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UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

DATE: Friday, November 17, 1989

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

355TH ACRS GENERAL MEETING

Nuclear Regulatory Commission

Room P-110

7920 Norfolk Avenue

Bethesda, Maryland

Friday, November 17, 1989

The above-entitled proceedings commenced at 8:30 o'clock a.m., pursuant to notice, Forrest J. Remick, Committee Chairman, presiding.

PRESENT FOR THE ACRS SUBCOMMITTEE:

Carlyle Michelson, Vice Chairman

James. C. Carroll, Member

Ivan Catton, Member

William Kerr, Member

Harold W. Lewis, Member

Paul G. Shewmon, Member

Chester P. Siess, Member

David A. Ward, Member

PARTICIPANTS:

H. Alderman

C. Miller

D. Scaletti

J. Tsao

C. Dillman

C. Sawyer

G. Thomas

T. Chandrasekeran

K. Parczewski

J. Spraul

P R O C E E D I N G S

[8:30 a.m.]

MR. REMICK: Good morning, ladies and gentlemen. The meeting will now come to order.

This is the second day of the 355th meeting of the Advisory Committee on Reactor Safeguards.

I am Forrest Remick, Chairman of the ACRS.

During today's meeting the Committee will discuss the GE Advanced Boiling Water Reactor, ABWR, and continue our discussion from yesterday of Generic Issue 87, the HPCI Steam Line Break without Isolation, and we'll hear and discuss ACRS subcommittee activity reports on planning and procedures, a discussion of selection of ACRS members and officers and preparation of any reports.

Items for consideration on Saturday are listed on the schedule posted on the bulletin board outside the meeting room.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act and the Government in Sunshine Act.

Portions of the meeting may be closed as necessary to discuss proprietary information applicable to the topic being discussed and/or information of a personal nature where disclosure would constitute a clearly unwarranted invasion of personal privacy.

Mr. Herman Alderman, on my right, is the designated

Federal official for the initial portion of today's meetings.

We received no written statements or requests to make oral statements from members of the public regarding today's sessions.

A transcript of portions of the meeting is being kept and it is requested that each speaker use one of the microphones, identify himself or herself and speak with sufficient clarity and volume so that he or she can be readily heard.

As I indicated, the first item for today is the GE ABWR.

Mr. Carlyle Michelson, the Vice-Chairman of ACRS, is the Subcommittee Chairman in this case so, Carlyle, I turn the meeting over to you.

MR. MICHELSON: Thank you, Mr. Chairman.

The ABWR Subcommittee held a meeting with the Staff and the General Electric Company on October 31st to discuss Module 1 of the draft Safety Evaluation Report for the ABWR. You have in Tab 8 a copy of the minutes of this meeting -- our status report rather on this meeting. They appear starting on page 2 of Tab 8 and the minutes are rather complete and I think will give you a history of what's going on.

What Module 1 is is a consideration of Chapters 4, 5, 6 and 17 of the Standard Safety Evaluation Report.

Chapter 4 is the reactor, chapter 5 is the reactor coolant system, 6 is the engineered safety features, and 17 is the quality assurance program.

The Subcommittee I believe it was Mr. Carroll and David Ward I believe were the only two that made it. We had I think a very fine Subcommittee meeting, received good answers I think to all the questions we asked. Clearly there are still many open items and incomplete sections yet in the document, in Module 1 of the document, but I think the meeting was a very good one. It gave us quite a bit of information.

As a consequence of the meeting and in preparation for this full committee consideration, I have done a couple of things.

First of all, I have asked the Staff and the General Electric Company to come in and make presentations primarily focusing on those parts of these chapters that are now complete and ready for our consideration, keeping in mind there are still a number of incomplete portions within these chapters.

We'll focus on what we have ready to review and we'll have to wait for the remaining material until the next time we write a letter on a module and then we can pick up whatever was missing from this module.

Also in preparation for this meeting I have prepared a first draft of a letter on the ABWR to allow the members to kind of see where we think comments are needed so that you can ask additional questions, particularly in those areas if you are so inclined during the meeting.

I think that's going to be -- I think it's the pink one over there, Herman -- that's the color I got -- so with those thoughts in mind, I'd like to ask, first of all, if any of the other Subcommittee members have any particular questions or comments to make on the Subcommittee meeting or anything else relating to the ABWR.

MR. CARROLL: I have none.

MR. MICHELSON: Seeing none, I would like to make only one general comment before we get started and that is that I am trying to make an attempt to go back to look at our old letter of 1987 in which we indicated, I think we commonly call it the Camel letter, which we indicated the kind of things that we would like to see in an improved light water reactor.

I believe that at this stage of the game we haven't come across material that would be applicable to that particular letter but as we do come across it, of course, or if the members sense that we have already come across some that need to be highlighted we should bring up those in the letter on these modules, on the appropriate module as it comes along because I think we do need to go back and address our old thoughts and desires.

There is also a number of other letters along the same line. A partial package of these letters is going to be handed out -- I guess it's not copied yet or is it? Maybe it's already at our place, the Camel letter and those replies.

MR. ALDERMAN: No, it hasn't been.

MR. MICHELSON: We will get to you a copy of the Camel letter, in case you've forgot what it contained and a copy of the General Electric response and I think there are going to be a couple of other responses in there as well.

MR. SHEWMON: If I recall, it had almost as many signers as --

MR. MICHELSON: They are in the original letter, yes, and I think it would be well to track this letter just to be sure that our early thoughts are being appropriately incorporated as we see fit as we go along.

I believe that is about all the introductory remarks that are needed at this time. We would like to start and does the staff for General Electric want to start first?

MR. SCALETTI: I guess in accordance with the Agenda, that the presentations are to be scheduled by General Electric Company, the Staff is here to answer any questions that the Committee may have but we have planned no formal presentations.

MR. MICHELSON: Okay. Your plan is to just listen and answer questions as we may have them?

MR. SCALETTI: That's correct.

MR. MICHELSON: Okay. Then I believe the General Electric Company is ready to make their presentation, so we'll proceed. Thank you.

[Slide.]

MR. DILLMANN: Good morning. I am Charles Dillman, Manager, Mechanical Equipment Design, for General Electric.

Today Dr. Craig Sawyer and I will be presenting sections of this module, Chapters 4, 5 and 17.

[Slide.]

MR. DILLMANN: The sections that we are presenting are the ones that the Staff and the Subcommittee feel are ready for consideration and closure and this list is of those sections. If at any time you want to talk about other sections we'd be happy to try to accommodate that.

Also, if you would like to change the order of any of these sections, we can accommodate that.

[Slide.]

MR. DILLMANN: We have planned to go through this material by approximately noon-time and stand ready to spend any further time necessary to answer all your questions and provide you the information that you need.

MR. MICHELSON: One of the things I would like to point out to the Committee, of course, is Chapter 4 is an example of the somewhat incomplete nature of the SSAR and thee DSER, the Design Safety Evaluation Report.

We have in this chapter only the two sections that were on the previous slide, 4.5 and 4.6. The other sections are going to be submitted as I understand it as a supplement later on, is that correct? They have not yet been submitted?

MR. SAWYER: They have been submitted to the Staff but we haven't got the Staff Safety Evaluation on some of the sections.

MR. MICHELSON: Okay. We just don't have their evaluation report here.

If they have been submitted to the Staff, I didn't get them yet in my packages either. They are missing from mine, yes, they are.

MR. SCALETTI: All of Section 4 should be in.

MR. MICHELSON: I am missing 4.1, 4.2 -- well, it says right in it in the one I have, it says "to be submitted later." My document must be getting old. I didn't bring it with me. Those are all to be in a supplement later.

Is there some way of knowing what we're supposed to have?

MR. SCALETTI: You are supposed to have through Amendment 8 of the SSAR.

MR. MICHELSON: Now how do I know -- if it's incomplete, how do I know that? Are the pages, each one marked Amendment 8?

MR. SCALETTI: Only Amendment 8 pages are marked Amendment 8 but if you have the external event analysis and the latest update to the PRA, then you have Amendment 8.

MR. MICHELSON: Well, we'll go into that later.

MR. SCALETTI: It is in Chapter 19.

MR. MICHELSON: Go ahead.

[Slide.]

MR. DILLMANN: We will start with Section 4.5 on Reactor Materials. If at any time during this discussion you have questions or want to proceed in greater depth, please ask. Of course the Staff also is welcome to provide supplemental input as required.

In general, the reactor materials comply with all the applicable codes, regulations and guides. In addition, materials in contact with the reactor coolant incorporate the experience and development efforts of the past 15 years for materials in contact with BWR water.

Materials that we used in the design, fabrication of the ABWR are the materials that have been demonstrated by successful experience and by extensive laboratory testing.

The pressure vessel steel includes a low initial NDT combined with a very low radiation buildup, because we can control the constituents that affect radiation buildup.

In all materials and all fabrication we implement process controls to assure that the material properties are not degraded, including the resistance to stress corrosion, cracking. We avoid sensitization, for instance.

Furthermore, to further enhance the stress corrosion cracking situation we apply stress rules since stress corrosion cracking is produced by coincidence of susceptible material, stress and environment. We control all three factors.

[Slide.]

MR. DILLMANN: The materials used in contact with the reactor coolant include nuclear grade 304 and 316 stainless steel, which has low carbon to avoid weld sensitization, because the low carbon tends to reduce the strength.

We control the nitrogen to keep the strength level up, consistent with the ASME requirements. We employ grain size control and we use solution heat treatment as much as possible, and certainly where the material has been exposed to sensitization operations, such as thermal treatment.

The welding material for the stainless steel, we control the as-deposited ferrite. We have requirements on that. We control the composition. Stainless steel castings are the low carbon CF-3 grades; again, for their enhanced resistance to stress corrosion cracking.

We control the ferrite and we solution heat treat them to enhance their resistance to stress corrosion cracking.

MR. SHEWMON: Can you tell me what the code is that talks about the control on the ferrite and stainless steel castings?

MR. DILLMANN: We have our own requirements on it.

MR. SHEWMON: Then, can you tell me what that is?

MR. DILLMANN: Yes. It's a ferrite number greater than eight and generally less than 25.

MR. SHEWMON: Generally less than 25 can still get you up into a lot of aging. I'm a little surprised you aren't lower than that, and generally isn't very specific. So it's the upper limit I'm concerned about.

MR. DILLMANN: The upper limit we impose, I say generally, on most applications we impose 25. In some applications, we've imposed a slightly lower limit, around 20.

MR. SHEWMON: That's really no improvement, then, over what you were doing 20 years ago, where we've got some old stuff that's aging that's about 25 percent ferrite.

MR. DILLMANN: I'm not aware of any problems we've had with that.

MR. SHEWMON: Well, you haven't broken them yet, but if you do the toughness measurements, they're pretty poor, if you follow the research that the French have been doing or the Americans have been doing.

MR. DILLMANN: I'm afraid I'm not familiar with that.

MR. SHEWMON: So you have no standard, there is no code that the NRC makes you follow on this; there's no ASME code. It's whatever you people want to do and you commit to something under 25 usually. Is that your answer?

MR. DILLMANN: Yes.

MR. SHEWMON: I'm surprised it's that lax.

MR. REMICK: Could you enlighten me on what the experience has been over the last couple of years with the nuclear grade 304? I know that a number of pipe replacements were made and so forth. What has been the experience?

MR. DILLMANN: The replacements have been made and there's been no subsequent findings of any problem. I guess the first of those replacements probably goes back about seven or eight years and the subsequent examinations have shown no problems in those replacements.

Preceding that, there was extensive laboratory work on the material in our pipe test lab, for instance, to show that it was resistant to stress corrosion cracking under very adverse conditions.

MR. REMICK: When you say no substantial problems, could you elaborate a little bit more? Have they found any cracks or any leaks?

MR. DILLMANN: No. I said there's been no findings.

MR. REMICK: No findings. Thank you.

MR. MILLER: Mr. Chairman, John Tsao from the staff would like to make some remarks at this time.

MR. REMICK: Go ahead.

MR. TSAO: This is John Tsao. I wanted to reply to Mr. Shewmon's question as to NRC's guide. It would be under Reg Guide 1.31. The title of the guide is Control of Ferrite Content in Stainless Steel Welds Metals. That guide, 1.31, requires --

MR. SHEWMON: I'm not interested in the welds. I'm interested in castings. Does it also cover castings?

MR. TSAO: Yes, sir.

MR. SHEWMON: The title says welds, but I want to know if it's also castings.

MR. TSAO: It is.

MR. SHEWMON: And the limits are what? You can go as high as 25? Is that admissible?

MR. TSAO: The limit in that guide is 5 to 20.

MR. SHEWMON: And will GE be urged to comply with that instead of their 25?

MR. TSAO: GE says they're going to use average of ferrite number eight.

MR. SHEWMON: It's not the average that gets you in trouble.

MR. DILLMANN: To answer your question, we will comply with Reg Guide 1.31. I had missed the 20. If Reg Guide 1.31 says 20, we will comply with that.

MR. TSAO: In fact, GE has indicated in their SSAR that they are going to follow Reg Guide 1.31.

MR. SHEWMON: Very good. Thank you.

[Slide.]

MR. DILLMANN: The next material is XM19. This is an austenitic stainless steel with higher strength, originally developed for gas turbine aircraft engine applications. This is used primarily for fasteners and for other special applications, some pump shafts we have made from XM19.

In the case of XM19, we test each lot for stress corrosion resistance. When we use it as a threaded fastener or a threaded component, we also apply special stress and fluence limits. Alloy 600, Inconel by trade name, is used where higher strength is required or where a thermal expansion matched with carbon and low alloy steel is required. For instance, the lower portion of the shroud support and, of course, safe ends are made of Alloy 600.

In Alloy 600, we avoid creviced welds. Our first rule is not to have any welds in a creviced environment. Where it is absolutely impossible to avoid creviced welds, we use a stabilized material. That stabilized material has been extensively tested for resistance to stress corrosion cracking and is in use in BWR environments in Japan for several years now.

Also, in the case of Alloy 600, as in the case of other materials, we apply stress rules depending on the environment and the presence of a crevice or absence of a crevice. The material is used in a solution annealed or a solution annealed plus special heat treatment form to provide further stabilization.

[Slide.]

MR. DILLMANN: Carbon steel. We have a requirement for intrinsically tough carbon steel. We have minimum impact requirements. We also apply special fatigue design rules to the carbon steel. Low alloy steel we apply to the special chemistry controls in the high fluence zones, consistent with the latest knowledge on copper, phosphorous and nickel. It is also consistent with the latest issue of the NRC Reg Guide on this subject.

MR. SHEWMON: Sir, do you know whether you'll be using forgings or plate in the core barrel?

MR. DILLMANN: We will be using forgings in the core barrel region. There were no vertical weld seems opposite the core.

MR. SHEWMON: That's not in the requirements document we have. What is there says you could also use plate and talks about how you would make and inspect the welds that were in the core barrel. So the submittal is in error or incomplete?

MR. DILLMANN: It's less definitive than our pressure vessel specification that's at a lower tier in the documentation, that specifically requires forgings in the belt line region.

MR. SHEWMON: What is of particular concern to me is that the spec you have in there allows 04 sulfur in it which is about a World War II grade steel; has low toughness, anisotropy, all kinds of nasty problems that no good company would put into a pressure vessel, even if it is in the code.

I am some interested in knowing why you don't have a spec on the sulfur which gets it up at least to 1980 practice.

MR. DILLMANN: Our pressure vessel specifications bring it up to current practice --

MR. SHEWMON: Not if it's 04 sulfur. Not even if it's 025.

MR. DILLMANN: I'd have to look at what we have in the SSAR.

MR. SHEWMON: Well, what's in the documentation I was given from that was SA533 and 508, and if you go look at the code for 533, which is of particular concern, it says you can specity something like 015, which would be good modern practice, but you don't. You call out the copper and you call out the phosphorous and you call out the nickel, but you don't call out the sulfur.

MR. DILLMANN: In the SSAR, we specifically call out the copper, nickel and phosphorous because they are specifically addressed in the Reg Guide and we're trying to show compliance with the Reg Guide.

Again, below the tier of the SSAR, we have considerable specifications that implement not only the commitments of the SSAR and the codes and regulations, but also implement the best practice that we know today.

MR. SHEWMON: Could I see some indication of what that would mean for sulfur?

MR. DILLMANN: I will have you sent a specification that shows what we're controlling there.

MR. MICHELSON: It would appear to me that the SSAR is the controlling document and the one that is going to be ultimately certified. I think any important considerations of this sort must, of necessity, be documented in the SSAR, irrespective of what you say in your lower tiered documents, which are not necessarily a part of the certification. They may never even be reviewed by the staff in some cases.

So I think that if it isn't in the SSAR, we assume that you may not do it. I think that's not unreasonable. An important basic criteria.

MR. DILLMANN: I understand your point. The other side of it is that the volume of material that implements our detailed design is way -- is impractical to have in the SSAR.

MR. MICHELSON: We understand that, but I think the basic criteria, the basic requirements, the key ones have to appear in the SSAR or we assume they may not be carried out as we thought.

MR. SHEWMON: 04 sulfur is a miserable steel. You don't want to put that in.

MR. DILLMANN: No. We would not build parts out of that kind of material.

MR. SHEWMON: I would hope not.

MR. DILLMANN: I will get you a copy of that specification. We also specify low transition temperatures depending on the particular part. It's minus 38F to minus 20F.

We also have some special requirements on some materials. For instance, we use high purity material in high fluence locations, specifically today in the control bleeds. In the thicker section material, the high purity material is somewhat short on strength and we control the fluence in those parts rather than using the high purity material.

We use low cobalt materials in the internals and, in general, use cobalt-free wear materials where in contact with reactor coolant.

[Slide.]

MR. CARROLL: What's an example of a cobalt-free wear material for such as thing as, say, the rollers on the control board?

MR. DILLMANN: The rollers on the control blades that we're currently using and the latest blades we're shipping and would use in ABWR is -- the pins are X750 and the rollers are colmonoy. That was developed about seven, eight years ago, partly under EPRI's sponsorship and partly under GE work and has been implemented in the last four or five years.

MR. SHEWMON: What is the limit you have on the cobalt content in the stainless steel that's exposed to the primary coolant?

MR. DILLMANN: I'll have to look that up. I don't have that in my mind. I will look it up. I'll advise you later.

MR. SHEWMON: Fine.

