and it has no relation to the general corrosion of the carbon.

MS. COFFIN: It's for the other mechanisms.

That's why the question's there.

DR. UHRIG: Have you had any significant problems with the microbiological induced corrosion?

MR. ROBINSON: No, we have not. This is Greg Robinson. In fact, to my knowledge we have had no indication of problems with MIC at all. We're in the foothills of the Appalachian Mountains, and the water quality is pretty high.

DR. UHRIG: Usually, at least my limited experience has been in water -- in stagnant water in pipes.

MR. ROBINSON: Yes, that would produce a conducive environment for that to occur.

DR. SEALE: Everything grows fast in the water in Florida.

[Laughter.]

DR. UHRIG: This is a Tennessee plant, Bob.

DR. KRESS: South Carolina.

DR. UHRIG: The one that had the MIC.

DR. KRESS: Oh, I'm sorry. I thought you were talking about Oconee.

DR. UHRIG: No, no, it was at the Tennessee plant.

DR. KRESS: Fungus really grows fast.

CHAIRMAN BONACA: Okay. If there are no further questions, I think we exhausted that agenda for today, and I wonder if you have any presenters for tomorrow's items that we can continue.

MR. SEBROSKY: The short answer is yes, we have people for 3.3 and 3.4 that we can move from tomorrow to this evening. As a matter of fact, they're here. So we can go ahead and if you want go ahead and talk about 3.3 next.

CHAIRMAN BONACA: I think we should, items 3.3 and 3.4. Yes, 3.3 being containment structures.

MR. SEBROSKY: We don't have -- after we do the presentation on 3.4, we haven't made any arrangements to bring other people up.

CHAIRMAN BONACA: So we will resume tomorrow after that. Okay.

MR. ASHAR: I am Hansraj Ashar from Division of Engineering and I will be making a short presentation on containment structures as to what the licensee has provided to us in license renewal application. Yes?

DR. SHACK: I think the answer is if you can read it, we can read it.

[Laughter.]

MR. ASHAR: I can read from here. You can read from there.

Before I jump to the open items, I think I would like to say something about what the applicant has provided in the LRA and how the Staff has reviewed it in a very brief manner.

For containment structures at Oconee, the Applicant has grouped components of the containment structure in three groups, the concrete components, the steel components, and the post-tensioning tendon components. In concrete components it includes the dome and cylindrical wall, the basemat and the floor. Steel components includes the liner plate and penetrations including equipment hedge, the access openings, and the other process, the piping, and post-tensioning tendons that includes the wires, the tendons, the anchorage components.

Now for all these components, the applicant has identified the aging effects and based on those aging effects it has provided aging management programs.

There are three programs which the applicant is counting on for managing the aging of containment structures. Containment ISI plan, which is inservice inspection plan for containment, containment program, and the containment leak rate testing program. All these three programs has been evaluated by the Staff in accordance with the 10 elements for evaluating any of the plants or programs which are something like a scope of the program, the preventative actions, parameters monitored, et cetera. There are 10 elements against which we evaluate these types of programs.

One open item -- we believe the application has fulfilled their requirement for aging management of the containment structure components.

Now I will explore a little bit on the open item. It talks about the lack of A&P to manage the aging effects on tendon galleries. Now tendon galleries, I do not know whether all of you are aware of it, where they are and what they are, but the tendon galleries are the bottom of the basemat of the containment structure. They are mainly used for access to the tendon anchorages, so that during the installation also they don't need it and during the inservice inspection they need to go in that area to make sure that they can inspect the grease caps and the anchorage components in the tendon galleries.

Now what we see in the application is that the applicant is not telling us how the tendon galleries will be -- the effect of the degradation on tendon galleries will be managed, and the reason we are not asking for this, because we consider tendon galleries as pressure boundary for containment. We consider it is a nonpressure boundary, however the environment in the tendon galleries does give aging effects on the tendon anchorage components, and we have seen that in a number of plants that the bearing place of some of the anchorages had corroded. We have seen quite an infiltration of water at a number of tendon galleries, and the high humidity in the tendon galleries, and that is why we believe that the most cost-effective way of ensuring that the tendon anchorages degradations are managed well, the basic thing that the applicant has to do is to manage the aging effects on tendon galleries and make sure the environment in the tendon galleries is not conducive to corrosion and degradation of the anchorage components. That is the open item I am talking about.

The license renewal issues -- the license renewal issues on tendons, this is mainly a discussion of temperature effect on tendons. Our experience has shown that the pre-stressing tendon forces in containments have seen more losses than what were estimated at the time the construction was -- when the design was performed, and that is the reason this particular issue came as one of the license renewal issues.

Now the Staff feels that the applicant's ISI plan, which I mentioned earlier, plus an adequate TLAA for tendons, which the applicant has performed to some extent -- we have some problems but that will come under the topic on TLAA, but we believe that the ISI plan and the adequate TLAA would take care of this particular license renewal issue.

The second issue on 98-0049, inaccessible areas, 10 CFR 50.55(a) has a requirement to look for the degradation in the inaccessible area if there are symptoms that indicates that there would be some problems in the inaccessible areas. The basic concern in 98-0049 I believe is related to the other areas which are not being indicated by the accessible areas. Mainly there's the groundwater chemistry that might affect the degradation and aging of the below-grade containment structures.

The applicant response to some of the questions that we asked has provided us with chemical composition of some of the contaminants in the groundwater and it amounts to something like less than 10 ppm of chlorides and close to about 500 ppm or less of sulfates, which are the basic contaminants we think are detrimental to the concrete structures, so we believe that that particular issue for containment has been addressed.

The next license renewal issue is 98-052, which is related operating experience. The applicant has provided data on what has happened to pre-stressing tendons and the liner plate corrosion and the junction of the cylindrical wall and the basemat, that they have seen some corrosion and they have taken corrective actions on that, so the vital things that are necessary for operating experience has been provided in the license renewal application, so we believe it has been very well covered in that.

The next one, in 98-0057, relation to maintenance rule. I have not seen much discussion of this particular issue in application itself, but the way we perceive, the

Staff perceives it is that for maintenance rule, as a matter I had been to a couple of inspections on maintenance rule baseline inspections, in which the licensees in general, not this particular applicant but the licensee in general have taken credit for the ISI plan, which is an implementation of subsection IWE/IWL of ASME Section 11.

They have taken credit for preventive maintenance for the maintenance rule, so I think it applies to this particular application too.

On the subject of the next license renewal issue, 98-0087, which is related to the temperature, actually shield water temperature should go into Section 3.8 but I'll talk only about the containment temperature here.

The applicant has addressed this issue under environmental assessment and its effect on various parts of the containment structures, so the Staff believes that it has been addressed well in the application and we don't have any issue related to that particular license renewal issue.

Now items of interest, as I mentioned before about the TLAA for post-tensioning tendon forces, will be discussed probably tomorrow, in Section 4.22 of this SE.

TLAA for liner and penetration fatigue analysis, that will be discussed in 4.21 of the SE, probably tomorrow. That ends my presentation.

DR. SEALE: You mentioned that the experience with tendons has been less positive than expected, that you have had some relaxations of tendon tension which were more than you would have anticipated?

MR. ASHAR: Yes. The experience does show that at a number of plants the pre-stressing tendons were losing their pre-stressing forces more than what was anticipated or what was estimated during the design of the plant.

At Ginna, the licensee for Ginna had gone through extensive investigation of why that happened and what they found, they sent out the specimen of the wires and tendons to Lehigh University for testing as to why it happened, and the conclusion was that the steel that is being used for pre-stressing tendons is going through much higher relaxation at higher temperatures. In tighter temperatures we are not talking about very high temperatures, we are talking about in the range of 95 degrees instead of 72 degrees -- 95 degrees and 100 degree temperatures.

It was clearly indicated in some of the research that has been done at Lehigh that relaxation losses are occurring at a higher rate than would occur at 72 degrees temperature, for example.

DR. KRESS: How did they determine? Did they retorque the volts or they got crane gauges on it or --

MR. ASHAR: No. What they did was they took the pre-stressing wires from the plant --

DR. KRESS: Oh, they took it out.

MR. ASHAR: They took it out because they had to take it out as part of inspection and part of investigation. They took out the pre-stressing wires. Because they are greased you can take them out if you want, and as a matter of fact, as part of the inservice inspection they have to take one wire out of it in order to inspect the material properties and condition of simple wire. It is part of the inspection requirement.

So they took those wires out and stressed to various levels of pre-stressing force and then they left it for 1000 hours and 10,000 hours kind of timing to see how much it relaxes under various temperature conditions.

DR. KRESS: I see what you are saying. They did it in a lab.

MR. ASHAR: They did it in a lab, yes.

MR. GILL: Bob Gill, Duke. Just a footnote on this. The original tech spec requirements for testing our tendons required us to look at the same tendons periodically specified in the tech specs. Several years ago Staff was reviewing a report that we had made on that tendon testing and strongly suggested we convert to Reg Guide 135, just a random testing sample, and that was about concurrent with the imposition of IWL, the rulemaking that occurred three or four years ago, I forget, sc nave just recently shifted from specified tendons to random tendons, and so we really only have one datapoint with the random testing results, and as we get more data obviously we can do extrapolations in the future.

We have the projected loss based on the data we do have. We projected that out and it is well above the prescribed minimum limit at 60 years. That is in a document called Selected Licensee Commitments, which are a part of our FSAR. We will do the periodic testing and confirm that our actual datapoints are above that. That is part of the application. In fact, the FSAR supplement contains those curves in there and again we are in a transition mode at Oconee from what we had had for 20 some years to the new random selection process, and which tendons do you select, do you select some near main steam pipes which might be a little warmer than others? All that is in the process of evolution as a Part 50 type issue.

We tried to capture that in the renewal application but the target is still evolving in some respects.

DR. SEALE: Thank you.

DR. KRESS: What does containment leak testing tell you about aging of containment? Anything?

MR. ASHAR: Containment leak testing regarding the prestressing force, you mean?

DR. KRESS: No, about aging in general.

MR. ASHAR: Aging in general. Well, Type A test generally tells us the overall leak rate into the containment from the containment structure.

DR. KRESS: Tell you some of the elastomers have --

MR. ASHAR: Elastomers, yes, but that would be more seen in Type B type of test. Type B tests are the ones --

DR. KRESS: Where you go right to the --

MR. ASHAR: Where you locally --

DR. KRESS: Locally go to the --

MR. ASHAR: Pressurize the particular penetrations.

DR. KRESS: Yes.

MR. ASHAR: And try to find out the leakage rates. And there are limits on leakage rates. So when they exceed that leakage rate, then they ought to do something about it, why it's happening. And many times the seals and gaskets might come off it.

MR. GRIMES: This is Chris Grimes. I'd like to add that Appendix J also has requirements in it that speak to performing visual inspections. You basically want to check the condition of the containment before you pump it up so that you don't, you know, inadvertently blow a seal out or something or break light bulbs. So to the extent that Appendix J also provides for a visual inspection and just the setup and performance of the test causes you to have to go check on the condition of the structures, it provides you with an inspection activity that constitutes an opportunity to look for nonconforming conditions.

CHAIRMAN BONACA: Any other questions on containment structures?

[No response.]

If not, I think we have one more presentation on reactor coolant systems.

MS. BANIC: Good afternoon, ladies and gentlemen. My name is Lee Banic, and it's a pleasure for me to be here to discuss our safety evaluation of the reactor coolant system. As coordinator of the review for the Division of Engineering, I'll be making the presentation. Assisting me is Barry Elliot, who had most of the open issues --

[Laughter.]

And generic license renewal issues.

There were ten reviewers who contributed to this section, and many of them are here with me to answer any questions you may have.

Duke described its aging management review of the reactor coolant system in 17 sections of its application. We reviewed these sections to determine whether the effects of aging on the reactor coolant system components will be adequately managed. The components are piping, pressurizer, reactor vessel and internals, steam generators, reactor coolant pumps, control rod drive, tube motor housings, and letdown coolers.

The programs we reviewed were the Alloy 600 program, inspections for the pressurizer, reactor vessel internals, small-bore piping, control rod drive mechanism, nozzle and other vessel closure head penetrations, high-pressure injection connections, reactor vessel integrity, and steam generator tube surveillance.