MR. DILLMANN: So in summary, the materials are based on successful experience. The materials are controlled. The processing is controlled and when I say processing, that includes things like forming, welding, heat-treating, all the way from the basic melting through the installation.

We control contaminants that can come in contact with the material and then of course, once a plant is turned over for operation, we give guidance to the owners to control the abuse of the material after that point.

Our materials comply with all the regulations, codes and guides in addition to complying with all our experience. In going through some specific systems, the CRD system, material properties are equivalent to ASME Code Section 2, Reg Guide 1.85 and for some applications, we may specify additional limits.

We meet Reg Guide 1.31, ferrite control, Reg Guide 1.44, sensitized stainless steel and Reg Guide 1.37, cleaning. The internals are ASME 2 and 3 materials and again, we comply with the same Reg Guides.

MR. SHEWMON: Sir, I don't know when is the best time to get at it but you claim very low fluences apparently in your pressure vessel compared to what current BWRs are finding.

MR. DILLMANN: Yes.

MR. SHEWMON: And I wondered if -- I would like to hear why this is true, what fundamental changes there are in geometry and construction that would make it that way.

MR. DILLMANN: I'm going to touch on that briefly in a later section but the answer is that because of the reactor internal pumps, the annulus between the pressure vessel and the core is much larger. The core geometry is basically the core geometry that we used in the 251-inch BWR-6 with a slight difference in the pitch but the vessel diameter is 278 inches as opposed to the 251 inches.

So because of the large annulus, the fluence is down.

MR. SHEWMON: Okay.

MR. DILLMANN: That closes the presentation on materials. If there are no questions, I will move on to the next one which is the fine motion control rod drive system.

The fine motion control rod drive in itself is one of the things that makes the ABWR an advanced reactor. The fine motion control rod drive has redundant means of an insertion, somewhat different than our past product lines. It has a hydraulic scram and an electric motor-driven insertion. The electric motor drive, of course, is also used for the normal operation positioning of the rods.

The FMCRD and its system has a fine motion capability, 18 millimeter steps as opposed to the 6 inch steps that we've used in the past. Because of the fine motion plus the electric drive, we can allow a much more automated start up, avoiding operator error and improving the start up time. We can move large gangs of rods during start up.

It also facilitates load following. While the primary load following is done by recirc core control, we can move rods to get to the deep power reductions at weekends or nights.

MR. REMICK: A question -- on the automated start up and load following, I assume that requires some kind of controls, perhaps software and so forth. Will that type of capability be in the certification or would that be an add on if a utility chose to do that?

MR. DILLMANN: It's in our basic design.

Craig, do you have any comments on how much of that is in the certification?

MR. SAWYER: I believe that's all in the certification but it's not of course in this chapter. It's in the chapter on control.

MR. DILLMANN: Basically, just briefly, the control system is a solid state control system, software-based, and it's of course being implemented in the plants that we are constructing in Japan and is an extensive development and design verification behind the software aspects of the control system, not only for this but for all the control systems.

It's a basically state of the art system.

MR. REMICK: We would expect eventually to see some details of that.

MR. DILLMANN: You should see that covered in a subsequent module.

MR. SHEWMON: The fuel you used to sell to your customers wouldn't allow very much load following without a lot of seasoning and even then there were problems. Is this new modified fuel you're selling going to allow most of these good things to happen?

MR. DILLMANN: Yes, the barrier fuel has successfully avoided the PCI problems that imposed those limitations in the past and would allow load following to the extent that utilities need load following today -- or in the future for that matter. Our BWR 5 and 6 product lines especially were based on research "control" valves that allowed very good load following capability and, of course, the fuel was developed to -- the barrier fuel was developed to avoid the PCI, to allow the power shifts quickly.

So those problems have been handled. The fine motion control right drive was originally pursued by GE in the early '70s and parallel with the development of the barrier fuel to be another approach towards avoiding PCI but when the barrier fuel looked very good, we dropped the development of the fine motion control rod drive for that purpose but continued to pursue it for ABWR.

The other thing that's different from our previous product lines is that we have two FMCRDs per hydraulic control unit where in the previous U.S. BWRs, there's only been one drive, one hydraulic control unit, but this concept has been successfully used in Europe for years where they have as large number of drives. The Swedes, for instance, use up to 12 drives on one hydraulic control unit.

We decided to only go as far as two because it allowed using the basic components that we had experience with before where if we'd gone larger, we'd have had to go to a different type of accumulator and a different type of scram valve.

In doing this 2 FMCRDs per hydraulic control unit, we maintain separation of the two drives on one hydraulic control unit.

[Slide.]

MR. DILLMANN: Here's a basic schematic and I apologize for it being somewhat small, of the control rod drive system. We down here have the charging pump which is really redundant pumps taking water from the condensate system. Also shown on this PNID is a heater. That heater is somewhat of an accessory. It's there because in Japan where the plants are by the seaside, they've had some problems with condensation on the cold pipes, causing some cracking. So our basic design has a heater for those type of plants.

The charging pump of course maintains water to the hydraulic control units and then the hydraulic control units can scram the drives. The hydraulic control unit for this system is much simpler than the ones used in the locking piston. It consists of a gas bottle, an accumulator, scram solenoid valve, scram valves, and of course, associated instrumentation that monitors and alarms on low water level or low pressure.

These hydraulic control units do not include the motion equipment which is for the -- in this design is the electrical equipment. The gas bottle is a rather simple structure. The accumulator, we maintained the accumulator diameter that we have experience with in locking piston accumulators and increased the length to get the necessary water volume.

This makes our piston stability because we use of course a piston-type accumulator. It makes our piston stability demonstrated by the previous designs. Also, the scram solenoid valves are consistent with the ones we've used in our latest plants and the scram valves are the same basic design but slightly bigger to accommodate the increased water flow.

There are two charging pumps as I mentioned, one operating, one spare. These are non-safety because if the charging pumps are not working, the plant will scram but we can scram without the charging pumps because the accumulators are pressurized and have sufficient water in them to achieve the scram.

The scram lines, there's only one per drive where in the locking piston, we had both an insert withdraw line. We do have one scram line.

MR. KERR: Excuse me. Do you have any availability requirements on these charging pumps?

MR. DILLMANN: We have a requirement for their duty. We have a specified duty. I can't say that we have a specific availability requirement.

MR. KERR: Is there a reliability requirement? I'm not talking now about safety grade or non-safety but do you specify high quality pumps?

MR. DILLMANN: We specify high quality. I can't say that we say that it shall be available 90 some odd percent of the time.

MR. KERR: How do you specify high quality pumps then?

MR. DILLMANN: By a specification that controls the design of the pumps. One of the biggest problems we've experienced in the past with the pumps is if you don't adequately specify everything that the pump may be exposed to, you may end up with an inadequate pump.

So we pay special emphasis on specifying all the transients and all the off-operating conditions that the pump may see. We of course specify the materials based on our materials technology and we specify the various things we've learned over the years that have caused problems.

We specify a corrective action for those and then of course, the other aspect of getting a quality component is select a quality vendor and we have procedures in place to make sure that our vendors are qualified before we place a purchase order with them.

MR. KERR: Thank you.

MR. DILLMANN: In general, also I might comment that the CRD charging pumps have not been a source of problem in our plants. We had a few material problems in the early BWR-6s, which we quickly corrected and which we have covered, of course, in our specifications for these pumps. But, in general, those pumps have been very reliable.

MR. CARROLL: When and why did you make the change from control rod drive hydraulic pump to the PWR terminology charging pump?

MR. DILLMANN: I don't really know.

MR. CARROLL: But you've been calling them charging pumps for a long time.

MR. DILLMANN: I do. Another feature of this system is that we have redundant protection against control rod ejection in the event of a scram line break. We have a break on the FMCRD and we have a check valve in the FMCRD such that if the scram line breaks, the check valve closes and prevents the rod from withdrawing. The break also is locked to prevent the rod from withdrawing in the event of a scram line break.

[Slide.]

MR. DILLMANN: The electrical system consists of a stepping motor, a power supply, and the control logic. The significant feature is that both the hydraulic and electrical systems allow functional testing during operation. It allows independent testing of the scram channels and rod motion.

We have a rod pattern control system that minimizes rod worth and avoids withdrawal errors. We've eliminated the rod drop accident by a combination of mechanical design and control system. We've eliminated the scram discharge volume that existed with the locking piston drive.

This gives us several advantages. It removes a radiation source for personnel exposure. It eliminates a source of potential common mode failure and it's generally an improvement to the design. We have an internal shoot-out protection in the event of the housing weld failure, where, in the previous plants, we had an external shoot-out protection.

MR. SHEWMON: What weld is that?

MR. DILLMANN: Let me skip ahead and I will show you.

[Slide.]

MR. DILLMANN: First, let me put you in the big picture. The CRD housings come up through the vessel and we're looking in this region where there's a stub tube and a weld.

MR. SHEWMON: You have another drawing in your handout. In fact, you've got two of them. Fine motion control rod drive systems. I assume the weld there is also to a pressure boundary. Is that correct? If not, I'd like to come back to it, because that I know where I am and I still don't know where I am on this. Go ahead.

MR. DILLMANN: This is a stub tube going through the -- coming up from the bottom of the vessel. This is the CRD housing and this is the weld.

MR. SHEWMON: And what's the material on both sides of the weld?

MR. DILLMANN: The material on both sides of the weld is stainless steel.

MR. SHEWMON: You've got a stainless steel ferrite weld down where that stub tube comes out.

MR. DILLMANN: Yes.

MR. SHEWMON: And the anti-shoot-out protection you were talking about is --

MR. DILLMANN: Basically, the anti-shoot-out protection is like this. If this weld shears, such that the housing can be driven downward by the pressure, the drive is locked to the CRD guide tube. The CRD guide tube, up at the top, is too large to go through the core plate.

It drops about nominally about three-tenths of an inch and stops. We have an analysis. In fact, the analysis of this has been given to the staff recently for review.

MR. SHEWMON: Would you go to the other diagram which you havo in there, the fine motion control rod drive system.

[Slide.]

MR. SHEWMON: Is this sleeve is in there?

MR. DILLMANN: Yes. When I said that the drive is locked to the guide tube, this outer tube here is the drive pressure tube. This is all a welded structure, coming down here to this flange here. This flange, of course, concludes the check valve I talked about. But this flange is clamped between the housing flange and the seal housing flange.

So that if, indeed, this weld shears and this housing is driven this way, the fact that the guide tube is locked to this drive, your load path then is from the guide tube, from the guide tube sitting on the core plate down through this bayonette coupling, down through this outer drive cylinder to this flange, which then is holding the housing.

So we cannot eject a rod. We cannot eject the housing.

MR. SHEWMON: This one looks like the sleeve could slide out easily and you're saying there is something underneath it down here which will hold it in. Is that right?

MR. DILLMANN: This is bayonette coupled to the guide tube. This is a welded structure. This is a one-piece structure.

MR. SHEWMON: You're pointing inside the sleeve and it's the sleeve, the weld on the outside of the sleeve that we're talking about failing, so I don't quite see why the whole sleeve can't come out if the weld fails.

MR. DILLMANN: You're talking about this housing.

MR. SHEWMON: I'm calling that a sleeve, yes.

MR. DILLMANN: Okay. Let's look again at the previous picture.

MR. SHEWMON: That tells me nothing, because it shows not what's connected to what down other places.

MR. DILLMANN: This is the upper portion that isn't shown on that other drawing had. I had to have it in separate drawings because it gets so long, it would get too small.

Again, this weld would be the weld -- let's see how it works. This weld here is this weld right here. This guide tube, which is the control rod guide tube, is bayonette coupled to the control rod drive itself, and that bayonette coupling is up here.

If this weld shears, then this housing would be free to move out. In fact, it would be driven out by the pressure.

MR. SHEWMON: Let's come back over the other diagram, then. The pressure vessel is ferritic and the drawing there, then, doesn't show an interface apparently between stainless steel and ferrite because you told me that was a stainless steel-stainless steel weld.

MR. DILLMANN: Right. The stub tube is welded to the bottom head at this location.

MR. SHEWMON: And that's got some kind of an Inconel butter on it?

MR. DILLMANN: Yes. That, of course, is what generally would be called a paste-on type weld, such that if it, for instance, cracked, it can't come out because it's sitting on the solid head. But this weld, of course, if it sheared perfectly would allow the housing to drop.

MR. SHEWMON: Thank you.

MR. DILLMANN: When it drops, this locks here and the load passes down through here to this bayonette, and then we move over to this picture and we come down from that bayonette through this welded structure, and this welded structure is out to a flange here. So our load passes now from the core plate down through here into this inner drive tube, the drive tube down to this flange, and that flange prevents the housing from coming out.

MR. SHEWMON: Fine.

MR. WARD: This feature was changed considerably from the design they showed us a year ago or so.

MR. DILLMANN: Yes. This is in a recent amendment.

MR. MICHELSON: It's an amendment which we did not receive. We received through Amendment 7 and apparently it's in Amendment 8.

MR. WARD: I would be interested in the reasons for that change. I don't know if you remember the earlier design.

MR. DILLMANN: The earlier design had a collar here and the collar, to function properly, had to have a close clearance with this weld. Our concern became that the collar would produce the problem that we're mitigating. It would produce stress corrosion cracking in the weld.

So we eliminated the collar and went to a cleaner design.

MR. SHEWMON: Fine. I had this on my list because that's an uninspectable weld and it's a dissimilar metal, where there have been problems with that butter in some other BWRs. What I'm hearing is that yes, it is still uninspectable and it's still buttered the way it has to be, but if it fails, it will be a small leak.

MR. DILLMANN: Yes. Because the housing stays up in the hole, the leak is actually small enough that it can be made up by our normal makeup systems. So, indeed, that's our story. The shoot-out, per se, is designed more from a missile protection and rod ejection standpoint.

MR. MICHELSON: One of the things that bothers me a little bit on the figure you showed of the housing is that in looking at that, I would have thought that the nozzle to which the housing is welded is a formed nozzle, part of the head, because you showed no additional welds on the drawings.

Now, what drawings in the SSAR will show those additional welds so that we're assured the staff has reviewed this thing properly or do they get the drawings of the vessel itself to review?

MR. DILLMANN: Most generally, we don't -- again, it gets to be a matter of how much detail do you put in the SSAR.

MR. MICHELSON: It's no any detail. In terms of this drawing, it's just coloring in another weld and it's an important one because you explain how that weld, even if the weld of the housing shears, that there is a collar formation that prevents it from being ejected. That I wouldn't have gotten from looking at your drawing.

It's not a big deal. It's not a big job. I don't know what we're certifying ultimately. I think it's the SSAR and, if it isn't shown in the SSAR, then people can come in with a number of modifications of that arrangement that would be automatically certified because it's not a part of our SSAR.

MR. DILLMANN: Again, if nothing else, in the SSAR, we commit to certain capabilities and the capability that we're specifically committing to here is that we have an adequate method of preventing ejection of a rod or ejection of the housing and drive.

MR. CARROLL: That's in the beholder's eye, though.

MR. DILLMANN: Yes. I understand.

MR. MICHELSON: The beholder has to determine it is adequate. I don't know what the staff -- did the staff get the vessel drawings to review this feature?

MR. SCALETTI: I don't think that we had the details that would show us --

MR. MICHELSON: Were you aware that that nozzle was welded into the vessel?

MR. SCALETTI: We were aware of the changes to the weld of the stub tube. We were not aware that the stub tube was -- there was an additional weld at the bottom of the vessel to hold that.

MR. MICHELSON: I think you ought to be made aware of that, and this is the drawing that should have showed it. If your metallurgists had any problem with it, then they could ask. So I don't think it's a big deal, but it's an important feature.

MR. DILLMANN: We'll clarify that drawing in subsequent cleanup.

MR. CARROLL: One other thing I'm interested in. You are, if course, depending on the integrity of this bayonette coupling between the guide tube and the housing.

MR. DILLMANN: Yes.

MR. CARROLL: Is there any way in guide tube replacement that you could misalign the guide tube so that it isn't locked?

MR. DILLMANN: No.

MR. CARROLL: Why?

MR. DILLMANN: Actually, what we lock is the drive. The guide tube sits down -- the positioning of the guide tube is controlled by the pins on the core plate. The control rod drive then is locked to that guide tube and, if the control rod drive isn't in the proper orientation, we can't get the bolts in.

MR. CARROLL: Bolts?

MR. DILLMANN: The bolts through that flange.

[Slide.]

MR. DILLMANN: There's bolts through this flange here that have to line up. So we have a whole alignment system. It's there not only to make sure that the bayonet is lined up but also to make sure that the rod is properly positioned, to allow the fuel interface to be proper. It starts with pins in the core plate and in the guide tube.

MR. CARROLL: Has the staff looked at that detail to satisfy themselves?

MR. SCALETTI: I believe we have looked at the coupling of the rods, yes.

[Slide.]

[Pause.]

MR. DILLMANN: The FMCRD, as I say, the FMCRD originated in Europe and in Europe, there's about 2,700 drives in service, over 15,000 drives years of experience. Actually in Europe, there are two different designs, the Swedish design and the German design. The ABWR design is based on the German design.

Starting in the early '70s, GE picked up the German design, did some further work on it and then starting in the late '70s, that work was transferred to Hitachi and Toshiba and they have continued to test and develop that drive or improve it to a state as it is employed in ABWR.

This included life testing, seismic testing, as well as the various development testing. The control rod is positively coupled to the drive, as opposed to the locking piston drives where we had the collard and fingers. Here we have a bayonet coupling not only between the guide tube and the drive tube, but between the hollow piston, the translating portion of the drive and the control rod.

We also have separation switches and in a minute, I'll lead you through the picture to show you how all this works. We have separation switches that detect failure of the rod to follow the drive -- to withdraw. This eliminates the concern for rod drop. The separation switches, because they're there to mitigate this event, we've made them redundant in Class 1E including separation.

This allowed the elimination of the velocity limiter on the control rod itself with an ability to then make the pressure vessel a little smaller. The other big advantage to this control rod drive other than its fine motion capability and its compatibility with automation and its redundant means of insertion is that both its design and the experience in Europe show that it requires very little maintenance and that it has low radiation exposure associated with that.

The reason for that is that the reactor water never enters the drive. We do not have the reactor water come in during a scram or during rod motion.

[Slide.]

MR. DILLMANN: Now, to show you how some of these parts fit together, this is the picture we looked at a few minutes ago, but let me lead you through it some more.

MR. KERR: I'm sorry. I don't understand why the fact that water doesn't come into the drive results in low exposure. I assume you're talking about neutron exposure.

MR. DILLMANN: No, it's exposure due to maintenance, primarily gamma. The locking piston drive -- every time we scram, reactor water flows through the drive and deposits crud. In this drive, the water flowing through the drive is from the scram system and it's demineralized water.

MR. KERR: Okay, so you really aren't talking about radiation exposure to the drive. You're talking about crud deposition.

MR. DILLMANN: Let me try to clarify. I'm talking about personnel exposure. We have very tight goals in the ABWR design on personnel exposure.

MR. KERR: Okay. Thank you.

MR. DILLMANN: It's a critical item with us.

This is the outer tube of the drive coming down to the flange. This the check valve that is there to mitigate the effect of the scram-line break here. This check valve raises up into this port whenever the flow is reversed.

The scram mechanism is this member here called the hollow piston -- and let me jump to the other picture -- in this picture the hollow piston is the blue part. When the scram valve opens, water comes through here and here and flows past that hollow piston producing a pressure drop across the hollow piston and the hollow piston raises up and inserts.

That hollow piston normally is sitting on top of a carrier here which is called the ball nut. When it separates from the ball nut due to insertion, on the same scram signal, the motor starts and starts running that ball nut in. So the hollow piston goes in and latches but behind it, the ball nut is running up and the ball nut in something over two minutes will come up under the hollow piston and pick it up and hold it.

MR. CARROLL: Where do they separate?

MR. DILLMANN: They separate right at this black line here.

MR. CARROLL: Yeah, okay.

MR. DILLMANN: Now, if we have the rod inserted and we're withdrawing it, we're running the motor in reverse, so it's lowering the hollow piston. Now, if the rod, for instance, stuck on the channel, then the hollow piston would start to lift off that platform.

When that happens, down here we have what we call a weighing platform or weighing device in this figure, and it's a spring-loaded member and there's some magnets in that member and there's a separation switch probe, reed switches here.

So if this hollow piston is not sitting on this ball nut, then this weighing platform moves up and the separation switches are triggered, causing the motor to stop. So we cannot separate the hollow piston from the drive and still withdraw.

MR. MICHELSON: What is inside the hollow piston?

MR. DILLMANN: Do you mean in here or here? This is the lead screw.

MR. MICHELSON: Yes, but is there water in there also?

MR. DILLMANN: There is water in here all the time. Yes.

MR. MICHELSON: It's kind of a trapped volume; isn't it?

MR. DILLMANN: There's a small bleed hole up here that keeps it from being a stagnant volume.

MR. MICHELSON: Does that somehow assure circulation?

MR. DILLMANN: Yes. We run a continuous purge in this line.

MR. MICHELSON: But how does it get up through the hollow tube?

MR. DILLMANN: Because it produces a slight pressure drop across here, a portion of the flow goes up this way and the rest of the flow goes up this way.

MR. MICHELSON: So, it's continually swept, you're saying.

MR. DILLMANN: It is a swept volume.

MR. MICHELSON: Thank you.

MR. CARROLL: How do you calibrate or check the weighing device periodically? How do you know the springs haven't hung up?

MR. DILLMANN: We do that by backseating the rod. Up in the guide tube, there's a backseat feature whose primary purpose is, when you remove the drive, you backseat the rod to keep the water from pouring out but we can run down against that backseat and over travel. When we over travel, we should be separation. If we don't get separation, we have a problem and that's part of the normal surveillance testing to do that check.

MR. CARROLL: Now, has this general concept of a weighing device been proven out through experience in Europe?

MR. DILLMANN: Yes, and also in our test program.

MR. CARROLL: And it's worked very well, huh?

MR. DILLMANN: It works quite well.

MR. CARROLL: Okay.

MR. DILLMANN: We also have a position indicator probe here. This position indicator probe is only for scram testing. Our normal position indication for the rods is through the synchro device on the motor. That synchro device basically counts turns and reports the rod position.

MR. MICHELSON: Let me caution you. We've alloted 30 minutes for this. You've been at it well over 30 minutes. Do you have much left to finish up?

MR. DILLMANN: No.

MR. MICHELSON: Okay.

MR. DILLMANN: I think those are the key points. The other part here is the seal housing which has a packed joint with the shaft. This is a maintenance item and it's periodically maintained once every 10 years and when that's lowered, the whole drive assembly sits down and locks here on a spine joint.

So that's the control rod drive situation.

MR. CARROLL: One final question. The opening at the top of a drive, have you sized it like you used to so that a fuel clip will fall directly into the control rod drive mechanism?

[Laughter.]

MR. DILLMANN: No, it shouldn't be able to do that.

MR. CARROLL: Good. When you supply the utility with left-handed lock washers to hold the fuel clips down.

MR. DILLMANN: Yeah. We think we have that under control.

[Slide.]

So in summary, system and components are based on experience. We of course covered rod shoot somewhat out of order. The figures are in your charts. Components are fully tested and the system provides improved operability, lower maintenance, lower exposure, and because of the redundant run end and elimination of scram discharge volume, some improvement in the safety aspects of the control rod drive system.

[Slide.]

MR. DILLMANN: Next item is going to be very brief. Compliance with 10 CFR 50.55(a) section on codes and standards, basically in this what we are committing to is that the reactor coolant pressure boundary is classified in accordance with 10 CFR 50.55(a) and meets the requirements of ASME III, Class 1. Quality Group A. It's consistent with Reg. Guide 1.26.

The ASME code date that we've committed to is the

1986 edition and our code cases. As authorized by Reg. Guides 1.84 and 1.85 as was discussed with the subcommittee a couple of weeks ago, we had listed a couple of code cases that weren't covered by those Reg. Guides, but we are withdrawing those code cases. We don't need them and we will withdraw them from our list.

In the SSAR, there's a complete set of tables that go on for several pages, of which this is just one example. This is the example for the reactor pressure vessel, but in here, it shows the safety class, reactor vessel is Class 1, quality group classification A, quality assurance B, seismic category 1. The whole basic plant is described in these tables.

So, summation on this section is that we are in compliance with the regulations and are using up to date codes and standards and code cases for the design.

MR. MICHELSON: Which table did you just show from the SSAR?

MR. DILLMANN: That is Table 3. I believe it's 3.2. It's in section 3 anyway.

MR. MICHELSON: Yes, well, we're of course not dealing with section -- Chapter 3 at this time.

MR. DILLMANN: Right. The reason I showed it is it's reference in Chapter 4, Codes and Standards, and I just wanted to call your attention to the fact that we've done the classification effort.

MR. SHEWMON: Carl, I'd like to call out that the staff person involved, neither he nor I can find anything about casting -- the ferrite content of castings in Reg. Guide 1.31 and there ought to be some limit and it's not clear to anyone in the room that there is outside of these internal GE documents that aren't part of the SER.

MR. MICHELSON: So it should be in the SSAR, clearly.

MR. SHEWMON: Yes, and it may be there, but it would be nice to know where.

MR. MICHELSON: I think as we go through today, I would like to ask all the committee members if they have questions that they think belong on a letter, to write out the question or the concern or whatever, because then we can fit it into the draft letter very easily.

We'll find an appropriate place. If you would, write out what you think your concern is. Then we'll find a home for it.

Thank you.

Go ahead.

MR. DILLMANN: Okay, if there's no further discussions on these preceding sections, I give you Dr. Sawyer who will talk about over pressure protection.

MR. MICHELSON: Craig, we do need to point out we're running a little behind already. Maybe some of these will go much faster than we had allowed time for, but we do have a total limit on our time.

[Slide.]

MR. SAWYER: I'm Craig Sawyer from General Electric Company. Actually, I think we just caught up with that fast run through of compliance with 50.55(a). We did it slightly out of order compared with your agenda because we did it in accordance with our organization in the SSAR.

[Slide.]

MR. SAWYER: Over pressure protection; we conform to 10 CFR 50, Appendix A. It has several elements to it. We have an automatic depressurization system included as part of the ECCS, and we're going to talk about that again when we talk about Section 6.3 later this morning, which is tied in with the low pressure flooder systems and the rest of the ECCS.

The ADS subset makes use of eight of the 18 safety relief valves which we have, which are operated by pneumatic actuators. As you recall, our SRVs are dual function. They have a safety function which lifts against the spring force, which is compliant with the ASME code for over pressure protection, and it has a relief function opened on a demand signal using pneumatic actuators as the mode of force to depressurize either manually or automatically.

The purpose of the SRVs is to limit the reactor pressure to 110 percent of design pressure. Our design pressure is 1250 psi. That makes the over pressure limit 1375.

That is for the transients per ASME Code Section 3 for MSIV closure with the backup scram, the high flux scram.

MR. KERR: When you say that these eight limit the pressure to 110 percent, how many must function in order that that limit be achieved?

MR. SAWYER: I think this will answer your question, sir.

[Slide.]

MR. SAWYER: This is the nominal set point -- excuse me. This is the analytic set points. The nominal set points are about 27 psi lower than this in order to assure that we meet the analytical limit. This is the number of valves in each category. We have a couple of valves right at the low end so that, for example, on an isolation event, after the first lift, then only one valve, or if it's out of commission, its backup valve will continue to cycle.

The rest of them are grouped in fours. The spring set point pressure is 1190 and the relief set points are somewhat lower. We don't take any credit for the relief function in compliance with the ASME code, although in practice, of course, it will happen.

[Slide.]

MR. SAWYER: This chart shows the peak pressure for the ASME closure event, or other kinds of events in the low 1200s. So on the first lift off of those kind of events, all of the valves will lift.

MR. KERR: How many have to fail in order that you'd not be able to achieve your objective?

MR. SAWYER: I don't know the exact number, but it's several. Several of them would have to not operate in order for us to --

MR. KERR: Have you calculated the likelihood of simultaneous failure of that many or is that part of your PRA, which --

MR. SAWYER: We've done it as part of the PRA, and it turns out that the dominant failure is not of the mechanical valve itself, but of the logic that tells the valves to lift. So in the case of meeting the ASME code, for example, where we only take credit for the spring pressure, the probability of having many valves not lift is extremely low.

MR. KERR: I don't know what extremely low means.

MR. SAWYER: I forgot what we've used for our failure rate per valve, but I can look it up for you.

MR. KERR: There would be somewhere that --

MR. SAWYER: We've documented in the PRA the failure rate that we use for the valves, and it's just a matter of my looking it up. I don't remember it offhand.

MR. MICHELSON: These pressure set points, I assume, are the set points of the individual valve springs. Is that correct?

MR. SAWYER: That's correct.

MR. MICHELSON: So there isn't any common logic telling the valves when to open.

MR. SAWYER: Not on the spring.

MR. MICHELSON: Not on the spring. So I wasn't sure what your common logic statement meant.

MR. SAWYER: That was referring -- the common logic statement was referring to the use of it in the relief mode, and then I quickly recalled that the question was really getting after the ASME function to which that doesn't apply anyway.

MR. MICHELSON: Wnich has no common logic.

MR. SAWYER: That's correct.

MR. CARROLL: The ATWS number there is for the closure of MSIV.

MR. SAWYER: MSIV ATWS. It's the worst over pressure transient without scram.

MR. CARROLL: Does that bound the ra[illegible] of void coefficients and stuff that you anticipate?

MR. SAWYER: Yes. The part of this design for the nuclear boiler system, we've tried to anticipate any future moves of the fuel designs and have used a substantially more negative void coefficient than actually the fuel today is achieving.

I should point out that the compliance criterion for

ATWS doesn't require us to meet 1375. It would be more like 1500. We chose to provide enough relief capacity that the 1375 could also be met for ATWS.

MR. KERR: That assumes that all of the relief valves function.

MR. SAWYER: That's correct. For the ATWS case, we assume all the relief valves function. I can provide you with capacity charts that we have in our records at home that we haven't provided as part of the SSAR on the peak pressure as a function of the number of valves that lift.

MR. KERR: That also assumes that pump trip occurs or is there --

MR. SAWYER: Yes. We trip not all of the pumps for ATWS. I believe we trip four of them.

MR. MICHELSON: I think the staff has a comment.

MR. MILLER: Dr. Carr, in response to a question you asked concerning the number; the staff asked the question to GE and has got a response here and George Thomas, the reviewer, would like to make remarks at this time.

MR. THOMAS: Of the 18 SRVs, only 14 are required to meet the ASME limit. So if four of them fail, they can still meet the ASME limit.

MR. MICHELSON: Thank you.

MR. KERR: Is that also the case for the ATWS situation or that the situation to which you refer?

MR. THOMAS: This is only for the ASME portion, not for ATWS. ATWS will require all of them.

MR. KERR: You mean the ATWS requires that all function in order to meet the limit.

MR. THOMAS: Yes.

MR. KERR: Thank you.

MR. SAWYER: I don't believe that's the -- I agree with -- when I said many, four sounds about right to me for this case. We're below 1375 for 18 out of 18 valves. So we clearly can withstand some failures, whether it's one or two valves in order to meet 1500. I'm certain of that.

MR. CARROLL: What's 1500?

MR. SAWYER: That's the emergency limit for the ASME code. ATWS is not considered to be a normal transient.

MR. CARROLL: I understand. How do you get from 1250 to 1500?

MR. SAWYER: 120 percent.

MR. CARROLL: 120 percent. Okay.

MR. SAWYER: If there aren't any further questions on the over pressure protection, for the next subject Mr. Dillmann is back up again to talk about reactor materials.

MR. CARROLL: I guess I did have one question on ATWS. How long does it take -- what are you assuming in the event of an ATWS event for the time it would take to become sub-critical over to standby liquid poison injection?

MR. SAWYER: Our position on ATWS is that we have supplied, in addition to the system we have, the ARI function plus the rod run-in function, both of which we take credit for in mitigating ATWS.

Now, the rod run-in function takes approximately two minutes, provided that that --

MR. CARROLL: To get the rods all the way in.

MR. SAWYER: Yes. To get them all the way in.

MR. CARROLL: You really only have to go sub-critical.

MR. SAWYER: Yes. To get to hot sub-critical, probably, it's going to be, I don't know, less than a minute. We have also provided the staff, per their question, the answer to what if that fails also and I have to rely on the SLC function only as part of the give and take on whether the SLC should be manually or automatically actuated.

We provided the staff with the peak suppression pool temperatures for both two-pump and one-pump SLC operation. And we get adequate results from the containment performance even in those cases.

MR. CARROLL: How long does it take to go sub-critical?

MR. SAWYER: As I recall, the power level begins -- until the standby liquid control system begins to inject boron, the power level is up around 20 percent. It decays to decay heat type of level in, I don't know, ten minutes, something like that.

MR. KERR: So the answer to Mr. Carroll's question is I don't know.

MR. SAWYER: It depends on what you mean by sub-criticality.

MR. KERR: That means sub-critical at the condition that prevails at the time of operation.

MR. SAWYER: If it's important to you, I can go look up the transient --

MR. KERR: I don't know whether it's important to him or not. I just didn't hear an answer and I was curious as to whether I missed something.

MR. SAWYER: I'm going on my memory of the case that was run.

MR. KERR: There isn't anything wrong with saying you don't know if you don't know.

MR. SAWYER: I was trying to give you a flavor for the order of magnitude at least.

MR. CARROLL: I'd be curious. From the time the operator turns the handle, assuming manual, until reactor sub-critical.

MR. SAWYER: I can certainly get that number for you.

[Slide.]

MR. DILLMANN: We have another brief section here on materials. Again, this is because we are following the organization of the SSAR. This is reactor coolant pressure boundary materials. Materials in the pressure boundary are carbon, stainless and low alloy steels. Low alloy steel is limited to the bolts and various places, including valves, and the RPV.

We have limited use of precipitation hardened material in valve spindles and stems. The material requirements, as I discussed a little while ago, apply to these materials also. We also employ prefilming of stainless materials to minimize radiation buildup. We nave some data that says that's effective.

We have specification on condenser tubes and the tube sheet to be titanium to control the introduction of oxygen through that mechanism. We also are looking at the hydrogen addition to mitigate IASCC.

While our materials and water chemistry controls, combined with our stress controls provide great margins against IGSCC, IASCC or radiation assisted stress corrosion we believe that the hydrogen water chemistry adds margin there.

MR. CARROLL: This titanium condenser is going to be designed as a titanium condenser and --

MR. DILLMANN: Yes. It's not a retrofit with inadequate tube supports. This is ground-up design.

[Slide.]

MR. DILLMANN: Here is a table from the SSAR that shows the water chemistry requirements. These are consistent with the latest technology on water chemistry and also consistent with the EPRI guidelines on water chemistry.

[Slide.]

MR. DILLMANN: Furthermore, specific corrosion-erosion resistant low alloy steels in places where we could have erosion adding metallic material to the coolant, specifically things like drain lines and heater staging lines. The non-metallic insulation applied to austenitic stainless steel has leachable elements controlled to avoid stress corrosion cracking and meets Reg Guide 1.35.

Ferritic material meets the impact requirements of the ASME code. We have welding controls, including controls on preheat and inner pass temperatures to avoid things like under --

MR. KERR: What leads you to believe that Reg Guide 1.35 is adequate?

MR. DILLMANN: I am not exactly saying it's adequate. I'm saying that we meet Reg Guide 1.35, but we have controls based on our -- on not only Reg Guide 1.35, but what we believe is necessary to avoid problems with leachable products out of the insulation.

MR. MICHELSON: When you say you have controls, are those controls identified in the SSAR as existing?

MR. DILLMANN: No. Again, they would be in the specifications. The SSAR would say that we meet Reg Guide 1.35 and that we have controls on leachable products.

MR. MICHELSON: I think the Committee has to think seriously about what we are actually going to end up certifying, and we have a number of promises about controls, but where are those certified promises? Again, it's the whole question of what are we going to end up certifying in the 1990 starting certification.

MR. CARROLL: You mentioned these were in specifications.

MR. DILLMANN: Right.

MR. CARROLL: Where do we stand on that, Charlie? Do they submit their specifications and do those become part of the certification?

MR. MILLER: No, they don't. 10 CFR Part 52 requires that the designer have available sufficient information to be able to write the specifications, but does not require the specifications themselves to be submitted.

MR. MICHELSON: But as I understand it, the licensing basis letter seems to tell me that the basic criteria, any basic important requirements have to be identified, I think, in the SSAR or some other document that's a part of the certification. Otherwise, later on in time, 20 years from now, a guy can do other things as long as he meets whatever the SSAR said, which is the only certified document.