Duke earlier described its aging management programs for the reactor coolant system in four Babcock & Wilcox owners' group topical reports. These reports were on the reactor coolant system piping, pressurizer, reactor vessel, and reactor vessel internals. We previously approved reports on the piping and pressurizer. We had a few open and action items in our safety evaluation for those reports, and we found that Duke addressed them in its application. We reviewed the reports on the reactor vessel and internals concurrently with Duke's application. We had no open items regarding the reactor vessel. We did have open items for the internals, which we list in our safety evaluation for the application.

We found that except for the open items shown on the slides that Duke has shown that the effects of aging on the reactor coolant system will be adequately managed so that we can make our reasonable assurance finding.

And now for the open items. We had two open items about Duke's identification of aging effects. They're shown on the slide. Duke is to identify that the aging effects for pressurizer spray head are cracking and reduction in fracture toughness due to the thermal aging of cast stainless steel and to provide the basis that void swelling is not an issue for reactor vessel internals or provide an aging management program for it. The staff is concerned that void swelling could change the dimensions of a component and thus its ability to perform its intended function.

We had a number of open items about aging management programs. This first open item relating to inspection of pressurizer components exists because they are made of Alloy 600, an alloy susceptible to primary water stress corrosion cracking.

The next item is open --

DR. SHACK: Is the pressurizer spray head the only internal component that's cast stainless?

MR. ELLIOT: That's the only one identified, yes, so far, on the pressurizer. We have cast -- on the internals there is cast stainless steel.

DR. SHACK: Okay.

MR. ELLIOT: But on the pressurizer -- this is the only one they've identified.

MR. RINCKEL: This is Mark Rinckel from Framatome. That's correct. It's the only cast item in the pressurizer.

MR. ELLIOT: The internals have cast stainless. That's a separate item here.

DR. SHACK: What would be the internals component that would see high enough fluence that you'd worry about the dimensional changes from void swelling?

MR. ELLIOT: It would be -- what's the name of it, whatever's nearest the core.

MR. RINCKEL: It would probably be the baffle plates or baffle bolts. Those receive the highest fluence, some around 10 to the 22, 10 to the 23.

DR. SHACK: I can see them getting stressed perhaps by swelling, but, I mean, what --

MR. RINCKEL: I --

DR. SHACK: Requirement?

MR. RINCKEL: Well, I guess there are questions with that type of fluence. The NRC said that the swelling could be between 4 and 14 percent. I think this is very much a research issue, and, you know, we're certainly looking at responding to that and looking into it. But my understanding is that any dimensional changes would really impact the baffle bolts, and we have a program to look at the baffle bolts, so that's really going to be the focus of our response.

MR. ELLIOT: That you're inspecting the baffle --

MR. RINCKEL: We will at some time. Right now we just do visual of those.

MR. ELLIOT: The intent of this question is to make sure they have a reactor vessel internals program, which you've heard discussed before. We want to make sure that part of that program that they address void swelling.

MS. BANIC: The next item is open because Duke did not identify thermal fatigue as an aging mechanism for the letdown coolers. However, Duke had thermal fatigue damage on four coolers due to operating them in an improper manner. Duke repaired them, but we're asking Duke for information to convince us that the four coolers will not fail again due to thermal fatigue. We had open items about stainless steel components. The first one applies to managing thermal aging of reactor vessel internals, valve bodies, the pressurizer spray head.

The next item applies to reactor vessel internals. Duke is to identify and include limiting wrought stainless steel nonbolting components and welds in internals in its ISI program. This action is necessary to manage the effect of neutron irradiation embrittlement in these components.

The next item is to manage the effects of irradiation-assisted stress corrosion cracking, IASCC, of stainless steel bolting of reactor vessel internals.

The next open item addresses synergistic effects of thermal and neutron embrittlement on the fracture toughness of cast stainless steel internal components.

DR. SHACK: On that one, you seem to have a criterion for the fluence on the cast stainless. Has that really ever been looked at independently or do you just assume that if you've got enough ferrite in it to embrittle it thermally, it's going to embrittle when you irradiate it too?

MR. ELLIOT: Our approach on -- Barry Elliot -- our approach on the cast stainless steel internals is to look at both mechanisms simultaneously. Failure to satisfy either mechanism, whether it be embrittlement or cast -- whether it be neutron embrittlement or thermal embrittlement, if you cannot satisfy the criteria we've written into the safety evaluation, then an augmented or supplementary inspection would be required. So in this case you have to satisfy both criteria. Satisfying one is insufficient.

We've established two criteria. We have a thermal embrittlement criterion in the SER. We have a neutron embrittlement criterion in the SER. And if they can satisfy both those criteria, then they don't need to do any supplementary inspection. If they can't satisfy both criteria, then they would have to do some kind of supplementary inspection.

DR. SHACK: But the neutron embrittlement criteria is basically a fluence level --

MR. ELLIOT: Yes.

DR. SHACK: And then a ferrite level that's essentially equivalent to what you use for the thermal aging. Is that --

MR. ELLIOT: Right. And also we're allowing as part of the neutron embrittlement and thermal embrittlement, if the stresses are very low, if they can show the stresses are very low, then we would --

DR. SHACK: You don't really care.

MR. ELLIOT: We don't really care. That may be one way the spray head can be removed from inspection, for instance, is there's probably very little stresses on the spray head. You know, that would be something they have to look at. We just established a criterion. It's up to them to, once we established a criterion, to convince us that nothing -- no supplementary inspection is required.

MR. RINCKEL: I had one question. In the B&W owners' group RCS piping report, BAW 2043(a), that was approved by the NRC in 1996, you had accepted a different position for the evaluation of CASS valve bodies, and I guess three years have elapsed and things have happened. But the position that you have here is different than what was accepted before. And I wonder if you might just tell us what your thinking is here and what's transpired in three years to lead to this.

MR. ELLIOT: In the last three years, in fact in the last year, the industry has come up with a criterion for evaluating CASS stainless steel. We didn't have that criterion three years ago. We've reviewed that criterion now. We've adopted it as well as provided additional criteria we think should be added to it, and we think that Duke should implement the industrywide criteria at this time

MR. GRIMES: That raises a good point in terms of all of the topical report approvals are subject to verification to make sure that they're still current, and so before we complete the final safety evaluation we'll make sure that the evaluation basis for all the B&W topicals is current.

MR. RINCKEL: I think that's the only one so far that I've seen that there's been a, you know, something -- a different position. So I just wanted to get clarification.

MS. BANIC: This last open item concerns vent valve and retaining rings, which are precipitation-hardened stainless steel being subject to supplemental examination unless Duke can show that loss of fracture toughness from thermal embrittlement and neutron irradiation embrittlement is not significant.

We had no confirmatory items. There were three license renewal issues. As shown on the slide, they are thermal aging of cast stainless steel, vessel surveillance, and internals embrittlement. We cover all of these issues in our safety evaluation. As you have heard, Duke's treatment of the thermal aging of cast stainless steel and internals embrittlement resulted in open issues. We had no issues with the vessel surveillance program. And we had no items of interest.

MR. GRIMES: As they're leaving the table, I'll say are there any questions on section 35?

[Laughter.]

DR. SEALE: We noticed they were pretty slick.

CHAIRMAN BONACA: It was quite a fast move.

Any other questions from any Members here?

[No response.]

There are none, so we thank the staff for the presentations they've given to this point, and we have gained some time for tomorrow morning. We will resume the presentations tomorrow morning with I guess SER Section 3.5, Engineered Safety Features, and also because we gained some time, we will have time for the subcommittee for our deliberation and decisions on what we need to bring to the full committee in September, as well as some topics for the ACRS interim letter.

So with that, we thank the presenters both from Duke and from the staff, and we'll move on to -- we have one hour here right now for us to have some brief discussion c⁻ what we heard, and again we have time tomorrow again at midday.

Chris, could you stick around?

MR. GRIMES: Certainly.

CHAIRMAN BONACA: I think for the following discussion we will go off the record.

[Whereupon, at 3:40 p.m., the meeting was recessed to reconvene at 8:30 a.m., Thursday, July 1, 1999.]

REPORTER'S CERTIFICATE

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

NAME OF PROCEEDING: MEETING: PLANT LICENSE RENEWAL

CASE NUMBER:

PLACE OF PROCEEDING: Rockville, MD

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

Jon Hundley

Jon Hundley
Official Reporter
Ann Riley & Associates, Ltd.

INTRODUCTORY STATEMENT BY THE CHAIRMAN OF THE
PLANT LICENSE RENEWALSUBCOMITTEE
11545 ROCKVILLE PIKE, ROOM T-2B3
ROCKVILLE, MARYLAND
JUNE 30-JULY1, 1999

The meeting will now come to order. This is a meeting of the ACRS Subcommittee on Plant License Renewal. I am Mr. Mario Bonaca, Chairman of the Subcommittee.

ACRS Members in attendance are: Drs. George Apostolakis, Thomas Kress, Robert Seale, William Shack, and Robert Uhrig.

The purpose of this meeting is for the Subcommittee to review the NRC staff's safety evaluation report related to the Oconee license renewal application, crediting of existing programs, and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. Mr. Noel Dudley is the Cognizant ACRS Staff Engineer for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the *Federal Register* on June 1, 1999.

A transcript of this meeting is being kept, and will be made available as stated in the Federal Register Notice. It is requested that speakers first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We have received no written comments or requests for time to make oral statements from members of the public.

On June 16, 1999, the NRC staff completed the safety evaluation report for the Oconee license application. This is the second safety evaluation report for a license renewal application. The report identifies 43 open items that must be resolved for the staff to complete the evaluation. The open items include the basis for excluding specific structures and components from an aging management review; applicability of certain aging effects to structures and components; and the need for additional periodic inspections. The safety evaluation report also identifies six confirmatory items that involve documentation of certain information or commitments. The ACRS plans to review and comment on this safety evaluation report at its September 1999 meeting.

On June 3, 1999, the staff issued a Commission paper identifying options for crediting existing programs for license renewal. The ACRS plans to review and comment on crediting existing programs at its July 1999 meeting. This is one just example of the license renewal policy issues that the staff is evaluating and that the Subcommittee plans on considering.

We will now proceed with the meeting and I call upon Mr. Christopher Grimes, Chief of the License Renewal and Standardization Branch to begin.

PLANT LICENSE RENEWAL
SUBCOMMITTEE

JUNE 30, 1999
AGENDA ITEM III

[redacted]