MR. MILLER: One of the things that the staff has yet to evaluate and has to look into -- we've had several discussions with the Subcommittee about this -- is the information that will be part of what we call the test inspections and analysis. We're in the process of trying to develop a plan that goes along with that.

We had some discussions at the Subcommittee meeting concerning how are we going to assure that the level of detail is there and we're taking advice from the Subcommittee to look into that further. I'm really in the stages of trying to put together a plan. What I'm anticipating doing is -- we don't have the resources to go out and look at everything. I'm trying to put together a plan where we can go out and do some kind of audit inspection on these kinds of information to assure that it's available at the vendor shop.

MR. MICHELSON: I think, though, the thrust of the concern here is if we're not going to include the documents that are certified, how are we assured 20 years from now that all these good intentions are still being carried out? I think it has to be a part of the documentation that's certified and the staff hasn't decided yet what they're going to certify, and I was only trying to caution the Committee to realize that you don't know whether these good sub-tier documents are even a part of the certification.

MR. MILLER: We're going to have to continue to examine that and discuss that in future meetings.

MR. CARROLL: But we're trying to write letters now, of course.

MR. CARROLL: We can make the comment in our letter, I guess.

MR. MICHELSON: That's right.

MR. CARROLL: Going back to your water chemistry thing for a moment, this says that the limits given on that slide should be met at least 90 percent of the time. What are the limits for the other ten percent of the time?

MR. DILLMANN: That statement is in there to accommodate upset conditions and there's no specific limits. The basic guidance or requirement is that you, as quickly as possible, get back to these requirements.

MR. SHEWMON: Isn't there anything that says if our water gets crappy enough, we shut down?

MR. DILLMANN: Yes. We have shut down -- normally, when we get to the plant technical specifications there will be requirements in there for water chemistry before you can start up and water chemistry that would cause you to shut down after a specified period of time.

MR. SHEWMON: So that's at least in the BWOG water chemistry spec, which I presume --

MR. DILLMANN: Yes. Again, our basic requirements are consistent with the EPRI BWR owners' group requirements. We meet or exceed those.

MR. CARROLL: And they do have requirements before you can pressurize?

MR. DILLMANN: You can't pressurize unless you've met certain requirements

MR. CARROLL: In other words, thou shalt shut down and thou has so many hours to get back in specs if they're at this level.

MR. DILLMANN: Yes. All that is applied at the operating requirement level.

MR. CARROLL: But certification is going to require that that be established, isn't it?

MR. DILLMANN: I probably shouldn't speak. That's more of what the NRC has got to decide. Underlying all this, we have our own need to make sure that we do everything that's right and we have controls in place to control these lower tiered documents to that in 20 years they can't be changed.

We have strict procedural controls on changes and technical evaluation changes. I know that doesn't give you the overview that you would like, but it's there.

[Slide.]

MR. DILLMANN: We also have special welder qualification requirements on areas of limited accessibility. This is addressed to Reg Guide 1.7, but we have requirements that are somewhat different and perhaps a little more stringent than that.

We have heat input control on welds. We also prohibit electroslag welding.

MR. SHEWMON: Can you tell me, just out of curiosity, what those erosion-corrosion resistant materials are? Are they half-a-percent chrome or what?

MR. DILLMANN: Again, I'm going to have to beg the issue and get you an answer to that. I don't remember the exact compositions.

MR. SHEWMON: It's not stainless, though.

MR. DILLMANN: No. We use stainless in certain baffles and so forth for similar reasons, but no. It's an alloy content in an alloy steel.

MR. SHEWMON: Fine.

[Slide.]

MR. DILLMANN: In summary, as discussed earlier and consistently here in the pressure boundary materials, we have controls on processing fabrication contaminants, water chemistry. We've paid specific attention to cobalt sensitization, ductility, and IGSCC and IASCC and, of course, we comply with all codes and standards.

MR. REMICK: Let's take our morning break at this point, returning 20 minutes past ten.

[Brief recess.]

MR. REMICK: Please continue.

[Slide.]

MR. DILLMANN: The next subject is the coolant boundary leakage detection. We have leakage detection systems applied where required in the plant. Methods of detection include temperature pressure, radiation flow, Delta flow. The actions resulting from the leakage detection system include alarm and, in some cases, isolation.

The systems that are covered by leakage detection are primarily the main steam lines, high pressure core flooder system, residual heat removal system, reactor water cleanup system, feedwater system, coolant systems within the dry well, pressure vessel itself, and some miscellaneous small systems.

[Slide.]

MR. DILLMANN: In the dry well, small unidentified leaks, the primary method of detection is some pump activity and sump level. The detection capability is one GPM within one hour. There is continuous indication recording in the control room. There is no isolation trip from this system, but rather an alarm.

There are also other methods of detection; pressure and temperature in the dry well. High dry well pressure does cause isolation. High dry well temperature will cause an alarm only.

MR. REMICK: What does the word sensitize mean there when you say --

MR. DILLMANN: That's a typo. It should have said sensibility.

[Slide.]

MR. DILLMANN: We also have what we call small identified leakage. This is primarily from valve stems. It's leakage from a source where we expect leakage and have piped it to an equipment sump. The limit there is 25 GPM and when that limit is reached, we have an alarm.

The dry well system in total measures dry well temperature, temperature in the area of steam line guard pipes, high sump flows, high steam line flow rate, high dry well pressure, high fission product radiation. Also, as part of the leakage detection system, is the reactor vessel water level.

We have a temperature in the RPV head seal drain line so we know if that's leaking. We have the SRV discharge temperatures.

MR. CARROLL: How do you measure high fission product radiation?

MR. DILLMANN: We have radiation censors. I don't have the details of them.

MR. CARROLL: I can believe that you measure radiation. I'm not sure how you distinguish fission products from nitrogen 16, for example.

MR. DILLMANN: No. There is a distinguishing method, but I'm afraid I don't know the details. I can get you that information, however.

MR. CARROLL: Does the staff know?

MR. MILLER: Could you repeat the question, please?

MR. CARROLL: How high fission product radiation is measured in the dry well.

MR. CHANDRASEKERAN: We have particulate radioactivity monitoring detection system and also a noble gas radioactive monitoring system, leak detection system. I'm sorry. ABWR has these two systems.

MR. CARROLL: Particulate and --

MR. CHANDRASEKERAN: And noble gas. Radioactivity leak monitoring systems.

MR. KERR: Are you saying that you simply measure gamma radiation?

MR. CHANDRASEKERAN: I would think it would be gamma radiation.

MR. DILLMANN: Why don't I provide you the details separately.

MR. CARROLL: Okay.

[Slide.]

MR. DILLMANN: For leakage external to the dry well, the areas covered include the equipment areas in the reactor building, main steam tunnel and the turbine building. Within the reactor building, the parameters that are monitored include steam line flow rate, RCIC steam line flow, RVP water level, of course, is measured, high flow rate from the sumps, high equipment space temperature.

RCIC, RHR and the hot portion of the RWCS is equipped with -- and, of course, RWCS is reactor water cleanup system -- is equipped with high temperature measurements and alarms. The RCIC turbine exhaust; there are diaphragms in the exhaust pipe to protect it against over pressure and we measure the pressure in those diaphragms. They are double diaphragms and measuring the pressure between them tells you that the inner one is leaking.

[Slide.]

MR. DILLMANN: We measure high Delta flow in the RWCS. In other words, the difference between the flow to the system and the flow returning from the system must be within a specified amount.

We also, in the reactor building cooling water system, the discharge of the RHR reactor water cleanup, reactor internal pump and fuel pool cooling heat exchangers are radiation detectors so that if there is leakage from the radioactive side to the cooling water side, that will be detected.

We also have an RCIC low steam line pressure, in addition to the steam flow. Within the steam tunnel, we measure both radiation and tunnel air temperature and look for changes in either one. Within the turbine building, the measurements are steam line low pressure, low condenser vacuum, and high area temperature around the steam lines.

MR. KERR: Within the steam tunnel, it would seem to me that you always have high radiation.

MR. DILLMANN: The way that is handled is a background radiation limit is determined. In other words, the normal operating radiation is determined and then the alarm is set at a value above that.

MR. KERR: It just seems to me that nitrogen 16 would be so high that in order to see anything, it has to be a rather major source. I'll look further into that.

MR. REMICK: All your plants have high radiation monitors in the steam tunnel, don't they?

MR. DILLMANN: Right. This has been the standard BWR approach for a long time.

MR. SAWYER: Dr. Kerr is right. It takes the release of fission products from multiple fuel bundles to trigger that alarm. The purpose of the design --

MR. REMICK: But you are able to detect it.

MR. SAWYER: Yes. Sure.

MR. REMICK: Do you still use the same logic in those? They used to have the once out of the two twice in those monitors. Do you still have that?

MR. DILLMANN: I believe it's still the same.

MR. REMICK: Still the same logic.

MR. SAWYER: Let me correct that, Chuck. Everything we're doing in this plant is two out of four, including that.

MR. CATTON: Why not measure humidity or is that a dumb question?

MR. DILLMANN: I think that the temperature is probably quicker and more accurate than the humidity.

MR. SHEWMON: There are conducting tapes and other sorts of things that people have developed and are there. Whether you want to use them or not is a separate issue.

[Slide.]

MR. DILLMANN: The key features of the leakage detection system is that these leakage limits I've been describing to you, 1 GPM for unidentified leakage and 25 GPM for identified leakage, are well within the makeup capability of the RCIC system, which is 800 GPM.

Requirement on exceeding leakage rate is to result in orderly shutdown. We, of course, as I said, have differentiation between identified and unidentified leakage, wherein if we expect leakage from a location such as a valve stem, that's piped to a place where we can measure it, and that's identified leakage. Stuff going into the floor drains and sumps is unidentified leakage and has a separate capability.

The system is testable. Each censor has a testability capability. The system meets Reg Guide 1.45, but it also is based on consideration of the potential events and what is necessary to control or mitigate those events, as well as to control the leakage.

MR. MICHELSON: Let me ask a question before you remove that slide. My vague recollection about leakage requirement, leakage detection requirement was that you wanted to detect leakage that was within the normal makeup capability of your system.

My vague recollection is that normal meant non-engineered safety feature makeup capabilities. Clearly it must have meant that.

MR. DILLMANN: Yes.

MR. MICHELSON: Yet, you're now referring to RCIC within its makeup capability, but that's an engineered safety feature, not a normal makeup.

MR. DILLMANN: The RCIC in ABWR is classified for certain events as an engineered safety feature, but it is also classified as a normal makeup system. The history of the RCIC in the BWR is that it was originally installed to provide, as its name implies, an isolation cooling function and was considered to be a normal makeup system.

As time has gone on, it has also been used to mitigate certain events and has safety functions. So we consider it and have described it in the documentation, including SSAR, as having a dual role. It's an engineered safety feature for certain events. It's also a normal makeup system.

MR. MICHELSON: There is a Reg Guide that tells you what you're supposed to do on this, I believe, isn't there? Craig, do you have any knowledge of that? Isn't there a Reg Guide for leakage detection?

MR. DILLMANN: Leakage detection, yes.

MR. MICHELSON: Does the Reg Guide allow you to use engineered safety features as normal makeup?

MR. CHANDRASEKERAN: This particular section is -- we look into the compliance of not meeting the guidelines of Regulatory Guide 1.45.

MR. MICHELSON: Does it allow you to use an engineered safety feature in deciding what normal makeup consists of for leakage detection?

MR. CHANDRASEKERAN: That particular guide does not have any position on what should be the capacity of the makeup system, except that it is generally understood that the makeup capabilities should be available. But there is no quantification of any such number there.

MR. MICHELSON: I guess I could have then designed it for RHR level makeup instead of RCIC and made it even a much bigger leak before I had to detect it. I thought it had a threshold to keep you from using big systems and, therefore, lots of leakage in setting your detection threshold. I haven't looked at it for a long time.

MR. DILLMANN: As a practical matter, the amount of leakage we're talking about is a total of 25 GPM identified leakage.

MR. MICHELSON: That's why the 800 surprised me, because I thought you would certainly want to set some other nominally much lower threshold than that.

MR. DILLMANN: As a practical matter, that can be made up by the CRD pump.

MR. MICHELSON: I don't recall. In your SSAR, did you specify what the maximum leakage would be that you would be detecting?

MR. DILLMANN: In the SSAR, for unidentified leakage, we said 1 GPM or 5 GPM. However, in the latest amendment, we've made it solely 1 GPM, and that ties into our decision to use the leak before break criteria for piping.

MR. MICHELSON: So it's well within the control rod drive cooling mechanism to keep up with it.

MR. DILLMANN: Yes.

MR. MICHELSON: So why the RCIC?

MR. DILLMANN: We've used that for certain events and, again, we consider it to have a dual function. It's a normal makeup system most of the time. But as a practical matter, the 25 GPM identified leakage can be made up by the CRD system.

[Slide.]

MR. DILLMANN: The next subject is, again, materials; this time, reactor vessel materials. Again, the materials here are low alloy steel plate and forgings. The plate is SA533 and is not used in the beltline region. The forgings are SA508 Class 3. The fine grain practice, vacuum degassed. Copper is limited to .05; phosphorous .015; nickel 1.2 percent in the beltline forgings.

Weld metal is .08 percent copper limit; .02 percent phosphorous limit; nickel limited to 1.28. We require 100 percent UT examination to the requirements of ASME III, Division I. Fracture toughness, also to Division I.

The studs, nuts and washers are SA540 Grade B23 or B24.

MR. SHEWMON: Can you go back and reread that line and tell me what it means?

MR. DILLMANN: The nickel content is limited to 1.2 percent in the beltline, in the forgings, and 1.29 percent in the weld metal. That's in accordance with the latest findings --

MR. SHEWMON: Okay. I noticed the three numbers there. So the .08 is --

MR. DILLMANN: Yes. In the weld metal, it's in the same order as the previous line.

MR. SHEWMON: Thank you.

MR. DILLMANN: Of course, nickel has recently been found to be a contributor to radiation embrittlement, and that's why we have controls on it.

MR. SHEWMON: Do you have any idea what fine grain practice means chemically?

MR. DILLMANN: Chemically, it's more a matter of the control of the pouring, as I understand it, than it is of the chemistry.

MR. SHEWMON: They can do it by adding aluminum, niobium.

MR. DILLMANN: Yes.

MR. SHEWMON: Some of those will have a real impact on sulfur and others won't. That's why I asked the question. Now, at GE, is that something that's in Code Section 3 or Section 2 on materials or is that, again, something that's --

MR. DILLMANN: No. We specify it. Again, my memory fails me on what additives we allow and which ones we prohibit. I believe we prohibit aluminum, but I'd have to verify that.

MR. SHEWMON: If you prohibit aluminum, then you've got to use something that's very strong carbide form because you haven't got the nitrite form.

MR. DILLMANN: I'll have to verify exactly which one we use.

MR. SHEWMON: Okay.

MR. CARROLL: Again, that is something that's in your internal specifications?

MR. DILLMANN: Yes.

MR. CARROLL: Is this an example, Paul, of something you believe ought to be cast in concrete for purposes of certification?

MR. SHEWMON: The first line on that thing is sort of -- it meets the code; it doesn't meet good modern practice, because the code on those fields must be at least 40 years old from what they've got in for compositions.

So GE isn't going to -- and the customer isn't going to accept such a crappy thing. It bothers me some to sort of see these sub-minimal requirements put down in what we're approving when everybody knows that they should do a better job and will do a better job.

MR. CARROLL: So you think for purposes of a certified design that's going to be available for use for many years, things like that ought to be spelled out.

MR. SHEWMON: It would sure make me a little more comfortable.

MR. MICHELSON: Will you prepare a comment accordingly?

MR. SHEWMON: I won't use the word crappy.

[Slide.]

MR. DILLMANN: Again, as I said earlier, we have no welds in the high fluence zone. We use forgings there. Processing meets all code standards and regulations.

MR. SHEWMON: Where is that all welds? Is that in the document we have or did we agree that that also is just in GE's internal stuff?

MR. DILLMANN: I believe it's in the SSAR, but I'd have to check and see.

MR. SHEWMON: It wasn't in the part on materials.

MR. DILLMANN: I have the applicable sections with me. Let me look and see if I'm correct that it's in there.

MR. MICHELSON: Check which amendment you look at when you tell us.

MR. DILLMANN: I believe it was put in in response to some staff questions, that we clarified it at that point.

MR. MICHELSON: It may be in Amendment 8 and then, Paul may not have it.

MR. SHEWMON: No. All I had was Section 5.

MR. MICHELSON: Amendment 8 of Section 5.

MR. DILLMANN: Yes. We meet all codes, standards and regulations. In addition, we have further requirements based on the latest technology in general and results of our specific program.

We also include in the vessel surveillance specimens as required by 10 CFR 50, Appendix H, and ASTME185, and we have a withdrawal plan for those specimens that's derived from the

ASTM requirements, but allows continually monitoring of the radiation embrittlement to verify our calculations in the material response.

MR. MICHELSON: Could I interrupt just a minute and ask the staff a question. We have received a number of questions which the staff has asked GE and then we received the answers. I assume that every question has to have a documented answer.

What I'm wondering about is having seen now the documented answer, how do I know -- if the staff is unsatisfied with the answer, is there a new question asked then or is the old question still open?

MR. SCALETTI: Let me just briefly give you an overview of how it's done. We will identify, on our initial review, any questions that we have of General Electric. General Electric will then respond. Normally, the response comes in in a letter form, later on to be amended as part of the SSAR.

For expeditious reasons, we like to see it come in quickly in a letter form so that we can get to work on it. Again, you're right, some of those answers may not be satisfactory. However, the staff will then write a draft safety evaluation report identifying where they believe the deficiencies still exist.

It may identify the question specifically. It may just identify an open issue in the safety evaluation. This safety evaluation will go to GE, as you've seen in this one, identifying outstanding issues, which the staff will then resolve these, work out the resolution with General Electric or General Electric will work out the resolution with the staff and provide more information.

Now, some of this information on some of the outstanding issues has already been received. The staff has it under review now and maybe some of the issues have been resolved, but they're not reflected in this safety evaluation you have before you.

MR. MICHELSON: Eventually, do you go in and when you finally publish the question and the answer, and do you close out and indicate that this is accepted or it's still open on the answer, or do we ever know the answer was to your satisfaction?