51-1234566-02

Figure D1 - OC-1 Heatup/Cooldown Cycles per Year

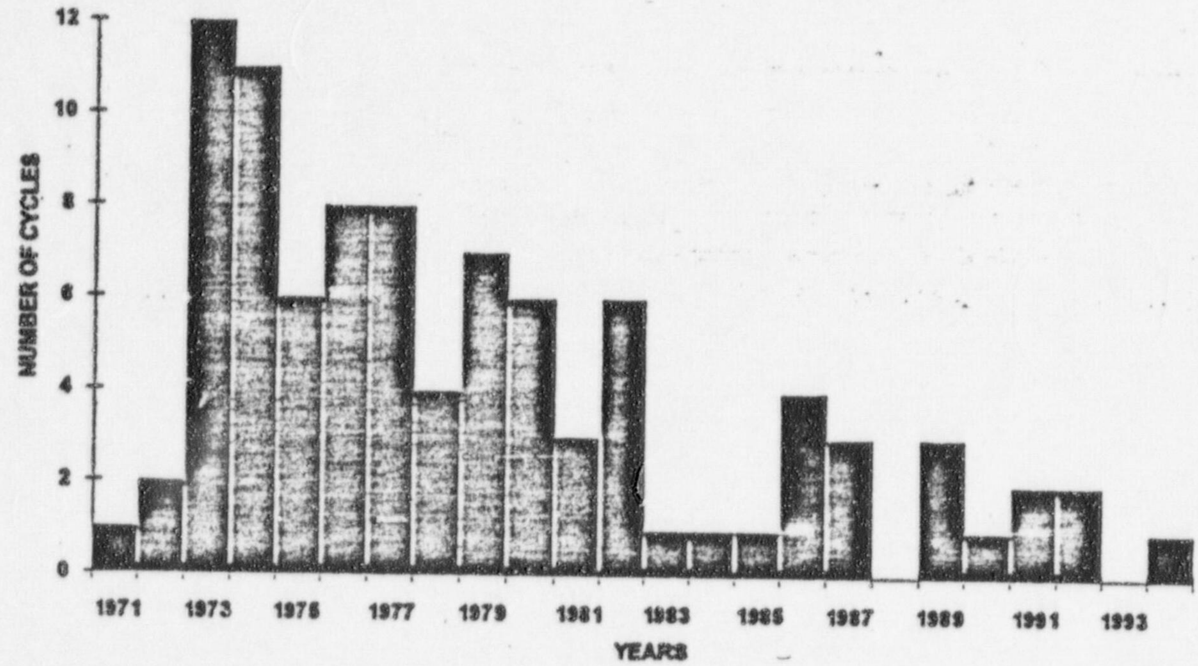
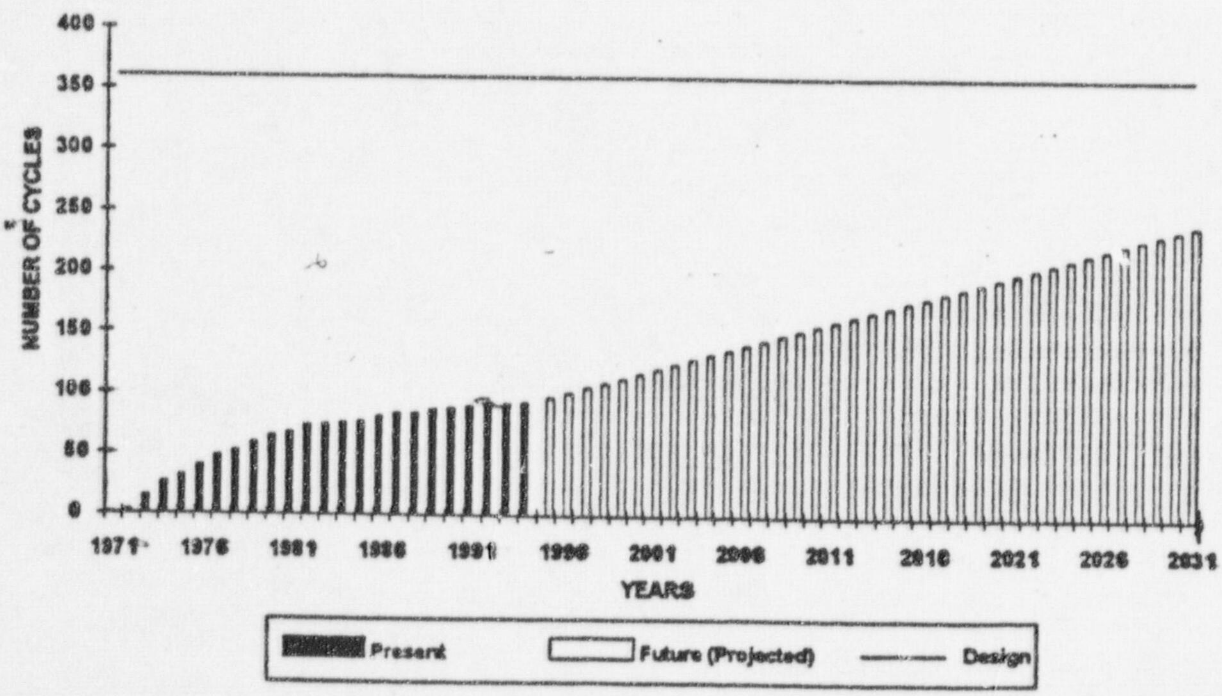
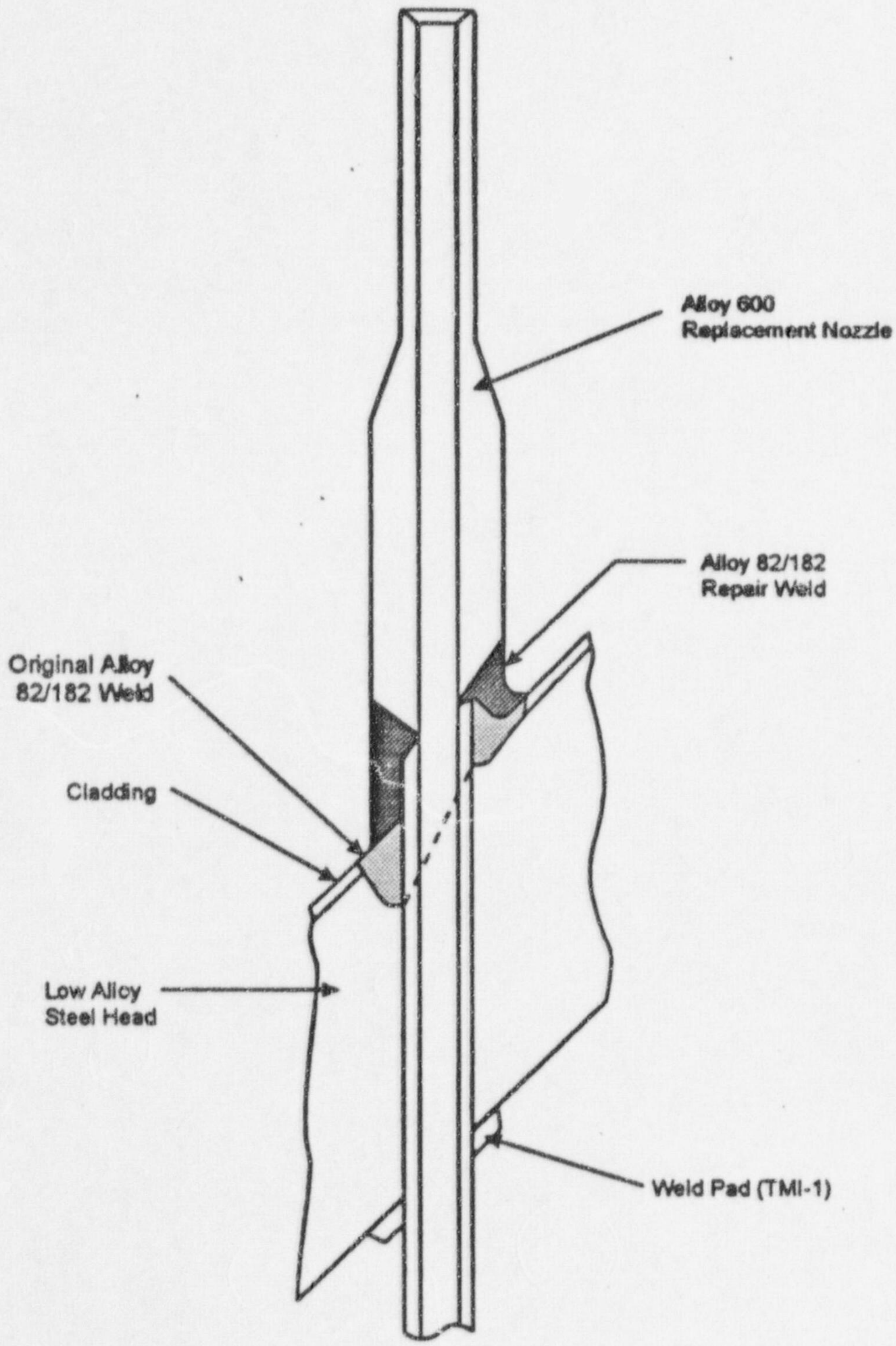



Figure D2 - OC-1 Heatup/Cooldown Cycles Projected for 60 Years



PLANT LICENSE RENEWAL SUB
JUNE 30, 1999
AGENDA ITEM III

**Figure 2-10 Incore Instrumentation System Nozzle**



UNITED STATES NUCLEAR REGULATORY COMMISSION

# ACRS LICENSE RENEWAL SUBCOMMITTEE
## JUNE 30 - JULY 1, 1999

## NRC SAFETY EVALUATION
## RELATED TO BAW-2251


**Barry Elliot**
**DE/EMCB**

# FINAL SAFETY EVALUATION RELATED TO BAW-2251

## "Demonstration of the Management of Aging Effects for the Reactor Vessel"

## OPEN and CONFIRMATORY ITEMS

- NONE

## AGING MANAGEMENT PROGRAMS

- ASME Code, Section XI, Inservice Inspection Program
- Boric Acid Wastage Surveillance Program
- Technical Specification Leakage Limits
- B&WOG Reactor Vessel (RV) Integrity Program

# FINAL SAFETY EVALUATION RELATED TO BAW-2251 (CONT.)

## RENEWAL APPLICANT ACTION ITEMS

- Thirteen plant-specific renewal applicant action items identified

## License Renewal Issues

- 98-0085 - Vessel surveillance
- 98-0028 - Fatigue of metal components

## FINAL SAFETY EVALUATION RELATED TO BAW-2251 (CONT.)

### TLAAs

- Fatigue of metal components
- Pressurized Thermal Shock (PTS) analysis to 10 CFR 50.61 - Neutron fluence analysis contained in Topical Report BAW-2241P
- Charpy Upper Shelf Energy (USE) analysis to Appendix G, 10 CFR Part 50 - Analysis contained in Topical Report BAW-2275
- Growth of intergranular separations in low alloy steel forgings heat-affected zones under stainless steel weld deposit cladding - Analysis contained in Topical Report BAW-2274

## FINAL SAFETY EVALUATION RELATED TO BAW-2251 (CONT.)

### ITEMS OF INTEREST

- Master Integrated Reactor Vessel Material Surveillance Program contained in Topical Report BAW-1543, Revision 4, Supplement 2

- PTS analyses:

  Initially the Oconee 2 upper shell to lower shell circumferential weld $RT_{PTS}$ value was calculated as 304°F, 4°F above the PTS Screening Criteria at the end of the license renewal term.

  As a result of analysis of chemistry data and Charpy impact data (in response to Generic Letter 92-01, Rev. 1, Supplement 1) and revision to the RV neutron fluence, the $RT_{PTS}$ value was reduced to 297°F.

  Additional analyses will be performed as surveillance material is irradiated and tested.

UNITED STATES NUCLEAR REGULATORY COMMISSION

# ACRS LICENSE RENEWAL SUBCOMMITTEE
# JUNE 30 - JULY 1, 1999

# OCONEE LICENSE RENEWAL APPLICATION

# OCONEE LICENSE RENEWAL APPLICATION
# NRC PRESENTERS

| GENERAL TOPICS | PRESENTER |
|---|---|
| Status of License Renewal Activities | Joseph Sebrosky<br>DRIP/RSLB |
| Overview of Safety Evaluation Report (SER) Related to Oconee License Renewal | Joseph Sebrosky<br>DRIP/RSLB |

| SER SECTION AND TOPIC | PRESENTER |
|---|---|
| 2.1 Methodology for Identifying Structures and Components Subject to Aging Management Review (AMR) | Bob Latta<br>DIPM/IQMB |
| 2.2 Identification of Structures and Components Subject to Aging Management Review | Chris Gratton<br>DSSA/SPLB |

# NRC PRESENTERS (CONT.)

| SER SECTION AND TOPIC | PRESENTER |
|---|---|
| 3.1 Applicable Aging Effects for Mechanical Components | George Georgiev<br>DE/EMCB |
| 3.2 Common Aging Management Programs | Shou-Nien Hou<br>DE/EMCB |
| 3.3 Containment Structures | Hansraj Ashar<br>DE/EMEB |
| 3.4 Reactor Coolant System (RCS) | Lee Banic<br>DE/EMCB |
| 3.5 Engineered Safety Features | Stephanie Coffin<br>DE/EMCB |

## NRC PRESENTERS (CONT.)

| SER SECTION AND TOPIC | PRESENTER |
| --- | --- |
| 3.6 Auxiliary Systems | Stephanie Coffin<br>DE/EMCB |
| 3.7 Steam and Power Conversion Systems | Kris Parczewski<br>DE/EMCB |
| 3.8 Structures and Component Supports | David Jeng<br>DE/EMEB |
| 3.9 Electrical Components | Paul Shemanski<br>DE/EELB |
| 4.0 Time-Limited Aging Analyses (TLAA) | Paul Shemanski<br>DE/EELB |

# STATUS OF LICENSE RENEWAL ISSUES

## A. License Renewal Issues

- 108 license renewal issues - mostly NEI comments on the draft Standard Review Plan (SRP)
- Staff proposed resolutions - 9
- Continuing interactions with NEI to clarify issues & develop resolutions

## B. Standard Format for Applications

- NRC letter to NEI - Mar. 15, 1999, transmitted Calvert Cliffs and Oconee Safety Evaluation Report Table of Contents
- Public meeting with NEI - Apr. 13, 1999
- NEI response June 17, 1999

# STATUS OF LICENSE RENEWAL ISSUES (CONT.)

## C. Standard Review Plan, Regulatory Guide (RG), and NEI 95-10

- Draft SRP: Placed in Public Document Room - Sept. 1997
- Draft RG: Published DG-1047 for public comment - Aug. 1996
- NEI 95-10: Issued Mar. 1996. Draft RG proposes to endorse it
- SRP & RG update plan

## OVERVIEW OF OCONEE LICENSE RENEWAL APPLICATION

- **Application dated - July 6, 1998**
- **Staff completed requests for additional information (RAIs) - Dec. 4, 1998**
- **Duke completed RAI responses - Feb. 17, 1999**
- **Public meetings, letters, telephone calls, & site visits used to resolve staff questions**
- **SER issued June 16, 1999 (43 Open Items and 6 Confirmatory Items)**
- **Hearing status - petition denied**
- **Comparison - Oconee and Calvert Cliffs renewal reviews**

## OVERVIEW OF OCONEE LICENSE RENEWAL APPLICATION (CONT.)

### LICENSE RENEWAL INSPECTIONS

| | |
|---|---|
| • Scoping and screening process | Apr. 26-30, 1999 |
| • Aging management review | July 12-16 and 26-30, 1999 |
| • Final verification (optional) | 30 days prior to final licensing action |

### OCONEE SCHEDULE

| | |
|---|---|
| • Duke complete response to open items | Oct. 15, 1999 |
| • Staff issue Safety Evaluation Report | Feb. 12, 2000 |
| • ACRS Final Meeting | May 2000 |
| • Decision on Oconee renewal license | Aug. 2000 |

# PRIORITY 1 LICENSE RENEWAL ISSUES

| ISSUE # | DESCRIPTION | DUKE REVIEW CONSIDERATION |
|---|---|---|
| | | |
| 98-0003 | Operating experience | DUKE provided information |
| 98-0009 | FSAR content | Will follow 10 CFR 50.71(e) guidance |
| 98-0012 | Consumables (Not Priority 1) | Section 2.2 of SER |
| 98-0014 | EQ TLAA | Section 4.2 of SER |
| 98-0016 | Fuses | Section 3.9 of SER |
| 98-0028 | Fatigue (Not Priority 1) | Section 4.2 of SER |
| 98-0030 | Thermal aging of CASS | Section 3.4 of SER |
| 98-0048 | IWE/IWL: Tendons | Sections 3.3 & 4.2 of SER |
| 98-0049 | IWE/IWL: Inaccessible | Section 3.3 of SER |
| 98-0052 | IWE/IWL: Op. exp. | Section 3.3 of SER |

## PRIORITY 1 LICENSE RENEWAL ISSUES (cont.)

| ISSUE # | DESCRIPTION | DUKE REVIEW CONSIDERATION |
|---|---|---|
| | | |
| 98-0057 | Maintenance rule/struct. | Sections 3.3 & 3.8 of SER |
| 98-0068 | Code editions | Duke provided information |
| 98-0082 | Cascading support Sys. | Section 2.2 of SER |
| 98-0085 | Vessel surveillance | Sections 3.4 & 4.2 of SER |
| 98-0087 | Shield wall temp. | Section 3.3 of SER |
| 98-0100 | FERC dams | Section 3.8 of SER |
| 98-0103 | Internals embrittlement | Section 3.4 of SER |
| 98-0105 | Heat exchanger function | Section 2.2 of SER |

## 2.1 METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW

### Open Items

- Applicant to revise response to RAI 2.2-6, to include a description of the process used to identify events for Oconee license renewal scoping and an explanation as to how the 26 events identified during the May 11, 1999, meeting are sufficient to satisfy the requirements of 10 CFR 54.4(a)(1) and 54.4(a)(2).

- Subsequent to the receipt of the above information the staff will determine whether additional inspection activities will be needed to verify the adequacy of the applicants process for identifying structures and components that are within the scope of the rule.

### Confirmatory Items

- None

## 2.1 METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW (CONT.)

### License Renewal Issues

- None

### Items of Interest

- Evaluation of the scoping and screening methodology for Oconee was challenging for the staff because as stated, in part, in the application "...... a list of the Oconee safety-related systems, structures and components, in and of itself, will not meet the intent of 10 CFR 54.4(a)(1)."

## 2.2 IDENTIFICATION OF STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW

### Open Items

- Basis for the recirculated cooling water system not being included within the scope of license renewal.
- Basis for the chilled water system not being included within the scope of license renewal.
- Passive, long-lived skid-mounted equipment are excluded from an AMR.
- Structural sealants, water stops, and expansion joints are excluded from an AMR.
- Staged pumps, electrical cable, and switchgear have been excluded from an AMR.

## 2.2 IDENTIFICATION OF STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW (CONT.)

Open Items (Cont.)

- Provide a plant-specific justification for excluding the Keowee and Turbine Building roofs from an AMR.
- Provide a plant-specific basis for excluding the ventilation sealant material from an AMR.
- Provide a plant-specific justification for excluding fire detector cables from an AMR.

Confirmatory Items

- The applicant should docket information provided during two conference calls regarding pipe segments that provide structural support.

## 2.2 IDENTIFICATION OF STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW (CONT.)

### License Renewal Issues

- 98-0012 - Consumables
- 98-0082 - Cascading
- 98-0105 - Heat exchanger function

### Items of Interest

- System flow diagrams

## 3.1 APPLICABLE AGING EFFECTS FOR MECHANICAL COMPONENTS

### Open Items

- The aging effects discussed and accepted by the staff were not consistently applied by the applicant in the system specific discussions of aging effects.

- The full scope of buried piping and applicable aging management programs could not be identified and evaluated by the staff.

### Confirmatory Items / License Renewal Issues

- None

### Items of Interest

- The applicant has performed a comprehensive review of aging effects and exhaustive identification of aging effects.

## 3.2 COMMON AGING MANAGEMENT PROGRAMS

- **Boric Acid Wastage Surveillance Program**
- **Chemistry Control Program**
- **Duke Quality Assurance Program**

  **Open Item**

  **Commitment needed in either Quality Assurance Program or Updated Final Safety Analysis Report Supplement to apply requirements for corrective actions, confirmation processes, and document control to non-safety related structures and components subject to aging management review.**

- **Fire Protection Program**
- **Inservice Inspection Plan**

## 3.2 COMMON AGING MANAGEMENT PROGRAMS (CONT.)

- Inspection Program for Civil Engineering Structures & Components
- Reactor Coolant System Operational Leakage Monitoring
- Cast Iron Selective Leaching Inspection
- Galvanic Susceptibility Inspection
- Preventive Maintenance Activity Assessment
- Treated Water Systems Stainless Steel Inspection

## 3.2 COMMON AGING MANAGEMENT PROGRAMS (CONT.)

- **Heat Exchanger Performance Testing Activities**

**Open Items**

**Justify why the temperature difference across the standby shutdown facility (SSF) heat exchanger (HX) is not measured.**

**For decay heat removal coolers, reactor building cooling units, and SSF HXs, provide the acceptance criteria for performance testing, provide the basis for such criteria to meet heat transfer demand on normal and accident conditions, and provide/justify criteria to initiate corrective action.**

## 3.2 COMMON AGING MANAGEMENT PROGRAMS (CONT.)

- Service Water Piping Corrosion Program

Open Items

Explain how loss of material is managed for components other than carbon steel (e.g., copper, brass, ductile iron).

Basis for inspections of carbon steel components to serve as leading indicator for conditions of other materials susceptible to different corrosive mechanisms such as pitting and microbiologically influenced corrosion (MIC).

Program does not include inspection of Keowee systems. Justify how components in the Keowee systems are bounded by overall program results.

Justify the use of ultrasonic testing for detecting and sizing localized corrosion in stainless steel.

## 3.2 COMMON AGING MANAGEMENT PROGRAMS (CONT.)

### License Renewal Issues

- None

### Items of Interest

- None

## 3.3 CONTAINMENT STRUCTURES

### Open Items

- Lack of aging management program to manage the aging effects on tendon galleries.

### Confirmatory Items

- None

### License Renewal Issues

- 98-0048 - Tendons
- 98-0049 - Inaccessible Areas
- 98-0052 - Operating Experience (IWE/IWL)
- 98-0057 - Maintenance Rule/Structure
- 98-0087 - Shield Wall Temperature

## 3.3 CONTAINMENT STRUCTURES (CONT.)

**Items of interest**

- **TLAA for liner and penetration fatigue**
- **TLAA for post-tensioning tendon forces**

## 3.4 REACTOR COOLANT SYSTEM

**Use of B&WOG Topicals.**

**Aging Effects - Open Items**

- **Identify that aging effects for pressurizer spray head are cracking and reduction in fracture toughness due to thermal aging of cast stainless steel (CASS).**

- **Provide basis that void swelling is not an issue or provide an aging management program.**

**Aging Management Programs - Open Items**

- **Expand scope of inspection of Unit 1 pressurizer to include Alloy 600 heater sheath-to-sleeve plate and heater sleeve-to-bundle diaphragm plate. State when heater bundle will be removed for examination and basis for scheduling.**

## 3.4 REACTOR COOLANT SYSTEM (CONT.)

### Aging Management Programs - Open Items (Cont.)

- Provide assessment to ensure repaired letdown coolers are operating to preclude failure from thermal fatigue.
- Evaluate CASS components to criteria in EPRI TR-106092 and additional criteria in SER.
- Identify and include limiting wrought stainless steel non-bolting components and welds in internals in ISI program.
- Provide plan to manage aging of baffle-former bolts.
- For loss of fracture toughness from synergistic thermal and neutron embrittlement, perform supplemental examinations/evaluations of CASS internals. Provide estimates of fluence for each CASS component at end of license renewal term, method of determining, justification.

## 3.4 REACTOR COOLANT SYSTEM (CONT.)

### Aging Management Programs - Open Items (Cont.)

- Applicant to submit Pump & Valve In-Service Test Program for staff to determine whether supplemental examination is necessary for vent valve bodies and retaining rings.

### Confirmatory Items

- None

### License Renewal Issues

- 98-0030 - Thermal aging of CASS
- 98-0085 - Vessel surveillance
- 98-0103 - Internals embrittlement

### Items of Interest

- None

## 3.5 ENGINEERED SAFETY FEATURES SYSTEMS

### Open Items

- None

### Confirmatory Items

- Requested documentation that provides the bases for considering the results of the reactor building spray system inspection applicable to the nitrogen purge and blanketing system.

### License Renewal Issues

- None

### Items of Interest

- One-time inspection of the reactor building spray system planned before the end of the current operating license.

## 3.6 AUXILIARY SYSTEMS

### Open Items

- No identification of vibrational loading as causing aging effects for the HVAC system (e.g., cracking and loosening of fasteners).
- Basis for the acceptability of the scope of the reactor coolant pump motor oil collection system inspection.
- Basis for the acceptability of the frequency of sampling for the Keowee oil sampling program.

### Confirmatory Items

- Documentation of operating experience relative to system performance testing.
- Documentation of bases for acceptance criteria for the Keowee oil sampling program.

## 3.6 AUXILIARY SYSTEMS (CONT.)

**License Renewal Issues**

- None

**Items of Interest**

- Two one-time inspections planned before the end of the current operating license (reactor coolant pump motor oil collection system and Keowee air and gas systems inspections).

## 3.7 STEAM AND POWER CONVERSION SYSTEMS

### Open Items

- None

### Confirmatory Items

- None

### License Renewal Issues

- None

### Items of Interest

- Piping Erosion/Corrosion Program

## 3.8 STRUCTURES AND COMPONENT SUPPORTS

Scope of review

- Auxiliary Building
- Earthen Embankments
- Intake Structure
- Keowee structures
- Reactor Building internal structure & vent stack
- Turbine Buildings
- Yard structures
- Class 1 component supports

## 3.8 STRUCTURES AND COMPONENT SUPPORTS (CONT.)

### Open Items

- Aging effects of 183°F temperature on spent fuel pool concrete (cracking and change in properties).
- Identification of applicable aging effects should consider results of Oconee baseline inspection and instances of the reported unusual events (e.g., spent fuel pool liner water leakage).
- Secondary shield wall tendon forces should be monitored for the extended period of operation.
- Inconsistent treatment of aging effect (loss of material) for cable trays and conduits located inside and outside the Oconee containment.