MR. SCALETTI: The answer would normally be -- is satisfactory unless we have so identified in the safety evaluation.

MR. MICHELSON: But you don't identify it by a question, but rather by some open issue, and I'm having trouble looking at open issues and relating them to the questions to decide which ones are really not answered. There isn't that kind of connect in the paperwork I've seen.

MR. SCALETTI: All I can say is between the -- the issue has to be resolved either through the response to the question or through a revision to the text of the SSAR. You might have to go to both places to resolve this. A question, in some instances, may refer to a section of the safety evaluation.

Again, if the staff is more concerned about the issue, then the specific question and whether or not the answer to that specific question was totally combined within the response of that question or it could be spread out through the SSAR, as long as the information is there is what we're concerned with.

MR. MICHELSON: I see. It's a little hard for the casual reader to pick up on that.

MR. SCALETTI: It's hard for the staff to pick up on it sometimes.

MR. MICHELSON: Thank you.

[Slide.]

MR. DILLMANN: The next subject is reactor pressure vessel pressure and temperature limits.

MR. CARROLL: Before we move to that, the preceding subject was pressure boundary leakage. I note in the SER that there was a section on intersystem leakage, part of which seems to deal with intersystem LOCA concerns. We didn't talk about that. I wonder why that is.

MR. DILLMANN: We are talking about the leakage detection. Mr. Sawyer does have some material on interfacing system LOCA if you want to cover that at some point.

MR. CARROLL: Because this thing on Page 520 of the SER sort of brushes it all aside. It says other intersystem leakage, bla-bla-bla, is highly unlikely since this leakage would have to occur through closed check valves or containment isolation valves.

MR. MICHELSON: We questioned that at the Subcommittee meeting and didn't get much of an answer. It may be that we'll just have to pose it as a comment. You might want to put together a comment on it.

MR. DILLMANN: As I say, Dr. Sawyer has some material with him and when he gets into the ECCS systems, maybe he can cover that.

MR. MICHELSON: Okay.

MR. DILLMANN: We sort of anticipated that might come up.

MR. MICHELSON: Yes.

MR. DILLMANN: The reactor pressure vessel pressure and temperature limits. We've done a calculation based on the 10 CFR 50, Appendix G. The results of that calculation say that the bolt-up temperature can be 70 degrees F. In other words, normal room temperature.

We calculated the shift in RTNDT per Reg Guide 199, Rev. 2, which is the latest Reg Guide reflecting the latest experimental data on shifts due to radiation. And we calculated a shift of 28 degrees F for the weld metal and 8 degrees F for the shell.

The low shift is due to, one, the material having the control in the contaminants and, two, the low fluence. As we discussed earlier, the low fluence is primarily due to the large annulus. As a further calibration on that, I've compared the radiation on the internals, on the shroud specifically, with calculations from earlier plants and they are consistent.

So it's not a calculational thing. It is, indeed, due to the annulus. The reason we see more of a shift in the weld metal than in the shell is, of course, due to the nickel. We've done an evaluation of margin to non-ductile failure, looking at the worst upset cases.

There's a figure in here. The two limiting upset cases are that we would reach 1215 psig at 528 degrees F, while critical, and then after a scram, the pressure would drop to 930, but the temperature would drop to 250, but now we're not critical so that's a different requirement.

[Slide.]

MR. DILLMANN: These are the curves and they are presented in the form of pressure at the top head. Of course, the reason we specify top head is that at the bottom head, there's a little more pressure due to water level. The minimum reactor vessel temperature to avoid brittle fracture.

We have a curve for system hydro test with fuel in the vessel. There's another curve for non-nuclear heating, which is curve B, and a critical curve. The low temperature point, I mentioned earlier, the 930 at 250 degrees F, is this point right here.

As you can see, it's got quite a bit of margin in terms of pressure to curve B, which is the appropriate curve to compare that point to, and it has about 30 or 40 degrees F margin on temperature. In fact, it's almost on the curve for core critical. The core critical point, which is at 528 degrees F, of course, is way over here well away from any of these curves.

So we have margin to non-ductile failure looking at our worst upset events.

MR. CARROLL: The fluenc value given in here is for 40 year life or 60 year life?

MR. DILLMANN: This is for 60 year life. Even though current regulation or current law precludes 60 year life, we are doing our evaluations on 60 year life.

MR. SHEWMON: As I understood yesterday, the staff is doing their evaluation on that, too, but they just can't write a license for that.

MR. MILLER: We are prohibited by legislation, the NRC is, from issuing any license that's more than 40 years.

[Slide.]

MR. DILLMANN: The next subject is related to the pressure temperature limits. It's reactor vessel integrity. To assure reactor vessel integrity, we, of course, have the material controls we've discussed, fabrication controls, and operational margin, specifically identifying all the operation conditions and assuring that the design of the pressure vessel addresses each of those conditions.

Our design practices, of course, are to use ASME III Class 1 as a minimum, but we also add any other requirements that we know are necessary to assure a good operating condition. Again, a very important point is to make sure we address all the transients and the environmental effects.

Another portion of reactor vessel integrity is in-service inspection and then, of course, the surveillance program.

MR. MICHELSON: Before you leave that; Paul, do you have a question?

MR. SHEWMON: No.

MR. MICHELSON: I'm sorry. I do have a question. In the DSER, the staff indicates several reasons why they're happy with the vessel, one of them being that if you do get into trouble, it can be annealed. We discussed this at the Subcommittee meeting and it was our understanding GE says that it's not being designed to be annealed.

It perhaps might even be very difficult to anneal. I wondered if the staff is going to delete that particular statement from the DSER or what is the staff's position on annealing?

MR. SCALETTI: It was offered as an option that it could be done. The staff really does not have a conclusion on that right now. It will look into further to find out if it does really determine that annealing is a feasible practice for the ABWR. It will look at that in the context of the evaluation for the 60 year life of the vessel.

MR. MICHELSON: Because I kind of inferred from the DSER that you certainly had that in the back of your mind when you said the vessel was okay.

MR. SCALETTI: I guess probably that was in the context of given enough money and enough time and whatever and people willing to spend it, it probably could be done. We will address that further at another time.

MR. MICHELSON: Thank you.

MR. CARROLL: Is there something that GE has done that precludes annealing?

MR. DILLMANN: One thing that obviously comes to mind is that we have stainless material welded into the vessel. If we start to anneal the vessel, we will sensitize that material. I'm thinking primarily of the shroud as being welded in. The other large core structures are removable, but I'm not sure I'd ever want to remove them. The core plate and the top guide can be removed, but the question is would you ever be able to get them back in again once you remove them.

Basically, the design was not -- we didn't say one of the requirements for the design was that it be compatible with in-service annealing. Of course, in-service annealing is a very difficult process. On and off over the last 20 years, I've seen various proposals on how to do it and they all have obvious problems.

We don't believe it's required and the reasons are, in fact, on this chart.

[Slide.]

MR. DILLMANN: The RTNDT, at the end of life, is well below 200 degrees F. An evaluation of DBA in emergency core cooling events shows that there would be no risk of brittle fracture; in other words, no cold repressurization that can cause brittle fracture.

On that basis, we say the vessel would never be in a condition that would make it necessary to anneal.

MR. KERR: I am curious about the significance of the term DBA and ECCS. In your PRA, you probably use a value for vessel rupture which is comparatively low if it is like most PRAs. I don't think that those numbers are based just on consideration of DBAs and ECCS, are they?

MR. DILLMANN: The numbers that we use in the PRA, I believe, Craig, are based on WASH-1400, aren't they?

MR. SAWYER: No. Actually, they're based on fracture mechanic studies that have been done and taking into account leak before break and the critical crack growth and all those kinds of things, too.

MR. KERR: It seems to me that one -- if that number has any significance, it must take into account the possibility of events outside of the DBA.

MR. DILLMANN: Yes.

MR. SHEWMON: Has anybody in the room read Appendix G lately? Because Appendix G is where this comes and it's sort of -- at the end of the section where it says if you can't assure it's going to be above this toughness and that temperature, then you have to make provisions for annealing. But I thought it was a conditional statement. So I'm a little surprised you haven't said we haven't met the -- we've avoided the conditions under which that would be required.

MR. DILLMANN: Basically, I think those conditions are what I have here, but we've gone at it more from the technical point as well as the --

MR. SHEWMON: It couldn't have been core cooling shocks because this was before -- it was written long before PTS even came out. I doubt if DBA comes into it, but apparently we don't have anything with the appendixes here.

I think what -- to defend GE, not that they aren't capable of defending themselves, but my impression is that -- I asked him early on how they could justify such low fluence or changes in RTNDT, and he said we've got basically an extra foot of water out there and that cuts the fluence down by an order of magnitude or something.

So if, indeed, the water is there, it should do this, then they've got very low flux out there at the core.

MR. MICHELSON: But the staff still put it in their argument as to why the vessel is okay, and I'm just saying I think the staff ought to take it out of their argument or at least settle the issue. Do you or don't you have to anneal?

MR. SCALETTI: We will settle the issue to our satisfaction and hopefully yours, and if it requires taking that statement out of the safety evaluation, we certainly will do that.

MR. KERR: Who in the U.S. is capable of fabricating this vessel that you described?

MR. DILLMANN: Today, I don't believe there is anybody in the U.S. capable of fabricating this vessel. Babcock & Wilcox has closed their facility at Mt. Vernon. We've closed our facility that we had at CBIN, and I understand Combustion Engineering has closed their facility.

There are offshore sources for these vessels. Of course, we are building these vessels starting next year in Japan for the Kashiwazaki Project. I am unaware of anybody in the U.S. that can build these vessels.

MR. KERR: Thank you.

MR. CARROLL: You mentioned the difficulty in getting the core support plate out.

MR. DILLMANN: Yes.

MR. CARROLL: What happens if you drop a fuel assembly and distort the core support plate?

MR. DILLMANN: We'd have to repair it or replace it.

MR. CARROLL: So it is a great big core support plate.

MR. DILLMANN: It is a big core support plate and it would be --

MR. CARROLL: It's not the individual plates that sit on top of the guide tubes.

MR. DILLMANN: Well, the core support plate is a big support plate and the support it provides is lateral support. The vertical support is, as is typical of the GE BWRs, it's the fuel support casting sitting on the guide tube, the guide tube sitting on the housing.

But if you drop something really heavy, you could distort the core support plate, but I can't conceive of distorting it so bad you'd have to replace it. You might have to go in and rework it under water, but not replace it.

MR. SHEWMON: Let me read, if I may, from Appendix G fracture toughness requirements. The last paragraph says reactor vessels for which the predicted values of upper shelf energy at end of life is below 50 foot pounds, or for which the predicted value of the adjusted reference temperature at end of life exceeds 200 degrees F, must be designed to permit a thermal annealing treatment at a sufficiently high temperature to recover material toughness properties of the ferritic materials in the reactor vessel beltline.

If you don't use that high sulfur steel to make it out of, you'll meet your upper shelf requirement and you've showed that you're not going to get enough radiation to have trouble with it above 200 F. So it seems to me unless the staff has another regulation they use to supercede this, that you've met the requirements to avoid annealing.

MR. DILLMANN: We've met those requirements, but then, in addition, we looked at the potential for cold stress or stress while cold and said we have no problem with that either.

[Slide.]

MR. DILLMANN: The next topic, reactor research system. This is another one of the features that makes the ABWR an advanced de-ign, though, indeed, these features have been used in Europe for several years.

We have ten internal pumps, rather than the large external loops with large external pumps. This is based, as I say, on European experience. These pumps are driven by adjustable speed drives and the pumps alone, with those adjustable speed drives, can provide load following over the range of 70 to 100 percent power.

The drives are solid state controlled units. The system also includes a recirc pump trip, as Dr. Sawyer mentioned earlier in conjunction with ATWS. The system also -- each pump has its own solid state power supply, but six of those power supplies are powered by MG sets. The MG sets include flywheels to keep their inertia up.

This enhances coast down in the event of a total loss of power to the recirc pumps, including loss of power to those MG sets. The MG set has a long coast down due to its inertia and continues to power the recirc pump during that coast down. This provides thermal margin in the event of all pump trip event.

[Slide.]

MR. DILLMANN: The RIP itself, the reactor internal pump, is again a designed based on European experience. Specifically, it is most closely related to the pumps used in plants in Sweden, at Forsmark and Oskarshamn.

Improvements have been incorporated in those pumps. Basically, the motor voltage was changed from 800 volts, as used in the European pumps, to 3KV, allowing smaller wiring and less congestion in the under vessel area.

The reason this change was feasible is improvements in the thyristor technology in the last ten years allowing use of higher voltage.

The other major improvement is the bearing design. The European pumps have some problems with bearing stability. We went through an extensive test program in conjunction with our partners, Hitachi and Toshiba, and came up with improved bearings that eliminate the problem.

The pump also includes a backseat and an inflatable seal for servicing.

MR. KERR: Is that an induction motor?

MR. DILLMANN: Yes.

[Slide.]

MR. DILLMANN: Let me lead you through a picture of the pump, starting with the pumping end. This, of course, is the pressure vessel knuckle region. This standpipe is actually fabricated right out of the base material. That is not a welded-on part.

So the basic forging is like this. This is machined out as part of the forging fabrication. There is a weld at this location that attaches this pump pressure housing. The pump itself consists of a diffuser. This diffuser is this part and, of course, this part and the veins here connect these parts. The diffuser is held to the stub tube by what we call the stretch tube that runs down through this annulus and there is a big nut at this location. So that diffuser is clamped across this area.

The impeller is this part. The impeller shaft -- the upper end of the impeller shaft is hollow and not filled with water. It's filled with air. It's seal-welded at this location. The reason for the hollow shaft is to improve the shaft critical speed.

These pins here are for alignment of the tooling used to remove the diffuser and/or the impeller.

MR. KERR: What is a shaft critical speed?

MR. DILLMANN: The shaft critical speed is about 2700 RPM, where the pump's maximum speed is 1500 RPM.

MR. KERR: I'm asking a much more naive question. What is meant?

MR. DILLMANN: Shaft critical speed is the point at which you get the shaft in residence with the unbalance.

MR. KERR: Okay. Thank you.

MR. MICHELSON: So if you leak water into that enclosed shaft, I guess you realize it when you start vibrating more?

MR. DILLMANN: You probably would not pick up enough vibration due to the water. The main reason for keeping the water out of there is to avoid a trapped water regi .

MR. MICHELSON: But let's assume for the moment that I have a leak in whatever sealed up the shaft. How do I know I have a leak and what difference would it make anyway?

MR. DILLMANN: It would make very little difference and you would probably never know it unless you removed the shaft from water and saw water coming back out of the leak.

MR. MICHELSON: Or unless it caused the shaft to break.

MR. DILLMANN: Yes. I think that's a low probability event, however. This little knob up here, again, is a locating feature for the tooling and is used to grapple the impeller out of the reactor.

When I talked about the backseat feature, the backseat feature is at this location. When the shaft is lowered, it contacts actually the stretch tube at this point, forming a backseat to avoid water dropping out when the pressure boundary is opened.

The secondary seal is at this location and is an inflatable seal that goes in against the shaft. The procedure for removing the motor then is to remove this small plug here and loosen this bolt here which allows the shaft to drop down and backseat.

Once it has backseated, the inflatable seal is inflated and the motor casing can then be drained, the large motor cover removed, and the motor lowered. If you then want to remove the impeller, you put the blind flange back up here and remove the impeller upwards out of the reactor.

The motor is what we call a wet motor pump. It's not a canned motor pump. The windings are actually in the water.

This technology basically goes back to boiler circulators starting in the 1930's. So it's not new technology.

There's a purge flow introduced at this point going up the shaft. The purpose of that purge flow is to avoid contamination from the reactor coolant coming into the motor housing.

MR. WARD: Under what conditions can the motor be removed? Is that fuel in the tank and depressurized?

MR. DILLMANN: During any outage. The normal practice in Europe and the practice we're planning on is -- well, the practice in Europe has been every four years. Our practice will be every five years to remove the motor for inspection and refurbishment as necessary.

Primarily, the thing that controls that are the elastomers in the inflatable seal. The motors, in general, have required very little maintenance based on European experience. Other than when they mishandled a couple and damaged the windings, there has been no requirement for motor replacement.

MR. MICHELSON: What happens when that inflatable seal deflates? What is the consequence?

MR. DILLMANN: The seal is normally deflated in operation. It is only inflated for refueling. Excuse me. It's only inflated for motor servicing. If it deflated during that servicing, the backseat would still be holding the leakage, but it would start dripping. It would be similar to what happens when you're servicing the CRDs under a reactor.

MR. SIESS: Excuse me. You said the seal is only used when you service the motor and you only have to service the motor because of the seal? Those are the words I heard.

MR. DILLMANN: That is exactly the situation. The seal is there to allow servicing the motor. However, the servicing interval at which motors are removed is based on the life of the elastomers in the seal. It's a dichotomy of a sorts.

MR. MICHELSON: So when you remove those elastomers, you just tolerate the dripping; is that the idea?

MR. DILLMANN: Yes. When you change the seal, you get some dripping and the normal way that is handled is there's a funnel arrangement that collects it and routes it off to the sump. There's a whole set of special tooling used to accomplish this servicing.

MR. MICHELSON: So those inflatable seals aren't really used very often to help you with your work, because most of the time, your work consists of getting the seals out and replacing them.

MR. DILLMANN: What they do is they keep it dry in here when you're doing work on measuring the bearing clearances and so forth, and then you only have to put up with the dripping while you're changing the seal, which is a very quick operation.

MR. MICHELSON: How often does that have to be done, measuring the bearing clearances?

MR. DILLMANN: It's only done on this five year interval. Experience in Europe and the earlier plants, bearings have generally never been replaced in ten years, with one exception. In the later plants, as I said, they had a problem with bearing stability and the bearing life has been two to four years

MR. MICHELSON: Sounds almost like you don't need the seals, then; therefore, why are they even in there?

MR. DILLMANN: Basically, one of our guidelines has been don't depart from past practice without a strong reason. We don't see a strong reason to depart.

MR. MICHELSON: You're departing, of course, on the motor voltage, which is a significant departure and not necessarily even proved by other than, I guess, some small tests. The traditional motors are much lower voltage.

MR. DILLMANN: Today, there is quite a bit of industrial experience at those voltages and higher with the thyristors.

MR. MICHELSON: With that type of motor?