## 3.8 STRUCTURES AND COMPONENT SUPPORTS (CONT.)

### Confirmatory Items

- None

### License Renewal Issues

- 98-0057 - Maintenance Rule/Structure
- 98-0100 - FERC Dams

### Items of Interest

- None

## 3.9 ELECTRICAL COMPONENTS

**Open Items**

- None

**Confirmatory Items**

- None

**License Renewal Issues**

- 98-0016 - Fuses

**Items of Interest**

- None

## 4.0 TIME-LIMITED AGING ANALYSES

Duke identified the following as TLAAs:

- Fatigue analyses for the containment liner plate and penetration
- The loss of prestress in the containment post-tensioning system
- Fatigue and fracture mechanics analyses for ISI reportable indications in the reactor coolant system and Class 1 components
- Neutron embrittlement of the beltline region of the reactor pressure vessel (RPV), including analyses for pressurized thermal shock and Charpy upper-shelf energy reduction. Also, intergranular separation in the heat-affected-zone (HAZ) of low alloy steel under austenitic stainless steel cladding.
- Flow-induced vibration, transient cycle count assumptions, and ductility reduction of fracture toughness for the reactor vessel internals

## 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

**Duke identified the following as TLAAs (Cont.):**

- **Fatigue analysis of the reactor coolant pump flywheel**
- **Fatigue analyses for mechanical components**
- **Environmental qualification of electrical equipment**
- **Fatigue analysis of the polar crane**
- **Aging evaluation of Boraflex in the spent fuel rack**

## 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

### 4.2.1 Containment Liner Plate And Penetrations

**Open Items**

- Provide discussion of the cumulative effects of all possible cycles in the fatigue analysis

**Confirmatory Items**

- Applicant should note that the performance-based Option B allows the 10-year frequency if previous leak rate tests had no problems. Additional leak rate tests may have to be performed after any major modifications (e.g. steam generation replacement).

**License Renewal Issues**

- None

## 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

### 4.2.1 Containment Liner Plate And Penetrations (Cont.)

**Items of Interest**

- Containment leakage testing program
- Containment ISI Plan

## 4.0 <u>TIME-LIMITED AGING ANALYSES (CONT.)</u>

### 4.2.2 Containment Post Tensioning System

**Open Items**

- The trend lines that would demonstrate the adequacy of the existing prestressing forces in the containment tendons have not been shown by the applicant for the period of extended operation.

**Confirmatory Items**

- None

**License Renewal Issues**

- 98-0048 - IWE/IWL: Tendons

**Items of Interest**

- Containment ISI plan

## 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

### 4.2.3 Fatigue Analysis of Reactor Coolant System (RCS)

Open Items

- TLAA of RCS is not adequate to address fatigue concerns for operation beyond 40-years (GSI-190).
- Corrective actions regarding the Section XI flaw evaluation.

Confirmatory Items

- Completion of (1) emergency feedwater nozzle analysis; (2) Class I analysis of the attached piping to the first isolation valve; (3) revised response to NRC Bulletin 88-08 by July 1, 2000.

License Renewal Issues

- 98-0028 - Fatigue of Metal Components

## 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

### 4.2.3 Fatigue Analysis of Reactor Coolant System (Cont.)

**Items of Interest**

- None

# 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

## 4.2.4 Reactor Neutron Embrittlement and Underclad Cracking

Open Items

- None

Confirmatory Items

- None

License Renewal Issues

- 98-0085 - Vessel Surveillance

# 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

## 4.2.4 Reactor Neutron Embrittlement and Underclad Cracking (Cont.)

**Items of Interest**

- Pressurized Thermal Shock
- Charpy Upper Shelf Energy
- Underclad Cracking
- Pressure-Temperature Limits

# 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

## 4.2.5 Reactor Vessel Internals

### Open Items

- Plan to develop data to demonstrate that the internals will meet the deformation limits.
- Applicant did not address the applicability of flaw growth acceptance in accordance with the ASME B&PV code, Section XI ISI requirements.

### Confirmatory Items

- None

### License Renewal Issues/Items of Interest

- None

# 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

## 4.2.6 Fatigue of Reactor Coolant Pump Flywheel

### Open Items

- None

### Confirmatory Items

- None

### License Renewal Issues

- None

### Items of Interest

- None

## 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

### 4.2.7 Fatigue Analysis of Class II and Class III Components

**Open Items**

- None

**Confirmatory Items**

- None

**License Renewal Issues**

- None

**Items of Interest**

- None

## 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

### 4.2.8 Environmental Qualification of Electrical Equipment

**Open Items**

- None

**Confirmatory Items**

- None

**License Renewal Issues**

- 98-0014 - Environmental Qualification (EQ) TLAA

**Items of Interest**

- Oconee evaluated the EQ TLAA while Calvert Cliffs provided a process description of the EQ TLAA.

# 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

## 4.2.9 Fatigue of Polar Crane

### Open Items

- None

### Confirmatory Items

- None

### License Renewal Issues

- None

### Items of Interest

- None

# 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

## 4.2.10 Aging of Boraflex in Spent Fuel Racks

**Open Items**

- None

**Confirmatory Items**

- None

**License Renewal Issues**

- None

## 4.0 TIME-LIMITED AGING ANALYSES (CONT.)

### 4.2.10 Aging of Boraflex in Spent Fuel Racks (Cont.)

**Items of Interest**

- Boraflex Monitoring Program
  - Visually inspects Boraflex coupons.
  - Monitors gap formation in Boraflex panels by blackness testing.
  - Monitors future performance of Boraflex by measuring silica in the spent fuel pool and using the RACKLIFE computer code.

Duke Power[SM]

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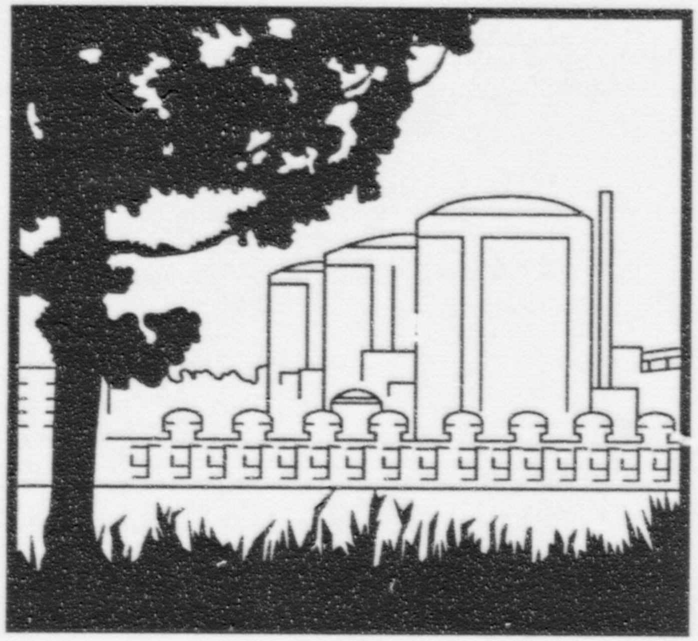
# Oconee
# License Renewal Project



Presentation to:

Advisory Committee on Reactor Safeguards – Plant License Renewal Subcommittee

June 30, 1999

Duke Power
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## Oconee License Renewal Project

*Presentation to:*

*Advisory Committee on Reactor Safeguards – Plant License Renewal Subcommittee*

*June 30, 1999*

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## Presentation Overview

- **Morning Session**
  - Introduction
  - Reactor Vessel Topical Report BAW-2251 Overview
- **Afternoon Session**
  - Application Integrated Plant Assessment and Time Limited Aging Analysis Overview



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## Presentation Participants

- **Morning Session**
  - Greg Robison (Duke Energy)
  - Mark Rinckel (FTI)
  - Matthew Devan (FTI)
  - Ken Yoon (FTI)
  - Bob Gill (Duke Energy)
- **Afternoon Session**
  - Paul Colaianni (Duke Energy)

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## Current Project Status

- Safety
  - Responses to 324 RAIs completed 2/17/99
  - Safety Evaluation Report issued 6/16/99
- Environmental
  - Responses to 39 RAIs completed 3/3/99
  - Draft Oconee Supplemental Environmental Impact Statement issued 5/20/99
- Hearings
  - The Chattooga River Watershed Coalition petitioned to intervene in 9/98
  - The Atomic Safety and Licensing Board denied petition in 12/98
  - The Chattooga River Watershed Coalition appealed the Atomic Safety and Licensing Board decision to the NRC Commission in 1/99
  - The NRC Commission affirmed the Board decision to deny the petition on 4/15/99

*June 30, 1999* *Oconee License Renewal Project* 7

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## License Renewal Technical Information

- Integrated Plant Assessment Process

  C + A + P + D = R.A.

  C = Component
  A = Aging Effects
  P = Program
  D = Demonstration
  R.A. = Reasonable Assurance

- Time-limited Aging Analysis and Exemption Reviews
  - 40 year boundary conditions established in the licensing basis of the plant

*June 30, 1999* *Oconee License Renewal Project* 8

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## License Renewal Technical Information

- The Integrated Plant Assessment (IPA) and Time-limited aging analysis (TLAA) activities associated with plant components are divided along discipline function lines:
  - Reactor Coolant System (B&WOG Reports fit here)
  - Reactor Building Containment
  - Mechanical
  - Electrical
  - Structural

*June 30, 1999* *Oconee License Renewal Project* 9

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## Plant Description

- Oconee Nuclear Station is a 3 Unit Site - 2538 MW total
- Construction finished in early 1970's
- Initial capital cost was approximately $500 million
- Commercial operation began in
  - 1973 - Units 1 & 2
  - 1974 - Unit 3
- Initial licenses expire in 2013 and 2014
- About 1300 people are employed at Oconee

June 30, 1999 Oconee License Renewal Project 4

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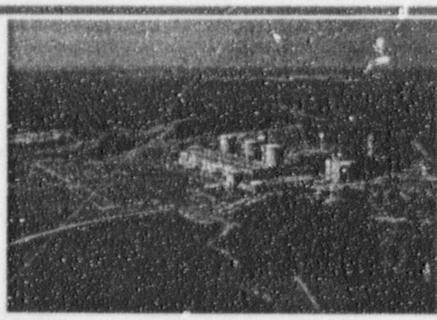
## Oconee Nuclear Station Photograph



June 30, 1999 Oconee License Renewal Project 5

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## Oconee License Renewal Background

| | 1985 | 1990 | 1995 Today |
|---|---|---|---|
| Industry | • Technical Reports<br>• Lead plant reviews<br>• Scientific perspective<br>• Aging mechanisms | • Owners Groups<br>• RGAE<br>• 10 CFR 54 and GEIS issued by NRC | • Application guide<br>• NRC/Utility Demos<br>• Engineering perspective<br>• Aging effects |
| Duke | • Industry Participant<br>• Leader in Technical Report development | Oconee License Renewal Team -<br>• Formed - 1992<br>• Designed Oconee LR Solution<br>• Active in Owners Groups and industry efforts<br>• **July 6, 1998 Application Submittal** | |

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## Reactor Coolant System Components

- Reactor Coolant System Piping (Safety Evaluation 3/21/96)
- Pressurizer (Safety Evaluation 8/18/97)
- Reactor Vessel (Safety Evaluation 4/26/99) → BAW-2251
- Reactor Vessel Internals (Draft Safety Evaluation 5/26/99)
- Once Through Steam Generator
- Class 1 Component Supports
- Reactor Coolant Pumps
- CRDM Pressure Boundary



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## Reactor Vessel Topical Report BAW-2251 Overview

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## B&WOG Reactor Vessel Topical Report (BAW-2251)

**Topics**

- Participating Plants
- Contents of RV Report
  - Scope
  - Aging Effects
  - Demonstration of Aging Management
  - Time Limited Aging Analyses
- Conclusions

## B&W Owners Group Participating Plants

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- ANO-1; Oconee 1, 2, and 3; and TMI-1
- All plants similar in design and age
- B&W designed 177 FA lowered loop plants
- Operating licenses expire between 2013 to 2016



## Reactor Vessel Report Contents

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- Define component intended function(s) that must be protected for period of extended operation (pressure boundary and support of reactor vessel internals)
- Provide description of component including materials of construction
- Define applicable aging effects for material, environment, and stress (Level A and B Service Conditions)
- Determination that aging effects are managed--Credit regulated programs (e.g., ASME Section XI)
- Evaluate applicable Time Limited Aging Analyses (e.g., Fatigue, and Irradiation Embrittlement)



## Reactor Vessel Report Scope

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- Vessel designed in accordance with ASME Section III, 1965 Edition, Addenda through Summer 1967
- Describe RV items in accordance with ASME Section XI, Subsection IWB, Examination Categories
  - RV Shell and Closure Head
  - RV Nozzles – Inlet/Outlet, Core Flood, IMS, and CRDM
  - RV Interior Attachments – Core Guide Lugs
  - RV Pressure – Retaining Bolted Closures (Closure Studs, CRDM Housing Flange Bolting)

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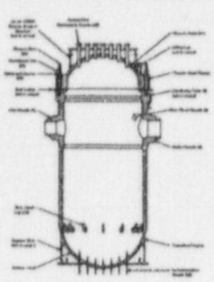
## Reactor Vessel
## Shell and Closure Head



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## Reactor Vessel (RV)
## Aging Effects

- RV Shell and Closure Head
  - Cracking at Welded Joints (Growth Pre-Service Flaws and Fatigue--TLAA)
  - Loss of External Material Near Bolted Closures Owing to Boric Acid Wastage
  - Reduction Fracture Toughness Beltline Region (TLAA)
  - Growth of Intergranular Separations (A508 Class 2 forgings-TLAA)

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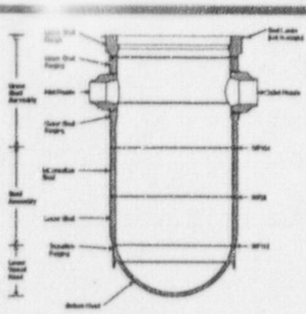
## Reactor Vessel Shell
## Unit 2



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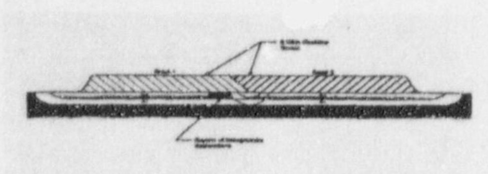
## Intergranular Separations



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## Reactor Vessel (RV) Aging Effects (continued)

- **RV Nozzles**
  - Clad Low -Alloy Steel--Cracking at welded joints, cracking at inside nozzle radius, and loss of external material
  - Alloy 600 (CRDM and IMS)-cracking at or near the HAZ
- **RV Interior Attachments**
  - Alloy 600 Guide Lugs--cracking at or near the attachment welds
- **RV Pressure-Retaining Bolted Closures**
  - Loss of Mechanical Closure Integrity (loss of material, cracking, stress relaxation)

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## Generic Aging Management Programs

- ASME Section XI, Subsection IWB, 1989 Edition (Appendices VII and VIII)
- BWOG Reactor Vessel Integrity Program (Compliance with 10 CFR 50.60 and 10 CFR 50.61)
- Technical Specifications (P-T Limits, Chemistry, Leakage Limits)
- Commitments to NRC Generic Communications (GL88-05, BL82-02)
- GL 97-01Concerning Vessel Head Penetrations-- Alloy 600

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## Generic Time-Limited Aging Analyses

- Thermal Fatigue
- Compliance with 10 CFR 50.60 and 10 CFR 50.61 (MIRVP)
  - $RT_{PTS}$ to 48 EFPY (10 CFR 50.61)
  - Upper-Shelf Energy (10 CFR 50.60)
- Growth of Intergranular Separations
- Flaw growth acceptance in accordance with ASME Section XI ISI (plant-specific)



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## Thermal Fatigue

- Approach for License Renewal
  - Summarize cumulative usage factors for all Class 1 components—including applicable design transients (e.g., heatups and cooldowns)
  - Determine if current number of design transients valid for the period of extended operation
  - Assess impact of environmental-assisted fatigue—if required



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## Summary of Fatigue Design Basis

- Matrices prepared summarizing usage factors and applicable Level A and B transients for all B&W designed Class 1 components
- Controlling transients identified as:
  - Heatups and Cooldowns
  - Reactor Trips
  - HPI Actuations
  - EFW Acttuations
  - Rapid Cooldowns
  - NC Cooldowns

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## Projection of Accrued Cycles

- For controlling transients, assessment made as to number of transients accrued to date for each plant
- Information sources included plant-specific transient logs, transient data, TAP reports, etc
- Projection of controlling transients made to end of period extended operation
- For RV--current design cycles acceptable for period of extended operation



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## Thermal Fatigue

- Demonstrated that existing usage factors, w/exception of ONS studs, remain valid for period of extended operation. ONS studs (CUF=1.04) to be addressed by ONS at time of application.
- Demonstration performed by identification of controlling design transients and projection to end of period of extended operation.
- Program in place at utilities to monitor occurrences of design transients.
- NRC position that environmental assisted fatigue (EAF) should be evaluated for license renewal--not a current day issue.
- EAF evaluated for limiting RV items (CUF < 1.0)



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## Generic Time-Limited Aging Analyses

- Thermal Fatigue
- **Compliance with 10 CFR 50.60 and 10 CFR 50.61 (MIRVP)**
  - Upper-Shelf Energy (10 CFR 50.60)
  - $RT_{PTS}$ to 48 EFPY (10 CFR 50.61)
- Growth of Intergranular Separations

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## Master Integrated Reactor Vessel Surveillance Program (MIRVP) – NRC Requirements

- 10 CFR 50.60 Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation
  - 10 CFR 50 Appendix G Fracture Toughness Requirements
  - 10 CFR 50 Appendix H RV Material Surveillance Program
- 10 CFR 50.61 Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

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## MIRVP – NRC Requirements

- **10 CFR 50 Appendix G**
  - $C_V$USE not less than 50 ft-lbs (unless equivalent margin of safety per Section XI, Appendix G)
  - Pressure-temperature limits
    - $\Delta RT_{NDT}$ (Regulatory Guide 1.99, Revision 2)
- **10 CFR 50 Appendix H**
  - ASTM E185
  - Approved withdrawal schedule
  - Integrated program rules
    - similar design and operating features
    - adequate dosimetry program
    - data sharing arrangement

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## B&W Fabricated Vessels

- Vessels designed by B&W and Westinghouse
- Material: Mn-Mo-Ni steels
  - Plate SA-302B (early)
  - Plate SA-533B (later)
  - Forgings A 508, Class 2
- Welds
  - Plate: axial and circumferential
  - Forgings: circumferential only
  - Automatic submerged arc (ASA)
  - Linde 80 flux: Low initial CVUSE
  - Copper-coated weld wire: Accelerated reduction of fracture toughness

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## B&W Fabricated Vessels

- Weld Identification
  - Each combination of wire heat and flux lot uniquely identified and qualified
  - Welds identified by WF-numeral (qualified in Mount Vernon) or SA-numeral (qualified in Barberton)
  - All weld seams traceable to a WF- or SA- identifier
- Surrogate Welds
  - Welds fabricated from the same wire heat, even if flux lots are different, have equivalent copper content and properties



## Surveillance Capsules

- Material
  - Base and weld metal
  - Early capsules: Weld metal not same WF/SA as in vessel beltline (requirement changed to same as limiting weld metal in later ASTM E185 Standard Editions)
- Test Specimens
  - Charpy V-notch impact, Tension Test, and Compact Fracture
    - No compacts in very early plant-specific capsules
    - 0.5T in some plant-specific capsules
    - 1T (round) in later B&W Owners Group capsules
- Neutron Dosimeter Wires
- Temperature Monitors—Fusible alloy wires



## Reactor Vessel Integrity Program

- Established in 1977
- Resolve fracture toughness concerns with Linde 80 ASA weld metal
- Original participant plants (B&W-designed): ANO-1; Crystal River-3; Davis-Besse; Oconee-1, -2, and -3; Rancho Seco; TMI-1 and -2
- Later participant plants (Westinghouse-designed): R. E. Ginna; Point Beach-1 and -2; Surry-1 and -2; Turkey Point-3 and -4; and Zion-1 and -2

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## Reactor Vessel Integrity Program

- Obtain materials and irradiation effects data
- Develop test methods and analytical procedures for determination of fracture toughness
- Provide effective communication
  - Among all owners
  - With NRC
  - With industry groups

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## Master Integrated Reactor Vessel Surveillance Program (MIRVP)

- Plant-specific capsules had deficiencies:
  - Limiting materials not in plant-specific capsules
  - Fracture toughness specimens not in plant-specific capsules
- Adding to the plant-specific RVSP irradiation capsules,14 capsules were fabricated and inserted in power reactors

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## Master Integrated Reactor Vessel Surveillance Program (MIRVP)

- B&W kept nozzle dropouts which contain a weld seam. This was the only remaining source of many beltline weld materials. These nozzle dropout weld materials were used to fabricate specimens for irradiation in the 14 capsules.
- Report BAW-1543 describes the MIRVP and is revised regularly to reflect program changes. NRC issued SER accepting the program. Requests for information in the SER were answered with no further NRC comment.

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## MIRVP Current Activities

- Post-irradiation testing of the W1 Capsule has been completed. The W1 Capsule provides Linde 80 weld metal data irradiated in a Westinghouse-designed reactor. Evaluation of the irradiated data for Capsule W1 and the other MIRVP capsules containing the same weld materials is currently being performed as part of the 1999 B&WOG Reactor Vessel Integrity Program.
- Post-irradiation testing and evaluation of the A5 Capsule has been completed. The A5 Capsule contained previously irradiated/reconstituted Charpy impact specimens to acquire high-fluence information.

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## Reactor Vessel Beltline Fluence

- Fluence of RV Beltline Region at 48 EFPY used to calculate $RT_{PTS}$ and USE
- Fluence monitoring using calculational-based methodology and cavity dosimetry
- Uncertainties in fluence estimates addressed through BAW-2241
- Plant-specific monitoring and updates to ensure evaluations in BAW-2251 remain valid

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## Generic Time-Limited Aging Analyses

- Thermal Fatigue
- Compliance with 10 CFR 50.60 and 10 CFR 50.61 (MIRVP)
  - $RT_{PTS}$ to 48 EFPY (10 CFR 50.61)
  - Upper-Shelf Energy (10 CFR 50.60)
- Growth of Intergranular Separations

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## $RT_{PTS}$ to 48 EFPY (10 CFR 50.61) Appendix A to BAW-2251

- Demonstrated that $RT_{PTS}$ values at 48 EFPY comply with requirements of 10 CFR 50.61 using Regulatory Guide 1.99 Revision 2
- $RT_{PTS}$ welds all participating units were calculated to be below PTS limits, with exception of weld WF-25 in Oconee Unit 2 RV and one weld at another plant
- Subsequent plant-specific analyses performed for Oconee Unit 2 result in $RT_{PTS}$ of 296.8 °F

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## Low Upper-shelf Toughness Fracture Mechanics Analysis

- Analysis Methods and Acceptance Criteria
  - Appendix K, Section XI
    ASME Boiler & Pressure Vessel Code
  - Technical Basis
    Welding Research Council Bulletin 413, July 1996
    "Development of Criteria for Assessment of Reactor Vessels with Low Upper Shelf Fracture Toughness"

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## Low Upper-shelf Toughness Fracture Mechanics Analysis

- All four Owners Groups completed this analysis for all reactor vessels for 40 year design life
- B&W Owners Group Report BAW-2275 addresses low upper-shelf toughness fracture mechanics analysis of B&W designed reactor vessels for 48 EFPY

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## Low Upper-shelf Toughness Fracture Mechanics Analysis

- Acceptance Criteria
  - Service Levels A and B:
    - Flaw depth a = t/4 interior semi-elliptical surface flaw
    - Applied J for pressure equals 1.15 times Pa and thermal load:

      < J material at da = 0.1 inch

    - Flaw extension at J for pressure equals 1.25 times Pa and thermal load shall be ductile and stable
    - J material shall be conservative representation for the vessel material under evaluation

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## Low Upper-shelf Toughness Fracture Mechanics Analysis

- Service Level C:
  - Flaw depth a = t/10 interior semi-elliptical flaw
  - Applied J < J material at da = 0.1 inch
  - Flaw extensions shall be ductile and stable using SF =1.0
  - J material shall be conservative representation for the vessel material
- Service Level D:
  - Same as Level C above except J material shall be a best estimate representation of the vessel material

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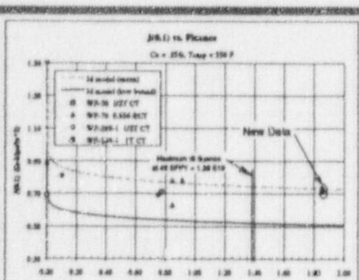
## Low Upper-shelf Toughness Fracture Mechanics Analysis

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## Low Upper-shelf Toughness Fracture Mechanics Analysis

- Analysis Results
  - All participating B&W Owners Group Reactor Vessels were found acceptable by the acceptance criteria of ASME Section XI, Appendix K



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## Generic Time-Limited Aging Analyses

- Thermal Fatigue
- Compliance with 10 CFR 50.60 and 10 CFR 50.61 (MIRVP)
  - Upper-Shelf Energy (10 CFR 50.60)
  - $RT_{PTS}$ to 48 EFPY (10 CFR 50.61)
- **Growth of Intergranular Separations**



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## Intergranular Separations-- Appendix C to BAW-2251

- Demonstrated acceptability growth of postulated flaws to 48 EFPY using ASME Section XI (1989 Edition), IWB-3612, acceptance criteria
- Linear elastic fracture mechanics
- For Service Levels A through D, the analytical results demonstrate sufficient margin beyond that required by the acceptance criteria of IWB-3612

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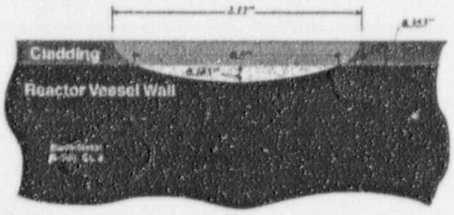
Fracture Mechanics Analysis of Underclad Cracks

## Postulated Cracks

- Postulated Cracks in SA-508 Class 2 Materials in Reactor Vessels



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Fracture Mechanics Analysis of Underclad Cracks

## Section XI Flaw Evaluation

- ASME B & PV Code $K_{Ic}$ and $K_{Ia}$
- Regulatory Guide 1.99 $RT_{NDT}$
- ASME Section XI, Appendix A Fatigue Crack Growth
- Raju Newman $K_I$ Equations



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Fracture Mechanics Analysis of Underclad Cracks

## Postulated Cracks

- Normal and Upset Condition – 19 Transients
- Design Basis Transients from Functional Specification
- Stresses in Selected Regions (Figure 3-3 in Appendix C of BAW-2251)
- 5 Fatigue Groups

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## Fracture Mechanics Analysis of Underclad Cracks
## Conclusions

- Postulated flaws are found acceptable per Section XI Flaw Evaluation Procedures for:
  - 48 EFPY fluence
  - Design base transients and duty cycles

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## Overall BAW-2251 Conclusions

- Demonstrated that aging of the RV will be adequately managed to ensure component intended function(s) during the period of extended operation (40 to 60 years) — AMR and TLAA
- The RV report has been built on experience and methodologies developed over past 20 years in responding to RV issues as well as aging management review experience gained on RCS piping and pressurizer reports
- License renewal applicants must comply with LR applicant action items when referencing the reports

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## Incorporation of BAW-2251 into the Oconee Application for Renewed Operating Licenses

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## Incorporation of BAW-2251 into the Oconee Application

### Topics

- Oconee Application for Renewed Operating Licenses
- Process to Incorporate All Approved B&WOG Topicals into the Application
- BAW-2251 Renewal Applicant Action Items and Oconee-specific responses
- Aging Management Programs for the Oconee Reactor Vessels
- Time-Limited Aging Analyses for the Oconee Reactor Vessels

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## Oconee Application for Renewed Operating Licenses

- Section 2.4.5 describes the Reactor Vessel Components that are Subject to Aging Management Review
- Section 3.4.5 Identifies the Applicable Aging Effects for these Components
- Aging Management Programs are described in Chapter 4
- Section 5.4 Identifies and Provides the Evaluations of the RCS Time-Limited Aging Analysis
- Approved B&WOG Topical Reports are referenced in each of these locations of the Application

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## Renewal Applicant Action Items for BAW-2251

- Presented Item-by-Item in a two-column format
- Oconee-specific response to BAW-2251 Items were provided by letter dated May 10, 1999
- For BAW-2251, there are 13 Renewal Applicant Action Items

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## Summary of Renewal Applicant Action Items for BAW-2251

- Verify that Oconee is bounded by the Topical Report
- Verify that Oconee programs and activities are the same as those credited within the Topical Report
- Perform the plant-specific TLAA identified
- Provide summary descriptions in the UFSAR Supplement

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## Aging Management Programs for the Oconee Reactor Vessels

- Reactor Vessel Integrity Program
  - Master Integrated Reactor Vessel Surveillance Program
  - Cavity Dosimetry Program
  - Fluence and Uncertainty Calculations
  - Pressure-Temperature Limit Curves
  - Effective Full Power Years
- CRDM and Other Vessel Closure Penetration Inspection Program
- Chemistry Control Program
- Alloy 600 Aging Management Program
- Inservice Inspection Plan
- Boric Acid Wastage Surveillance Program
- RCS Operational Leakage Monitoring
- Thermal Fatigue Management Program

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## Time-Limited Aging Analyses for the Oconee Reactor Vessels

- Thermal Fatigue
  - Managed by Thermal Fatigue Management Program
- Flaw Growth Analyses
  - One Oconee Unit 1 RV indication is being addressed
- Pressurized Thermal Shock
  - Fluence projection updated using BAW-2241P methods
  - Chemistry data updated using BAW-2325
  - RT PTS of all three Oconee vessels are less than screening criteria
- Charpy Upper Shelf Energy
  - Bounded by BAW-2251
- Intergrannular Separation in HAZ
  - Bounded by BAW-2251

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## Oconee Reactor Vessel Conclusions

- Oconee Reactor Vessels are bounded by Topical Report BAW-2251
- Oconee programs will continue to effectively manage aging of the Reactor Vessels
- Plant-specific time-limited aging analyses have been evaluated for 60 years of operation

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## End of the Morning Session

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## Integrated Plant Assessment (IPA) and Time-Limited Aging Analysis (TLAA) Overview

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## Oconee
## Integrated Plant Assessment (IPA)

- **Rule of Thumb:**
  - *The discipline boundaries, with few exceptions, are as follows: if a component or parts of it carry electrical current, it is Electrical; if it supports, protects or restrains the movement of a component, it is Structural; everything else is Mechanical. Exceptions to this are noted where applicable.*
- **Scoping components in each discipline:**
  - Structural - relies on Oconee CLB definition
  - Mechanical - works through a functional review process
  - Electrical - uses an encompassing approach



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## Oconee
## IPA & TLAA Reviews

- Reactor Coolant System Components
- Reactor Building Containment Components
- Structures & Structural Components
- Mechanical Components
- Electrical Components



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## Oconee
## IPA & TLAA Reviews

- **IPA Scoping and Screening**
  - **Structures and Structural Components**
  - **Mechanical Systems and Components**
  - **Electrical Components**
- IPA Aging Management Review
  - Structural/Mechanical/Electrical Components
- TLAA Reviews
- Programs and Activities Credited for License Renewal

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## Oconee Scoping & Screening Methodology

- Structural component and mechanical component methodologies are consistent with the guidance provided in NEI 95-10 Rev. 0.
- Electrical component methodology follows the §54 requirements and the guidance provided in the Statement of Considerations (SOC) published with the Final License Renewal Rule and is generally consistent with the guidance provided in NEI 95-10 Rev. 0.



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## Structure Scoping & Screening Methodology

- Structural scoping and screening Includes:
  - Identification of structures and structural components within the scope of the Rule and their intended functions
  - Identification of structures and structural components subject to an aging management review



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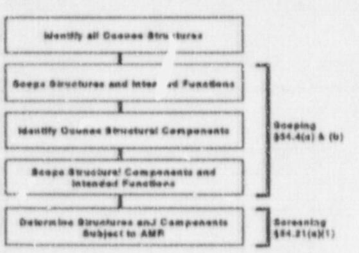
## Structure Scoping & Screening Process



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## Structure Scoping Summary (Example)

[illegible]



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## Structure Scoping Results

- **Auxiliary Buildings** (includes Spent Fuel Pools)
- **Earthen Embankments** (includes Intake Canal Dike, Keowee Dam, Little River Dam and Dikes)
- **Intake Structure**
- **Keowee Structures** (includes Breaker Vault, Intake Structure, Penstock, Power House, Service Bay Structure, and Spillway)
- **Reactor Buildings** (includes Internal Structures and Unit Vents)
- **Standby Shutdown Facility**
- **Turbine Buildings** (includes Switchgear Enclosures)
- **Yard Structures** (includes 230 kV Relay House, Switchyard Structures, Trenches, Towers, Elevated Water Storage Tank, Transformer Pads)



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## Structure / Component Matrix (Example)

[illegible]

### Mechanical Component Scoping Methodology

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- Mechanical scoping and screening process includes:
  - Identification of systems within the scope of the Rule and their system intended functions
  - Identification of components subject to an aging management review and their component intended functions



### Mechanical System §54.4(a)(1) and (a)(2) Scoping

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- Mechanical §54.4(a)(1) and (a)(2) scoping was performed using a four-step process
  1. Functional flow path identification using event mitigation calculations
  2. Fluid pressure boundary determination
  3. Physical interference identification
  4. Other designated item identification
- Evaluation boundaries are highlighted on mechanical system flow diagrams



### Oconee License Renewal Mechanical Scoping Events Set

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- Loss of Main Feedwater (Scoping)
- Loss of Offsite Power (LOOP) (DBE)
- Turbine Trip (DBE)
- Loss of Electric Power (DBE)
- Large Break Loss of Coolant Accident with LOOP (DBE)
- Steam Generator Tube Rupture (DBE)
- Main Steam Line Break (DBE)
- Fuel Handling Accident (DBE)
- Moderator Dilution Accident (DBE)
- Post Accident Hydrogen Control (DBE)
- Control Rod Misalignment (DBE)
- Cold Water Accident (DBE)
- Locked Rotor / Sheared Shaft (DBE)
- Loss of Coolant Flow (DBE)
- Startup Accident (DBE)
- Small Break Loss of Coolant Accident (DBE)
- Maximum Hypothetical Accident (DBE)
- Rod Ejection Accident (DBE)
- Uncompensated Operating Reactivity Change (DBE)
- Waste Gas Tank Rupture (DBE)
- Rod Withdrawal Accident (DBE)
- SSF Event Turbine Building Flood (Scoping)
- Natural Phenomena: Tornado, Wind & Hurricane (includes Tornado Missile) (Scoping)
- Natural Phenomena: Seismic (Scoping)
- Natural Phenomena: External Floods & Ground Water (Scoping)
- Natural Phenomena: Snow & Ice (Scoping)

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## Mechanical System §54.4(a)(3) Scoping

- Mechanical systems which satisfy §54.4(a)(3) criteria were identified through review of licensing commitments for the regulated events.
  - Fire Protection
  - Environmental Qualification
  - Pressurized Thermal Shock
  - Anticipated Transient without Scram
  - Station Blackout



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## Mechanical Component Screening Process

- A menu of mechanical component commodity groups installed in the plant was developed, which included components identified in NEI 95-10 and components not listed in NEI 95-10.
- Using the "passive" and "long-lived" definitions and guidance, mechanical components meeting these screening criteria that are within the highlighted portions of the license renewal flow diagrams are identified and listed.
- This list identifies the mechanical components subject to aging management review.