MR. DILLMANN: With this type of motor, yes. Those voltages are not --

MR. MICHELSON: In water.

MR. DILLMANN: -- unusual in the boiler circulators. The boiler circulators, in the earlier days, were driven by MG sets and used the higher voltage. It was the thyristor power supply that controlled the voltage in the early European plants with internal pumps.

MR. MICHELSON: So you do have the high voltage experience with the in-water units.

MR. DILLMANN: We have high voltage experience within water, motors in boiler circulators, and we have high voltage experience with the thyristor controls in industrial applications.

To rap up on this. Here is the upper journal bearing, lower journal bearing, and the thrust bearing is down here. The other feature in here is, at this location, there is a sprag device such that if a pump is tripped, it will not rotate backwards due to the core flow.

If one pump is tripped, then the flow is backwards up through this pump. We don't want it to rotate backwards because having it stopped gives an increased resistance and lowers the decrement and core flow due to that pump being stopped.

[Slide.]

MR. CARROLL: What is the experience on this anti-rotation device? Has that been in the design in the past?

MR. DILLMANN: That has been in the design and it had some teething problems in that on a couple of pumps, when they tripped, the anti-rotation device stuck and the pump started to run backwards and then the centrifugal force caused the anti-rotation device to lock up, and it locked up at high speed and jammed, which precluded restarting the pump.

A retrofit was accomplished using stiffer springs that prevented it from jamming and there has been several years of experience with that retrofit with no recurrence of the problem. So we feel pretty -- and this is actual plant experience, not our test program. So we feel pretty confident that that problem is over with.

MR. CARROLL: How do you know that the pump has stopped rotating when you trip it?

MR. DILLMANN: We would pick up the reverse rotation on the speed censor.

MR. CARROLL: And suppose somebody tried to start the pump when it was rotating backwards?

MR. DILLMANN: It most likely would start, because the procedure for starting a dead pump is you run the speed of the other pumps down. It most likely would start. If it didn't start, the only consequence would be the circuit breakers would drop out. The most likely event is that it would start.

We looked at a set of abnormal events in sort of an FMEA that we've submitted to the staff. It included missile potential. In other words, we said supposing the impeller comes apart at high speed, what would happen. We did an evaluation and said it would not do any significant damage and certainly would not cause pressure boundary damage.

We've looked at short circuit. We see no problem with short circuit. Loss of cooling; we've tested both loss of cooling and loss of purge and find that we don't overheat the motor. Natural circulation takes care of it.

We looked at casing failures and find no significant failure. In fact, a complete failure of the casing-to-vessel weld can be made up by the RCIC plus CRD in the worst case, and we don't expect the worst case.

MR. MICHELSON: Before you leave that, are you going to tell us about those motor restraint rods?

MR. DILLMANN: Yes. That's what I was going to get into now. Unfortunately, I don't have a picture of them.

MR. SCALETTI: Was the loss of cooling and loss of purge evaluated by test?

MR. DILLMANN: Yes.

MR. KERR: Presumably, if somebody miswired the motor, the anti-rotation device would keep it from running backwards and you'd just blow a breaker.

MR. DILLMANN: Right.

[Slide.]

MR. DILLMANN: Let me talk about failure of the weld.

If we have a hypothetical failure of the weld here, there's a whole set of things that mitigate that failure.

First of all, the failure is a low probability. But if that failed and it was a complete guillotine failure, the first thing is that, as I said, we have this stretch tube whose main purpose is to hold down the diffuser. But that stretch tube also is like a long bolt running from here, from this ledge here, across to this ledge here. It spans that weld. It's a strength member spanning that weld.

An evaluation of the blowout load and the resulting stress in that stretch tube said that that stretch tube would not over stress, that it would not yield, and that it would hold the joint together.

However, if that somehow failed also -- so now we have the weld failure and the stretch tube failure --

MR. CARROLL: Point out where the weld is for me.

MR. DILLMANN: The weld is right -- where this undercut is, the weld is right in this member opposite that undercut. That undercut is there to provide a clearance for the backside of the weld so we get good fusion back there.

So if the weld failed and the stretch tube failed, then this starts to drop. When it drops, the impeller comes down and backseats at this point, and now we have the impeller shaft, the impeller end shaft holding it, and down here we have the impeller with a bolt up through the thrust disk, and we'd have the whole casing and the motor sitting on that thrust disk and held by that impeller.

The weak link in that is this bolt. We've analyzed that bolt and that bolt can withstand the blowout load also without over stressing. However, suppose that also failed. So now we have a failure of the weld, a failure of the stretch tube, and a failure of this bolt. Now the casing could drop.

However, not shown in this drawing, we have support lugs on this casing and we have support lugs on the vessel. There are two lugs on the casing and two lugs on the vessel, and between them we have long stainless steel rods. Those stainless steel rods are designed to take a combined blowoff load and the torque of the motor at full power.

So the housing will drop down against those rods and, if the motor doesn't trip and it tries to twist, it will not over stress those rods. Those rods are designed to the same criteria and are specified to the same criteria as --

MR. MICHELSON: How can just two rods do that?

MR. DILLMANN: They wrap around. They are to the same criteria as pipe restraints. So that does two things. That prevents, first of all, this casing from becoming a missile and destroying anything under the vessel, like the scram lines. It also keeps the casing up in the hole. It keeps the shaft up in the hole.

So in the worst case event, if we don't have the shaft backseated and we have leakage down this path, our analysis says that that leakage is still within the makeup capability of the RCIC plus CRD system.

So the summation is, against this pump ejection, we have several redundant. First of all, the weld has to totally fail. The stretch tube has to fail. The shaft or shaft bolt has to fail. Then we still have the shootout protection.

MR. MICHELSON: If the shaft bolt fails, where does the shaft move to?

MR. DILLMANN: The shaft would still be down and backseated.

MR. MICHELSON: Now, the only other possibility, and it's a remote one, indeed, I guess; if, indeed, the shaft has been leaking and it's been filled with water for the last ten years and it's getting weaker and weaker but hasn't broken yet, this might be the time when it would break. But that's still kind of a limited leak, isn't it, at that point?

MR. DILLMANN: It's still a limited leak. We still would expect to be backseated up here. Of course, if we had a total failure of this thing somehow, despite the redundancy on top of redundancy, our large ECCS systems can handle that break.

MR. MICHELSON: How big a break is it, then, if just the shaft cross-sectional area, for instance, were a hold?

MR. DILLMANN: Do you remember the exact number on that, Craig?

MR. SAWYER: I don't remember the shaft cross section, but if you take that stub tube and say that cross section, it's 450 square centimeters.

MR. MICHELSON: That was the bigger one. That's still reasonable. Are you going to eventually describe those rods in the SSAR or don't you think they're worthy of description? You're giving me lots of good arguments on how they're nice, but why don't you describe them?

MR. DILLMANN: My memory says they are described. Let me check that.

MR. MICHELSON: No, they're not. At least not through Amendment 7 they weren't described.

MR. DILLMANN: We certainly should at least give an outline description of that feature.

MR. MICHELSON: I would think the staff would also evaluate that feature and say it looks good or if they have a problem with it.

MR. CARROLL: Am I supposed to find something about coolant pump blowout in Section 541?

MR. DILLMANN: I don't believe there's much in there about that.

MR. SAWYER: I think it was a response to a question from the staff where we responded on the failure modes and what defense and depth we had.

[Slide.]

MR. DILLMANN: The question came up both from the staff and in the previous Subcommittee meetings, and that's why I'm presenting it today. This discussion is not contained in the SSAR.

MR. MICHELSON: Although it does seem worthy of being in the SSAR, in my opinion, at least.

MR. CATTON: I missed the cross-sectional area if you had that pump blowout.

[Slide.]

MR. DILLMANN: If the stub tube blew out -- the question was the shaft and neither Craig nor I can remember the shaft. But if this was open, that would be 400 square centimeters -- 450 -- where, of course, about 650 is a square foot.

MR. SAWYER: Not to nitpick, but it's more like 900 is a square foot.

MR. DILLMANN: 25 times 25, right? I'm sorry. You're right. A square foot is about 900. So we're talking roughly a half a square foot. A little less than half a square foot.

[Slide.]

MR. DILLMANN: Other research system features. As I said, we have the purge system to maintain low contamination. Again, that's a servicing personnel exposure point, not a safety concern or an operational concern.

We have a core flow measurement system. Most of the measurements are pump Delta P, but we also measure core Delta P, and we use core Delta P for certain functions, plus we use core Delta P if we were operating with several pumps out of service, which is a capability we have.

We have capability for high power with one or more pumps out of service. We have a firm requirement for 100 percent power with one pump out of service. We've done evaluation with as many as three pumps out of service and show that we can operate 80 to 90 percent power without exceeding any safety limits and with adequate thermal margin.

The plants in Europe, the Forsmark plant specifically, has eight pumps where we have ten and they've operated for several months with two pumps out of service at power levels up over 80 percent. So that's an operational advantage with this system.

MR. MICHELSON: As a clarification of nomenclature, I sense that you do not have any kind of piping called reactor coolant system piping. Is that correct? There is a standard review plant section to discuss reactor coolant system piping and I didn't find it discussed, but I assumed it's because you don't think you have any.

MR. DILLMANN: We talk about coolant pressure boundary.

MR. MICHELSON: But do you have any piping in the --

MR. DILLMANN: There is no research system piping, per se, other than the cooling pipes to and from the heat exchangers for the RIPs.

MR. MICHELSON: So you don't have anything called reactor coolant system piping. Is that correct?

MR. DILLMANN: Not by that name, no.

MR. MICHELSON: That's what I concluded in reading the SSAR.

[Slide.]

MR. DILLMANN: At this juncture, Craig comes back and will talk about RCIC, RHR, reactor water cleanup.

MR. CARROLL: One other internal pump question. Has anybody figured out any safety problem having a high voltage winding literally inside the reactor vessel, source of an arc under accident conditions with respect to hydrogen?

MR. DILLMANN: With respect to hydrogen, we talked about that to quite some extent in the Subcommittee meeting a little over a year ago. The questions that had been raised at that time were more from the standpoint of arcing damaging the pressure boundary or the damaged material causing problems.

If there was some hydrogen generated, it would go up that annulus into the reactor and I wouldn't see it causing any problems. Our evaluation of the --

MR. CARROLL: I've got a moist vessel or a dry vessel and I've made some hydrogen and then somehow or other I --

MR. MICHELSON: Post-accident, I guess, you're referring to.

MR. CARROLL: Yes, a post-accident situation. Now I've got a detonator in the bottom of the vessel.

MR. DILLMANN: I see. You're going at it a different way. I wouldn't see that being a problem. Have you got any comments on that, Craig? I would suspect by that time, by the time you had that situation, the pumps would have been long tripped and no power to them.

MR. SAWYER: That was the comment I was going to make. If you get an arc, the first thing you're going to do is trip out a breaker. So you're not going to produce very much -- you're not going to deposit very much energy of any kind, much less a decomposition of water.

MR. DILLMANN: The event that I think is being proposed here is we have a core damaged and we've generated hydrogen in the vessel and we got the pump there as an electrical source to cause it to explode.

MR. SAWYER: The pumps are long gone.

MR. DILLMANN: The pumps would be long energized before we had that hydrogen.

MR. SAWYER: Right.

MR. WARD: Maybe in accident management you're trying to restart a pump for some reason.

MR. MICHELSON: This is severe accident recovery, now. Hopefully the vessel isn't dry when you start the pumps.

MR. SAWYER: I think I'd be very careful about the conditions under which I decided to start the pump if there's no water in the vessel.

MR. KERR: Where are you going to get these sparks?

MR. CARROLL: A failure of the motor winding.

MR. DILLMANN: I think you're looking at a sequence of events where you've had a core damage, you've generated hydrogen. You've also damaged the pump and now you try to restart it. It would seem to me that, at the first point, you wouldn't be restarting unless you had a higher degree of assurance you had water at a level above the pump and that water would pretty well protect you against having hydrogen down in the motor housing.

MR. CARROLL: Why are the pumps tripped at that juncture?

MR. SAWYER: The time we get to Level 2, which is way, way above the core, we've tripped all the pumps.

MR. MICHELSON: I think of somewhat different concern, but along the same line, though, have you looked at the maximum energy deposition in that pump area from a fault that's uncleared and does it cause enough pressure to rupture the pump housing or just what does it do?

MR. DILLMANN: We presented the assessment of that to the Subcommittee last year and came --

MR. MICHELSON: I don't remember that far back, I guess.

MR. DILLMANN: I have the material with me. I can take a look at it.

MR. MICHELSON: I knew we discussed the speed control and all that sort of thing. I don't recollect this other. But is that a part of a formal safety evaluation then?

MR. DILLMANN: No. We responded specifically to the question from your Subcommittee in that area.

MR. MICHELSON: So it's not a part of the docket at the present time.

MR. DILLMANN: No. It was not a question brought up by the staff.

MR. MICHELSON: Did the staff ask that question?

MR. DILLMANN: No.

MR. MICHELSON: It still seems like a reasonable question to ask to be documented for posterity as to why you don't have a problem with it. It seems to me at least to be reasonable.

MR. DILLMANN: Craig.

MR. KERR: Carlyle, are you assuming that the breaker does not work?

MR. MICHELSON: Yes. That would be the case, yes. It was a seismic event that maybe started this and, at the same time, the fault appeared. I'm just wondering if it had been looked at on a strictly analysis basis, just on the assumption you didn't clear the fault. I think the single failure has to be considered.

MR. KERR: I think you'd just have a nice hot water heater, it would seem to me.

MR. MICHELSON: I don't know. I'd like to see the analysis. I'm not an expert enough to know what it would look like if you had an electrical arc to ground inside that pump and didn't clear it.

MR. KERR: Are you assuming that water is there?

MR. MICHELSON: Yes. Water is there.

MR. CARROLL: Starting. Water may be there at the start.

MR. KERR: Then you'd just have a nice water heater, it would seem to me.

MR. CARROLL: You can --

MR. MICHELSON: A very rapid water heater.

MR. KERR: Well, you have a lot of water.

MR. MICHELSON: The pump is pretty small when you talk about the energy. I'd just like to see the analysis. If it's just a hot water heater, great. Then it will bleed off fast enough. An analysis has apparently been done and they've concluded it's a non-problem and I just think that's an important conclusion.

MR. CARROLL: That probably ought to be documented.

MR. MICHELSON: Yes.

MR. SAWYER: So noted.

[Slide.]

MR. SAWYER: The next three sections talk about systems attached to the vessel that do various functions, the first of which is the reactor core isolation cooling system, which, as Chuck mentioned before, is a dual function system.

It has a function to deliver reactor water makeup during isolation transients with loss of feedwater. It participates as part of the ECCS network in LOCA events and it's also available to handle station blackout, loss of all AC power events.

It's supposed to provide sufficient flow to avoid the need for emergency system initiation during normal transients, as in isolation events, and it supports the LOCA objective no fuel uncovery.

Summary of the features. It's one 800 gallon per minute system, driven by a turbine. It's upgraded in ABWR from pervious BWRs to be part of the ECCS network. That didn't require any major changes since many of the pieces of that system were already safety grade in BWRs 5 and 6 anyway.

Primary suction is from the condensate storage tank. The backup suction is from the suppression pool. There is automatic transfer capability with manual override, which might be required, for example, during a station blackout of extended time.

Condensate storage tank level switches are seismically installed. It has auto restart capability.

MR. CARROLL: What does that mean, seismically installed?

MR. SAWYER: That means that the switches will survive an SSE, even though the condensate storage tank itself might not, so that you'll get your automatic transfer.

MR. CARROLL: All right. Thank you.

MR. SAWYER: This has auto restart on low water level. The Level 2 is the set point for initiation of the RCIC. It will cycle between Level 2 and Level 8. Depending upon how much the decay heat load is, eventually it will overcome decay heat and, without operator action, it would increase water level to Level 8.

At that point, the system would trip and then when the water level buckles back down to Level 2, it would restart. The system initiation does not require any AC power. We've increased the turbine exhaust back pressure operation somewhat over the existing plants based on our severe accident review and also to support small break LOCA mitigation so that we can handle exhaust pressures up to 50 pounds, back pressure from the containment, and still provide the system function.

It has a bypass start feature, which is basically a small steamline which is used to get a start without causing overspeed trip problems, which were a bug-a-boo in some of the earlier RCICs in the existing plants.

MR. MICHELSON: Before you leave that slide, we discussed at the Subcommittee a little bit the question of if you don't have any AC power during operation, which you claim you don't need it, how do you control the environment around the turbine and so forth?

MR. SAWYER: It's passive. The environment is passive. There's enough heat sink, including the cold water pipe that it's pumping, which is actually a bigger pipe than the steam pipe which is supplying it. We've done room heatup evaluations and have shown capability to meet the room environmental conditions for at least eight hours.

MR. MICHELSON: That's the answer which we got before. I went back and read the SSAR again and I can't find any words that say you've done all this.

MR. SAWYER: I don't know that we've provided a specific station blackout evaluation for the SSAR. Do you remember?

MR. MICHELSON: Do you intend to do one?

MR. SCALETTI: I don't recall.

MR. MICHELSON: Do you intend to do one?

MR. SAWYER: I'll take a note on that. We have done a station blackout evaluation. I just can't recall whether we've documented it for the staff's review.

MR. MICHELSON: I would think that that would be an important basic document to any certification process, but I didn't find it. We'll leave it open. Thank you.

MR. REMICK: Could you elaborate the next to last bullet there where it says to support small break LOCA mitigation? I'm not sure I understand that.

MR. SAWYER: It is actually to support more than that. One of the functions of the RCIC system, which I have on the next page, is all by itself to avoid initiation of the other pieces of the ECCS compliment, including a one-inch line break. Now, why one inch? Because that's the size of our instrument lines. We don't want to have a small line like that cause the full ECCS compliment to come on.

But if you have a small line break, it can pressurize until the operator takes control and depressurizes the reactor to bring it to a normal shutdown. During that period of time, the small break is pressurizing containment potentially.

MR. REMICK: I see. Okay.

MR. SAWYER: Originally, the set point was 25 psi. We cranked it up to 50 not just for this reason, but also to handle an extended station blackout.

MR. REMICK: So that turbine does exhaust, then, into containment.

MR. SAWYER: It exhausts into the suppression pool.

That's correct.

MR. REMICK: Into the suppression pool. So it is quenched.

MR. SAWYER: Yes.

[Slide.]

MR. CATTON: What's the probability of a pump blowout? The internal pump.

MR. SAWYER: Did we quantify the probability level of as far as your evaluation, Chuck? Did you come up with a number?

MR. DILLMANN: I believe we have, but I don't remember what the number is.

MR. SAWYER: Neither of us can recall the number, but as part of that study that went through all the failure modes and how many welds had to fail, I know it was quantified. I just can't recall what the number was.

MR. CATTON: Okay.

MR. SAWYER: We'll get you the number. As I mentioned, for a normal isolation transient, we don't want to initiate even the other high pressure ECCS, the HCPFs that come on at Level one-and-a-half. We've demonstrated that it will do that. As I mentioned, all by itself, even without credit for the HPCF, one of its requirements is to be able to prevent Level 1 ADS and low pressure ECCS initiation.

Based on the design changes that have been made to the system, many of which have been backfitted to operating plans, we now think that this system has about a 97 percent starting and running reliability. That's based not just on analytical calculations, it's based on test information accumulated over the last four or five years with plants that have made things like the bypass start initiation feature as part of their retrofit.

MR. KERR: So about once out of 30 it would be expected to fail.

MR. SAWYER: That's correct. Well, once -- yes. And there are a variety of reasons why, including the fact that it itself might be down in a limiting condition of operation window when the demand comes.

MR. KERR: Sure. Is that an acceptable reliability as far as you're concerned?

MR. SAWYER: As far was we are concerned, it's quite acceptable and that's the number that we've used in our risk evaluation.

MR. MICHELSON: Before we leave RCIC, let me ask Ivan. You wanted to see the probability number of that event, is that what you were looking for? The internal pump, for internal pump blowout.

MR. CATTON: Yes.

MR. MICHELSON: You didn't want to see the missile study.

MR. CATTON: No.

MR. MICHELSON: I have one question on RCIC which we did discuss in Subcommittee, and maybe you can tell the Full Committee just whatever your view is. The SSAR says that the RCIC is designed for 30 minutes of operation. That seems like a strangely short number for the loss of all AC power, that is. That seems like an extremely short time. What is your reason for stating it that way in the SSAR?

MR. SAWYER: I recall that statement, and you are right. It's inconsistent with the station blackout evaluation. The difference is minor. It's a question of design basis evaluation, hands-off, no credit for operator, for example, disconnecting unnecessary battery loads on the division that supplies the RCIC and so forth. In other words, no credit for -- a design basis kind of evaluation as opposed to an evaluation basis which takes credit for reasonable operator actions, which is what we do for the station blackout evaluation.

MR. MICHELSON: None of those caveats, of course, appear in the SSAR, just a simple statement says the design basis for RCIC is only 30 minutes of operation or it is 30 minutes of operation during a loss of AC power.

MR. KERR: Decay heat is down to less than two percent at that point.

MR. MICHELSON: Yes, but is 30 minutes all we expect the design basis for RCIC to be? Is that a reasonable number?

MR. KERR: I don't have --

MR. MICHELSON: I would have thought the staff would evaluate the 30 minutes and comment on it, but I couldn't find anything in the DSER that even mentioned it.

MR. SAWYER: Other than station blackout, I don't know that there is any specific regulatory requirement for a length of time for RCIC operation.

MR. MICHELSON: But having specified 30 minutes, I would expect the staff to say yes, that looks -- I would expect them to comment if they thought that was not adequate. They didn't, so I assume they think it's adequate. Is that a correct assumption on my part?

What concerned me -- while they're deciding -- what concerned me was environmental control and so forth. I didn't know if you had run a bunch of calculations that showed you got to a pinch point at 30 minutes.

MR. SAWYER: No. That's not the problem.

MR. MICHELSON: I couldn't tell from reading the document what your problem might be.

MR. SAWYER: The only thing that causes a 30 minute limit is if you -- and actually we have -- that's a minimum requirement. Actually, the actual battery capacity, even assuming no operator intervention to shed unnecessary loads, it's probably good for a couple of hours.

MR. MICHELSON: Well, the SSAR did say that battery capacity is good for much longer, but that didn't tell me why you decided 30 minutes as your design basis for the system. Why not two hours?

I think it should be two hours to make sure that pecple do not put pinch points in later.

MR. SAWYER: I don't have any comment on it.

MR. MICHELSON: Did the staff indicate whether they had looked at that 30 minute number? We did discuss this at Subcommittee and I just wonder if you have looked at it since.

MR. THOMAS: My name is George Thomas. This 30 minutes is for station blackout type of conditions where the ECCS function is concerned. Eight hours can be operated on RCIC.

MR. MICHELSON: It doesn't say that in the SSAR. It just tells me the design basis for the system is 30 minutes and I didn't really know for sure what that even really meant, other than I assumed it would work for 30 minutes when needed for whatever reason. It did state, by the way, AC power loss for that event.

MR. THOMAS: We did ask a question on that one to GE --

MR. MICHELSON: Are you going to ask that the SSAR be amended, then, to indicate the eight hours?

MR. THOMAS: I believe they said in the questions and answers that it's already been amended.

MR. MICHELSON: I'm sorry. The SSAR I have only says 30 minutes. Amendment 7 says 30 minutes.

MR. SCALETTI: We'll rectify that.

MR. MICHELSON: You will fix it. Thank you.

MR. CARROLL: How do you cool a low boil on this turbine?

MR. SAWYER: You've gone about one question deeper than I can handle. I don't know the answer to that.

MR. DILLMANN: The low boil on this turbine is cooled by air. The HPCI turbine, which was larger, had the water heat exchanger on it, but the RCIC turbines do not.

MR. CARROLL: So there are no services needed for this system, other than DC power to operate valves.

MR. SAWYER: That's correct.

MR. DILLMANN: The DC power operates the valves and the turbine control.

MR. SAWYER: And it runs the turbine governor, too.

MR. MICHELSON: So room heatup may be the final limit on the thing.

MR. SAWYER: Yes, that's correct. Room heatup and control room -- not only the RCIC room, but also the control room under station blackout conditions.

MR. MICHELSON: You made me suspicious when you put the 30 minutes on it. Like that's the heatup pinch point.

MR. SAWYER: That doesn't turn out to be the case. I think there has been some misunderstanding and we can correct that. I think you're right. We probably should change the text of the SSAR itself and make it clear that the eight hours is a station blackout evaluation and so forth.

MR. MICHELSON: Whatever it is. That's fine.

[Slide.]

MR. SAWYER: RHR system. It has five basic modes. One mode we're going to talk about again in Section 6.3, which is below pressure core flooder mode, and as part of ECCS, it's supposed to maintain fuel cladding temperature limits, help maintain the suppression pool temperature under its design basis limit of 207.

For heat removal function, it's supposed to achieve this under N minus 1 conditions and with loss of off-site power.

The way it works. We have automatic pump start for high dry well pressure or --

MR. CATTON: How much energy does it take to heat the pool to 207 degrees?

MR. SAWYER: It takes about -- it takes LOCA blowdown plus about ten hours worth of decay heat. I don't know what that is in hours or btus.

MR. CATTON: Or fraction of full power hours.

MR. SAWYER: I can get you that number.

MR. KERR: What's the suppression pool capacity?

MR. SAWYER: The suppression pool is about a million gallons of water.

MR. KERR: You can calculate it then.

MR. SAWYER: You can calculate it. I've got the number in my files back home. It's pretty easy to recover that for you.

MR. CATTON: What is its normal operating temperature?

MR. SAWYER: Typically, we assume for accident evaluations that the starting temperature of the suppression pool is 95 degrees.

MR. CATTON: Okay.

MR. KERR: Now you want to know what a btu is?

[Laughter.]

MR. SAWYER: When it first starts up, it runs at minimum flow because injection is not permitted until the logic which controls the injection valve determines that the reactor pressure is sufficiently low. That's part of our interfacing system LOCA protection for this line. This is one of the lines that has a high low pressure interface and as part of the interface LOCA protection, these valves are protected by two out of four logic.

You get automatic flooder injection when the reactor reaches the shutoff head of the pumps. You get about 4200 GPM per loop at about 40 pounds. This compares with almost twice that for BWR-6. And when I talk about the ECCS Section 6.3, I'll explain why we can get away with so much less water at that time.

We've designed the system so that operator action is not required before 30 minutes. Of course, we don't prevent operator action earlier if he deems it appropriate to do so, but we have sized it so that it wouldn't be necessary.

Part of the reason why that's true is because this heat exchanger is in the loop. So whether you have this mode or any of the other modes I'm about to talk about, the heat exchanger function -- the heat removal function part of RHR is always there.

MR. CARROLL: What is the 30 minutes about? What does the operator have to do at 30 minutes?

MR. SAWYER: I don't have a specific answer for that. I don't know that there is anything specific. That would depend -- at that point, you'd be entering EOPs and making determinations of how much ECCS capacity you really needed to mitigate what's going on and whether it would be nice, for example, to switch over more of the systems to heat removal duty and less of them for core cooling and that sort of thing.

I don't think there's a specific area. It would be scenario dependent.

MR. CARROLL: Has the staff looked at that question?

MR. SCALETTI: I agree. I think most of this would be governed by the emergency operating procedures, whether you needed operator action, and the rest of it to be defined by the procedures.

MR. MICHELSON: Maybe I missed it, I guess, but when do you have to initiate containment spray?

MR. SAWYER: In theory, we don't ever have to initiate containment spray.

MR. MICHELSON: As I recollect, isn't that off the RHR system?

MR. SAWYER: Yes. It's one of the auxiliary functions I'm going to talk about in a couple slides.

MR. MICHELSON: So the 30 minutes, you're saying, well, the operator never needs to take action for containment spray, so something else pinches you first at 30 minutes.

MR. SAWYER: You could envision some scenarios, not all of which would be in the design basis, that after 30 minutes you might want to do some alternate things with the RHR system.

MR. MICHELSON: If you envision the need or the desire for containment spray, when do you think that desire appears in time?

MR. SAWYER: I would say it's well beyond 30 minutes.

MR. MICHELSON: It is still beyond -- before you even think you would need to spray. I didn't get that.

MR. SAWYER: We're not approaching any -- at the short end time, we're not approaching -- you've gone through your first peak on the containment pressure in the short term during the initial blowdown. Then the second peak doesn't occur for many hours, the one that's controlled by the peak suppression pool temperature.

MR. MICHELSON: It's just there for sort of a nicety then?

MR. SAWYER: We've built in -- yes. Well, not just nicety. In severe accident thinking, we've worried about multiple failures. For example, turning on the dry well sprays, for example, even though not required, is certainly a sufficient way to rapidly get the containment pressure down after you've determined that everything else is under control.

MR. MICHELSON: So most of your need, you think, is for unforeseen events, is that what you're saying?

MR. SAWYER: Yes. Performance in this mode is the feedwater line break happens to be the limiting break for containment performance and we've done evaluations that the staff has reviewed that show that, in fact, the suppression pool temperature is limited to 207.

The worst break as far as core cooling is concerned is one of high pressure core flooder lines, and as part of the ECCS network, this helps protect our no core uncovery objective.

MR. MICHELSON: Let me ask you on the temperature limit. My vague recollection is the old types of tests that were done on the suppression process used to indicate problems generating after you got up to 180-190 degrees fahrenheit. Have those kind of gone away up to 207 now?

MR. SAWYER: There are two limits. One is a vent limit, which is the 180 or 190. It's a chugging limit basically. The other one is a quencher limit. Now, the situation in these reactors are that it's the -- on the longer term, what you're controlled by is the quencher limit and that's why the 207 is --

MR. MICHELSON: Because you don't think you have a chugging problem?

MR. SAWYER: Basically, the quenchers have less submergence than the vents. So any steam generated is going to go that pathway to the pool.

MR. MICHELSON: I guess the resolution of this question in our minds will come later when we look at the details of who they've done the thermal hydraulics of quenching.

MR. SAWYER: Right.

MR. CATTON: I think so. As near I could tell from the meeting we had a week or so ago, we're going to revisit all the details.

MR. SAWYER: We're preparing to have another session with the Thermal Hydraulics Subcommittee on the containment thermal hydraulics.

MR. CATTON: I think that, at that time, Carl, we'll get into all the gory details.

MR. MICHELSON: Right. I just wanted to make sure it wasn't going to pass without notice.

MR. SAWYER: I don't know that we've scheduled it yet, but we'd like to get some -- while we're on that subject, we'd like to get some indication when you'd like to have that meeting.

MR. CATTON: I'll have to talk to Paul and let him try to arrange that. When would you be ready?

MR. SAWYER: I wouldn't say in a couple weeks, but we'd certainly be ready in a reasonable period of time, like anytime after maybe a month from now.

MR. MICHELSON: I think, in a practical sense, January is what we had in mind.

MR. CATTON: I think the soonest would be February.

MR. MICHELSON: Or February.

MR. SAWYER: We certainly can be ready for that.

MR. MICHELSON: Okay.

[Slide.]

MR. SAWYER: The next mode is the shut down cooling mode, which is the normal way the RHR system is going to be run 99.9 percent of the time. There are two requirements that we have to meet. One is the Reg Guide 11.39 requirement for, on an emergency basis, to get the reactor vessel to normal boiling point within 36 hours on an N minus 1 condition.

The way we do that, of course, is the reactor is depressurized to approximately 135 pounds and then we establish the shutdown cooling mode of the RHR. The flow path is the reactor suction through the pump, through the heat exchanger, and returned. This is a manually initiated mode and our studies have shown that we actually achieve this in much less than 36 hours, more like 12 hours.

MR. MICHELSON: Let me ask you. You talk about N minus 1 there in that second bullet, which is one looped failed. One of which loops failed? You mean one RHR?

MR. SAWYER: Yes. In other words, from a heat removal standpoint, we're supposed to only take credit for two out of the three heat exchanger loops working for meeting this criterion.

MR. MICHELSON: Now, you're saying, then, to meet that criteria, you need two out of three of the loops operating.

MR. SAWYER: Yes. With one out of the -- I did a study about three years ago and with one out of the three, we can almost claim 36 hours, but not quite. So we don't try to claim that.

The way it's normally used, of course, is that all three loops are available and our requirement is basically -- really, it's not really a regulatory requirement. It basically becomes an availability requirement for meeting your goals of having a minimum refueling outage.

We want to be able to cool the reactor vessel to 140 degrees within 24 hours. All three loops are assumed operational. We run the system exactly as before. There is one more loop available now, so this is -- this actually is a tougher objective to meet than that one because the Delta T begins to get small at the end of the cooldown cycle.

Our studies have shown that we can certainly meet that or beat that requirement, except under exceptional heat sink conditions.

MR. CARROLL: 140 degrees, is that the highest metal temperature on the vessel?

MR. SAWYER: Not necessarily.

MR. CARROLL: Have you done anything special to get rid of the age-old problem of how do you get the head cooled?

MR. SAWYER: Yes. That's part of the reactor water cleanup discussion. In this RHR, the function we used to have of head spray has been taken off to simplify the RHR system and it's been added as one of the functions of the reactor water cleanup system does.

MR. CATTON: How does it cool the head?

MR. SAWYER: We're getting ahead, but as long as we're on the subject. The reactor water cleanup system has the capability, through valving, to pump water into a spray nozzle located in the top head.

MR. CATTON: Okay.

[Slide.]

MR. SAWYER: Suppression pool cooling mode. Basically, it's supposed to cool the suppression pool after the reactor is depressurized or cool it periodically due to potential leaking SRVs. This is also a manually initiated function.

The flow path is slightly different because instead of reactor suction and return, it's now suppression pool suction and return. You have up to three loops available and two out of three is sufficient to be able to perform under the worst condition we can imagine for the suppression pool cooling function.

[Slide.]

MR. SAWYER: Containment cooling mode, which are the dry well spray or the wet well spray. The dry well spray provides team condensation after LOCA as a backup function. This, as I mentioned earlier, can help ease containment thermal environments and get the containment pressure down faster.

Two of the three loops have containment spray, dry well spray capability. There is a common spray header that is fed by either of those two loops. This is also manually initiated. It is interlocked so you have to have high dry well pressure before this function can be run. Approximately the capability with the extra head loss is about 88 percent of that 4200 GPM that I talked about.

When you turn it on, of course, it reduces the long term dry well temperature relatively rapidly and efficiently. Dry well spray function is to condense steam from dry well to wet well based on an assumption of bypass leakage. Once again, are two of the three RHR loops have this function. There is a common spray header; again, manually initiated.

This has lower capacity by design than the flow to the dry well spray and it keeps the wet well pressure below the design value, including our design basis wet well-dry well bypass leakage, which is .05 square feet in this design.

[Slide.]

MR. SAWYER: Finally, it can assist the fuel pool in a case where the fuel pool is overloaded because you take a large batch out or you decide that you need to remove 100 percent of the core for some inspection. So we have provided the capability of the RHR system to plug into the fuel pool cooling network and help remove the decay heat under those circumstances.

It overwhelms the fuel pool cooling system in terms of capability, because, in effect, when you do this, we end up with two 200 percent loops, either one of which can remove actually about twice the amount of energy that this amount of fuel in the fuel pool would generate.

This flow path, basically, goes through the fuel pool distribution sparger and then returns to the RHR system.

[Slide.]

MR. SAWYER: I didn't show the RCIC diagram because it was a relatively simple system. This diagram, basically, is a way you can trace everything I said in the previous discussion of all the modes of operation showing the valving arrangements that permit the various modes.

Mr. Michelson, you have a question?

MR. MICHELSON: Yes. Could you tell me where in the SSAR I can read about the materials of construction of the RHR loop itself? In particular, for instance, what the heat exchanger tubing material is?

MR. SAWYER: Chuck, isn't that in Section 5 on reactor materials or not?

MR. MICHELSON: It's a part of the reactor materials table?

MR. DILLMANN: It's actually engineered safeguards materials.

MR. SAWYER: So it's under Section 6.3

MR. MICHELSON: In 6.3 then. Okay. That describes materials for all these external loops, like reactor water cleanup?

MR. DILLMANN: It's for the engineered safeguards.

MR. MICHELSON: Only those. Okay.

MR. SAWYER: The RHR would be under engineered safeguards. Cleanup water would be under --

MR. MICHELSON: Can you tell me, just offhand, what the material is for the tube of the heat exchanger?

MR. DILLMANN: It's stainless steel.

MR. MICHELSON: You're going to use stainless steel.