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## Mechanical System Scoping Results

**Oconee Systems**
- Auxiliary Building Ventilation
- Auxiliary Service Water
- Breathing Air
- Chemical Addition
- Component Cooling
- Condensate
- Condenser Circulating Water
- Containment Hydrogen Control
- Control Room Pressurization and Filtration
- Coolant Storage
- Core Flood
- Demineralized Water
- Electro-Hydraulic Control
- Emergency Feedwater
- Feedwater
- Filtered Water
- Gaseous Waste Disposal
- High Pressure Injection
- High Pressure Service Water
- Instrument Air
- Leak Rate Test
- Liquid Waste Disposal
- Low Pressure Injection
- Low Pressure Service Water
- Lube Oil
- Main Steam
- Nitrogen Purge and Blanket
- Penetration Room Ventilation
- Post Accident Monitoring
- Reactor Building Cooling
- Reactor Building Purge
- Reactor Building Spray
- Reactor Coolant
- Spent Fuel Cooling
- Turbine Building Sump
- Vacuum

**SSF Systems**
- Air Intake and Exhaust
- SSF Auxiliary Service Water
- Diesel Engine
- Diesel Jacket Water Cooling
- SSF Drinking Water
- SSF Fuel Oil
- Governor
- SSF HVAC
- Diesel Lube Oil
- Reactor Coolant Makeup
- SSF Sanitary Lift
- Starting Air

**Keowee Systems**
- Carbon Dioxide
- Fire Protection/Detection
- Depressing Air
- Drain
- Emergency Power
- Generator
- Generator High Pressure Oil
- Governor Air
- Governor Oil
- Main Turbine
- Service Water
- Turbine Generator Cooling Water
- Turbine Guide Bearing Oil
- Turbine Sump Pump

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## Mechanical Component Screening Results (Example)

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## Electrical Component Scoping & Screening Methodology

- Basic Philosophy
  - Except for specific components that are scoped-out or screened-out, all plant electrical components are included in the aging management review.
- Process
  - §54.4(a) scoping criteria is applied only to specific groups of electrical components that are scoped-out.
  - §54.21(a)(1)(i) screening criteria is applied to all electrical component commodity groups.
  - §54.21(a)(1)(ii) screening criteria is applied only to specific electrical components that are screened-out.

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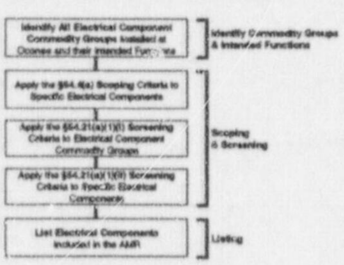
## Electrical Component Scoping & Screening Process

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## Identification of Electrical Component Commodity Groups

| | | | |
|---|---|---|---|
| Alarm Units | Fuses | Meters | Solenoid Operators |
| Analyzers | Generators | Motor Control Centers | Solid-State Devices |
| Annunciators | Heat Tracing | Motors | Surge Arrestors |
| Batteries | Heaters | Nonsegregated-Phase Bus | Switches |
| Chargers | Indicators | Power Distribution Panels | Switchgear |
| Circuit Breakers | Insulated Cables and Connections | Power Supplies | Switchyard Bus |
| Converters | | Radiation Monitors | Thermocouples |
| Communication Equipment | Insulators | Recorders | Transducers |
| Electrical Controls and Panel Internal Component Assemblies | Inverters | Regulators | Transformers |
| | Isolated-Phase Bus | Relays | Transmission Conductors |
| | Isolators | RTDs | Transmitters |
| Electrical Penetration Assemblies | Light Bulbs | Segregated-Phase Bus | Uninsulated Ground Conductors |
| | Load Centers | Sensors | |
| Elements | Loop Controllers | Signal Conditioners | |



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## Application of §54.21(a)(1)(i) Screening Criteria to Electrical Components

| Reference Document | §54.21(a)(1)(i) Criteria are Met | Electrical Component Commodity Groups |
|---|---|---|
| §54.21(a)(1)(i) | Yes | electrical cables and connections (i.e., insulated cables and connections, transmission conductors, uninsulated ground conductors), electrical penetration assemblies |
| | No | batteries, chargers (e.g., battery chargers), circuit boards and transistors (i.e., solid-state devices), breakers (i.e., circuit breakers), generators (e.g., diesel generators), indicators (e.g., pressure indicators, water level indicators), inverters (e.g., power inverters), motors, power supplies, relays, switches, switchgear, transmitters (e.g., pressure transmitters) |
| WD-SRP-LR Table 2.1-2 (based on NEI 95-10 Rev. 0 Appendix B) | No | alarm units, analyzers, annunciators, converters (e.g., voltage/current, voltage/pneumatic), electrical controls and panel internal component assemblies, elements (e.g., conductivity elements, flow elements), isolators, load centers, loop controllers (e.g., differential pressure indicating controller, flow indicating controller, temperature controller, speed controller), meters (e.g., ammeter), motor control centers, power distribution panels, radiation monitors, recorders, RTDs, sensors (e.g., temperature sensors, radiation sensors), signal conditioners, solenoid operators, thermocouples, transducers (e.g., watt transducer) |
| September 19, 1997, NRC letter to NEI | No | heat tracing, heaters, indicating lights (i.e., light bulbs), transformers |
| Oconee License Renewal Application | Yes | insulators, isolated-phase bus, nonsegregated-phase bus, segregated-phase bus, switchyard bus |
| | No | communication equipment, fuses, regulators, surge arrestors |

"No" in the second column indicates that the identified electrical component commodity groups are screened-out.



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## Listing Electrical Components Included in the AMR

| Electrical Component Commodity Groups | Included in the AMR | Intended Functions |
|---|---|---|
| Insulated Cables and Connections | An encompassing group of all insulated cables and connections except those (1) associated with the 525kV Switchyard, Radwaste Facility, or Oconee Retail substation, (2) included in the EQ program, or (3) used for fire detection is included in the AMR. | To electrically connect two sections of an electrical circuit. |
| Insulators | An encompassing group of all insulators except those associated with the (1) 525kV Switchyard, (2) Jocassee, Calhoun, Oconee and Dacus 230kV transmission lines, or (3) Oconee Retail substation is included in the AMR. | To insulate and support an electrical conductor. |
| Isolated-Phase Bus | An encompassing group of all isolated-phase bus is included in the AMR. | To electrically connect two sections of an electrical circuit. |
| Nonsegregated-Phase Bus | An encompassing group of all nonsegregated-phase bus is included in the AMR. | To electrically connect two sections of an electrical circuit. |
| Segregated-Phase Bus | An encompassing group of all segregated-phase bus is included in the AMR. | To electrically connect two sections of an electrical circuit. |
| Switchyard Bus | An encompassing group of all switchyard bus except bus associated with the (1) 525kV Switchyard, or (2) Oconee Retail substation is included in the AMR. | To electrically connect two sections of an electrical circuit. |
| Transmission Conductors | An encompassing group of all transmission conductors except those associated with the (1) 525kV Switchyard, (2) Jocassee, Calhoun, Oconee and Dacus 230kV transmission lines, or (3) the Oconee Retail substation is included in the AMR. | To electrically connect two sections of an electrical circuit. |

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## Oconee IPA & TLAA Reviews

- IPA Scoping and Screening
  - Structures and Structural Components
  - Mechanical Systems and Components
  - Electrical Components
- **IPA Aging Management Review**
  - **Structural/Mechanical/Electrical Components**
- TLAA Reviews
- Programs and Activities Credited for License Renewal

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## Oconee Aging Management Review Process

Identify Component Materials
\+ Identify Component Environments
\+ Identify Potential Aging Effects

= Determine Applicable Aging Effects
(those aging effects that if left unmanaged would cause a loss of intended function before the end of the extended period of operation)

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## Oconee IPA & TLAA Reviews

- IPA Scoping and Screening
  - Structures and Structural Components
  - Mechanical Systems and Components
  - Electrical Components
- IPA Aging Management Review
  - Structural/Mechanical/Electrical Components
- **TLAA Reviews**
- Programs and Activities Credited for License Renewal

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## Time-Limited Aging Analyses

- Involve plant-specific design analyses
- Focus on boundary conditions or assumptions based on 40-year operating term
- Action: Assure analyses are valid for the extended period of operation or that the effects of aging will be adequately managed for 60 years

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## Time-Limited Aging Analyses

- Oconee specific TLAA have been identified by reviewing the Oconee UFSAR, docketed correspondence and NSSS Topical Reports
- The resultant list includes EQ, fatigue, tendon loss of prestress, reactor vessel embrittlement, etc.
- No Oconee exemptions were based on TLAA
- The TLAA process is consistent with the guidance of NEI 95-10 and provides reasonable assurance that Oconee specific TLAA have been identified and evaluated

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## Oconee IPA & TLAA Reviews

- IPA Scoping and Screening
  - Structures and Structural Components
  - Mechanical Systems and Components
  - Electrical Components
- IPA Aging Management Review
  - Structural/Mechanical/Electrical Components
- TLAA Reviews
- **Programs and Activities Credited for License Renewal**

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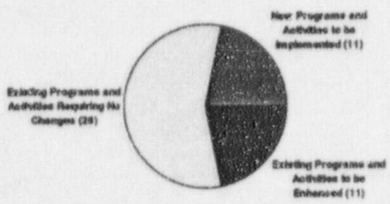
## Oconee Programs and Activities Credited for License Renewal

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## Oconee Programs and Activities Credited for License Renewal

**Existing Programs & Activities Requiring No Changes (29)**

- 230kV Keowee Transmission Line Inspection
- Battery Rack Inspections
- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Coatings Program
- Containment Inservice Inspection Plan
- Containment Leak Rate Testing Program
- Control Rod Drive Mechanism Nozzle and Other Vessel Closure Penetrations Inspection Program
- Crane Inspection Program
- Duke Power Five-Year Underwater Inspection of Hydroelectric Dams and Appurtenances
- Duke Quality Assurance Program
- Elevated Water Storage Tank Civil Inspection
- Environmental Qualification Program
- Federal Energy Regulatory Commission (FERC) Five Year Inspection
- Fire Protection Program
- Heat Exchanger Performance Testing Activities
- Inservice Inspection Plan
- Inspection Program for Civil Engineering Structures and Components
- Penstock Inspection
- Piping Erosion/Corrosion Program
- Program to Inspect the High Pressure Injection Connections to the Reactor Coolant System
- Reactor Coolant System Operational Leakage Monitoring
- Reactor Vessel Integrity Program
- Service Water Piping Corrosion Program
- Steam Generator Tube Surveillance Program
- System Performance Testing Activities
- Tendon - Secondary Shield Wall - Surveillance Program
- Thermal Fatigue Management Program

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## Oconee Programs and Activities Credited for License Renewal

**Existing Programs & Activities to be Enhanced (11)**

- Auxiliary Service Water Piping Inspection
- Borated Water Storage Tank Internal Coatings Inspection
- Component Cooler Tubing Examination
- Condensate Cooler Tubing Examination
- Condenser Circulating Water System Internal Coatings Inspection
- Decay Heat Cooler Tubing Examination
- Keowee Oil Sampling Program
- Main Condenser Tubing Examination
- Reactor Building Cooling Unit Tubing Inspection
- Standby Shutdown Facility Diesel Fuel Oil Tank Inspection
- Turbine Generator Cooling Water System Strainer Inspection

**New Programs & Activities to be Implemented (11)**

- Alloy 600 Aging Management Program
- Reactor Vessel Internals Aging Management Program
- Cast Iron Selective Leaching Inspection
- Galvanic Susceptibility Inspection
- Keowee Air and Gas Systems Inspection
- Once Through Steam Generator Upper Lateral Support Inspection
- Pressurizer Examinations
- Reactor Building Spray System Inspection
- Reactor Coolant Pump Motor Oil Collection System Inspection
- Small Bore Piping Inspection
- Treated Water Systems Stainless Steel Inspection

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## Oconee Safety Evaluation Report Open & Confirmatory Items

- 43 Open Items and 6 Confirmatory Items
- 28 of the 49 items are relatively straightforward to address
- 3 of the 49 items are related to the UFSAR Supplement
- 18 of the 49 items (5 topic areas) will require meetings:
  - Scoping Process and Results
  - Complex Assembly Boundaries
  - Consumables
  - CASS Components
  - RV Internals

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## Observations on Implementing the License Renewal Rule

- Develop clear definitions of terms
- Document alternate scoping and screening processes
- Develop a technically sound process for handling emerging issues

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