[Slide.]

MR. SAWYER: Next is reactor water cleanup system. Its function is to --

MR. REMICK: Before we proceed with that, I think it's a convenient place to break for lunch. Let's recess for lunch until 1:00 p.m. and continue the GE APWR discussion, but first, Paul?

MR. SHEWMON: Let me ask one other question which I, again, suspect is too detailed. On these buttered welds, there have been some examples of cracks starting in the Inconel butter and going on into the pressure vessel steel and nobody has been very happy with it.

There has also been cracking problems in steam generator tubing, which is, again, basically Inconel 600 and, there, they've gone to a higher chrome Inconel 690 in the new vessels, which Westinghouse hopes will now eliminate that class of problems and the lab tests indicate that it will.

My question is whether there has been any work on a higher chrome Inconel butter that could be used there which would have better stress corrosion cracking resistance?

MR. DILLMANN: No, we haven't. Our approach to the stress corrosion cracking of the Inconel, which has been primarily a creviced problem, is the stabilized grade material where we control the carbon and niobium ratio.

MR. SHEWMON: This is a stabilized Inconel?

MR. DILLMANN: A stabilized Inconel. The material we presented last year covered that. I have it with me if you'd like to have a copy of it again.

MR. SHEWMON: And this is niobium or --

MR. DILLMANN: You control the niobium carbon ratio, but basically it's a niobium stabilization.

MR. SHEWMON: Thank you.

MR. MICHELSON: Before we break for lunch, I'd like to point out to GE that we have until 2:00 to finish up. We'll have to keep moving, but I think we do need to hear you out on all this material and that we have time to answer the questions, but there are also other things that have to be brought up yet today.

So 2:00 is still going to have to be a fairly firm target.

MR. SAWYER: Both the Full Committee and the Subcommittee have seen the ECCS network several times, so it may turn out that we can go through that pretty briefly.

MR. MICHELSON: I think we're more interested now in specific things that members have. So let's see if we can try to finish it up at 2:00.

MR. REMICK: Let's recess for lunch.

[Whereupon, at 12:10 p.m., the Committee was recessed for lunch, to reconvene this same day at 1:00 p.m.]

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

MR. REMICK: Let's continue our discussion of the GE ABWR.

[Slide]

MR. SAWYER: Now we're on to the reactor water cleanup system, which the system function is to maintain the reactor water quality within the specified limits, and you saw what those limits were from Chuck earlier today, while minimizing the heat losses.

We discharge excess water during startup and shutdown. The hot standby conditions, also, is another function. And a third function is that it provides head spray if you want to have faster cool down.

[Slide]

MR. SAWYER: A quick view of the P&ID shows pick off point here for the suction at about the mid plane of the vessel. Also, a pick up point on the bottom head. Through a single regenerative heat exchanger; a pair of nonregenerative heat exchangers.

Each pump is rated for one percent -- filter demineralizers. It then returns either to the reactor, split to both feedwater lines here or it can provide the head spray function here.

In addition, this is the blowdown line to rad waste or excess liquid during heatup.

[Slide]

MR. SAWYER: It is rated at two percent of reactor feedwater flow which is more than our recent BWRs have had. The pumps are in the cold leg downstream of the nonregenerative heat exchangers. It's a seal-less motor design which is more reliable, as you are probably aware. We've had problems with some of our reactor water cleanup systems and this is our design approach to solve those problems.

The cold leg produces lower radiation should maintenance be required. Return flow is by feedwater. As I said, we have a one by two percent regen; two by one percent nonregen; two by one percent seal-less pump, so each pump is one percent, two by one percent filter demineralizers.

The backwash equipment for the filter demineralizers is shared with the fuel pool cooling and cleanup system and it's a fully automated system.

MR. REMICK: Are those demineralizers regenerable or are they disposable?

MR. SAWYER: Disposable.

MR. REMICK: Disposable.

MR. MICHELSON: In view of the fact that it's a larger pool than previous designs, could you tell me what the -- in the line going to the reactor water cleanup what pipe size is now involved through the isolation valves?

MR. SAWYER: It shows on the P&ID. My memory is, it's eight-inches.

MR. MICHELSON: I think that's one of those P&IDs that I couldn't possibly read and you were going to supply us one that was readable.

What size do you think it is?

MR. SAWYER: Eight-inches.

MR. MICHELSON: Eight-inches, okay. Through the isolation valves?

MR. SAWYER: Yes.

MR. MICHELSON: How about up to the regenerative heat exchangers, is it still eight-inches?

MR. SAWYER: I think it's an eight-inch line, basically all the way through the system.

MR. MICHELSON: It's a big one.

Now, as I understand it, you're only seismically qualifying through the second isolation valve; is that correct?

MR. SAWYER: That's correct.

MR. MICHELSON: And beyond that it will be what kind of piping code?

MR. SAWYER: Let me get the drawing back up.

MR. MICHELSON: Are you going to make that a Class 3 pipe beyond that or do you know?

[Slide]

MR. SAWYER: Through the isolation valves it's quality piping and beyond the isolation valves it's -- we call it safety -- what do we call that, Chuck?

We want to call that ASME Code Class D, don't we.

MR. MICHELSON: "D" like in dog?

MR. DILLMANN: Yes.

MR. MICHELSON: Thank you.

MR. REMICK: Is it a full pressure system?

MR. SAWYER: Full pressure system for the whole loop.

MR. MICHELSON: Now, eventually somewhere but not here I would expect to see the discussion of the capability of isolating that eight-inch break in the unlikely event the pipe should rupture.

MR. SAWYER: Right. Yes.

The Staff has asked the question on all of the breaks outside of containment and we're in the process of preparing our response and it should be in to the Staff within about a month.

MR. MICHELSON: And that response will include the capability of the cleanup valves?

MR. SAWYER: Which includes the cleanup system as well as all high energy pipe load with potential for break outside containment.

MR. MICHELSON: But it would include the capability of the valving to isolate such breaks?

MR. SAWYER: That's correct.

The design basis for the isolation valves for this type of break is a 30-second closure. That's pretty standard practice that we have used for a number of years.

MR. MICHELSON: And there will be an analysis of an eight-inch line break for 30-seconds.

MR. SAWYER: Correct. Which shows -- basically shows the subcompartment, the pressurization, the temperature effects, and so forth. I already know the results of that, but the Staff hasn't seen it.

MR. MICHELSON: Okay.

MR. SAWYER: Basically the bottom line of those results is going to be, it's not a problem.

MR. MICHELSON: Now, it's going to be -- is it going to become a part of the SSAR or will it be some kind of a topical or do you know?

MR. SAWYER: I guess we haven't figured that out yet, but there is no problem to make that part of the SSAR. I think the subcompartment analysis is supposed to be in the SSAR.

MR. SCALETTI: That's correct.

MR. MICHELSON: We'll treat it later on. Thank you.

MR. SAWYER: If there aren't any other questions I'll turn the floor back over to Chuck.

MR. MICHELSON: There is a general question on the RCIC end that might apply to the reactor water cleanup. The general question is, as I look at the P&IDs which are in the SSAR and I keep coming across terms like the designer will decide what the pipe size is and so forth. In other words, I can't read the P&ID and necessarily even know what the pipe size is.

MR. SAWYER: We have agreed to provide replacement P&IDs for the ones that you have -- that are more descriptive.

MR. MICHELSON: Was that the idea, that you will amend the SSAR later to pick up on it?

MR. SAWYER: Right.

MR. MICHELSON: Thank you.

[Slide]

MR. DILLMANN: We have a couple brief charts on materials for engineered safety features. Metallic materials, basically, in the engineered safety features we use the same materials and the same material controls that we have discussed earlier for things like internals and reactor pressure boundary. Processing controls are the same.

The fluid that these components are exposed to, for the most part, is pure deionized or demineralized water; therefore, there should be no additional material requirements.

The one exception is that the boron injection system, if it's ever used, injects through the HPCF sparger, so in that event the HPCF sparger would also be exposed to the borated water. However, the system is stainless steel and should be compatible with that.

So in the engineered safety features, our safety systems we still use the same material in processing and controls that we do in the reactor pressure boundary and internals.

The organic materials -- there's a typo on this chart. The organic coatings, we minimize their application. The most prevalent application is in the containment liner and not containment lines.

Also, the carbon steel structures in the containment and some equipment in the containment.

The coating used here is an epoxy coating. It's qualified to ANSI standards for the LOCA environment. It also meets Reg Guide 1.54.

There are some exceptions to this and those exceptions are small items like valve handles, some electronic equipment, name plates and covers. But the volume of that material is so small it should have a negligible impact.

Emphasis is placed on not having coatings that could shed and plug essential services. And, of course, those essential services are equipped with strainers to further avoid that plugging.

Other organic materials in the engineered safety features are required for their function like insulation and so forth are materials consistent with the expected environment and those materials are qualified for the environmental conditions that they will see both normally and abnormally.

MR. CATTON: The insulating materials, do you test them under flow conditions?

MR. DILLMANN: The insulating materials I was talking about is more like the electrical.

MR. CATTON: I was thinking of maybe for heated pipe or something.

MR. DILLMANN: The insulation that's applied to like piping and so forth, if it could be exposed to a steam or water spray is protected so it can't be sloughed off and distributed around the containment.

MR. CATTON: Well, if you have a steam line break somewhere it doesn't have to necessarily be in the direct flow; it can be in the adjacent room if there are doorways.

Do you look at that sort of thing?

MR. DILLMANN: To tell you the truth, I don't know the answer to that.

Craig, do you know anything about that?

MR. SAWYER: I am sorry, I don't understand this line of discussion about the insulation.

MR. CATTON: I can't hear you.

MR. SAWYER: I'm sorry, I don't understand your -- actually understand the meaning of your question about the insulation and the steam break.

MR. CATTON: Well, I'm just wondering if it can be blown off, and it doesn't have to be directly impinged with the jet to be blown off.

MR. DILLMANN: I guess I misunderstood your question. I wouldn't expect it to be blown off. I thought your question was, if it was exposed to long-term steam environment to create fall off.

MR. CATTON: No, no. Well, if you have a large enough break and you start filling up a room with steam, the flow away from the break, not necessarily in the jet is sufficient to do a lot of damage.

And I was just wondering if you had done anything to look into this.

MR. DILLMANN: I'm not aware of any evaluation of that sort of phenomena.

MR. CATTON: Well, there's a lot of evaluation in the test facility in Germany, the HDR containment.

MR. DILLMANN: What I said is, I am unaware of us having done anything. We'll take a look at it.

MR. CATTON: Does the Staff require anything like this?

MR. PARCZEWSKI: My name is Chris Parczewski from Material and Chemical Engineering branch.

Yes, the Staff do require the environmental testing of the insulation.

MR. CATTON: Well, that statement doesn't say anything, because normally environmental testing just means temperature and humidity and pressure.

Do you do anything more than that? Anything more than jet impingement?

MR. PARCZEWSKI: No, we don't provide it.

MR. CATTON: Are you familiar with the work that went on at the HDR facility in Germany?

MR. PARCZEWSKI: No, I'm not familiar with the work.

MR. CATTON: Because the jet impingement is the least of your worries.

MR. PARCZEWSKI: Actually, the person who does -- the person who is responsible for this particular activity in NRC is on vacation, so I'm not able to answer the question. I can provide you later the answer.

MR. CATTON: I would like you to look into it for me.

I think that this aspect is just taken care of inadequately. And it may be because the rules that guide you on how to do it are inadequate. But nevertheless, I think it's inadequate.

The flows resulting from some kind of a break do probably more damage than the direct impingement does. They can shred things; they blow things around. There are cases where they have literally lifted a concrete block and hurled it across a room, and this was not the jet it was just the flow.

MR. PARCZEWSKI: All right, we are going to look into it then.

MR. CATTON: HDR facility and you might talk to Roy Woods, he was there a couple of weeks ago when I was.

MR. MICHELSON: I might suggest, Iva, if you wish, you can prepare a question on this for our letter.

MR. CATTON: Okay.

MR. MICHELSON: Because the best way of assuring that you get the answers you want is to pose a comment which is normally replied to in writing.

MR. CATTON: Good. I'll do that.

MR. DILLMANN: Now, we are ready for our final --

MR. MICHELSON: Wait a minute, before you get to the ECCS. There is a Section 626 which appears in the DSAR which deals with containment leakage testing, and were you planning on discussing it at all? Because the DSAR contains a complete discussion. It was, I thought, ready for any comments or whatever.

MR. SAWYER: It wasn't on the agenda or discussed by either us or the Staff, so we were not prepared to discuss that.

MR. DILLMANN: Do you have specific questions that you would like us to address?

MR. MICHELSON: I had only one which I was going to ask after a presentation, but I'll just ask it now.

MR. SAWYER: Why don't you ask it and we'll take it under advisement.

MR. MICHELSON: Part of the DSAR discusses -- this is with respect to containment leakage testing -- part of the DSAR discusses inflatable seals and the testing of such seals, but so far nowhere in the SSAR can I find a description of the seals or the arrangement for pressurizing the seals or for assuring pressurization during post-accident periods, which I thought would be an essential description.

And my question is going to be posed as, how reliable is the air supply to keep the seals? How many days do you think you can keep them pumped up in a severe accident situation or whatever? And what is the rate of deterioration of these seals under elevated temperature and pressure conditions?

None of this I could find discussed, but it may be somewhere in the SSAR and I haven't looked for it yet.

MR. DILLMANN: Let us see if we can get you an answer to that or we will get you an answer to that.

MR. MICHELSON: I find the seals discussed in here as a leakage problem, but I don't find in the Staff evaluation any consideration of the reliability of the air supply or the viability of the seals under severe -- more severe temperature pressure, post-accident conditions.

MR. SAWYER: I think our plan had been to discuss that as part of the overall -- to tie that in with the containment thermal hydraulics.

MR. MICHELSON: Yes, it could be. Yes, that would be a good place to discuss it. But it certainly ought to appear somewhere in the SSAR. Maybe we haven't got the sections yet where it would be expected. I thought it was going to be in here, though.

MR. DILLMANN: That brings us to ECCS and QA which is the last two topics we had on the formal agenda, but we would be happy to try to address any others.

[Slide]

MR. SAWYER: Let me summarize the features of the ECCS network for you. We have three completely separate mechanical and electrical divisions in the ABWR for the most important functions which are the core cooling function; suppression pool cooling function; and shutdown cooling function.

I described the RHR. As I pointed out to you, there are some other functions that are two-divisional, but nothing to do with ECCS.

We have automated post-LOCA cooling which is a big improvement from past designs, so that the heat exchangers are always in the loop which means the operator doesn't have to decide whether he would rather have heat removal or core cooling. He always gets heat removal, no matter whether he's in core cooling or suppression pool cooling mode.

We have eliminated or transferred several complex modes. Steam condensing is out; RPV head spray is now with the reacto. water cleanup system; containment flood is now done by one of the severe accident systems which we have proposed which is the fire pump system. And that has reduced the number of valves and pipes particularly in the RHR system by about a third.

We have a significant capacity reduction, which I will explain to you in the next slide. We have greatly reduced the duty dur' g transients. For example, we now have N minus 2 capability at high pressure exclusive of feedwater. We've improved a small break response. Reduced the need for ADS. In effect, 1-HPCF, 1 high pressure core flutter is capable of handling the complete break spectrum and can meet Appendix K.

Within the design basis we have no fuel uncovery for any pipe break.

MR. MICHELSON: Relative to that slide there is something that came up during the subcommittee meeting that I think the full committee should be aware of and that is, we pursued with General Electric the question of how these three separate trains of equipment were -- how the room was environmentally cooled and so forth. And in the process of the inquiry we found out that there is a common nonessential heating and ventilating system that serves all three of these areas and ties all three of them together.

Now, this can be done, but I think at the meeting we cautioned you it's a feature which we would like to hear much more about and would expect it to be carefully documented in the SSAR and evaluated by the Staff, because this is the sort of thing we have a lot of concern about because of the potential environmental coupling in more than one area.

MR. SAWYER: We agree with you. That's going to be covered in -- I think it's going to be covered some in the fire protection review that you're going to go through. And also, in the subcompartment analysis which is going to be part of containment. But we understand your comment and we agree with it.

[Slide]

MR. SAWYER: Just for comparison, this shows the comparison of typical plants that are operating today compared with the ABWR. And the major reason for the large capacity reduction, particularly in the low pressure systems, is I don't need to have a large reflood system since I don't have these large pipes attached to the vessel below the top of the core anymore.

So the flow capacity of the low pressure systems can now be matched a lot more closely to the normal shutdown cooling or other heat removal requirements than to a reflood requirement.

Questions have come up in the past about N minus 2 capability and my answer is that, when you include the complement of pumps and the diesels that go with those pumps there's about 500 combinations approximately of double failures you can consider, if you want to just be mechanistic about double failures; and only one of those is uncovered and that's the one which basically is, break of that pipe diesel out-diesel out. That one is the only one of those combinations that can't be covered. Other than that we think we have N minus 2 covered.

[Slide]

MR. SAWYER: Just a quick run through. HPCF is a backup to RCIC for normal duty in addition to being one of the high pressure systems that comes on at a lower level, so that if the RCIC works for isolation HPCF should never be called on. It normally sucks from the condensation storage tank. However, it will be automatically transferred to suck from suppression pool on transfer command.

The RCIC we've already talked about.

[Slide]

MR. SAWYER: Low pressure systems, basically, this drawing is just another way of representing the RHR that you saw in my RHR presentation before lunch.

[Slide]

MR. SAWYER: The next couple of charts very briefly run through, first of all, an elevation view of the nozzle arrangements in the ABWR comparing with current plants which basically demonstrate the fact that we don't have any large piping attached to the vessel below the core elevation. The lowest elevation large pipe is basically the suction for the shutdown cooling system.

[Slide]

MR. SAWYER: And in azimuth, the next one shows basically how we have mechanically divided the ECCS into three zones and shows -- it's a little bit hard for you to understand until I tell you that this is in-board of the core shroud. This is the RPV boundary and this is intended to represent the containment boundary. So when you see something like this that says suction this means suction from the suppression pool.

MR. REMICK: On that previous slide could you show me where the -- oh, I'm sorry, I see it, head spray. All right, I see where it is.

[Slide]

MR. SAWYER: With that network our LOCA response is basically --

MR. MICHELSON: Before we leave that other picture could I ask, at the San Francisco meeting I asked the question about why that vessel has a much deeper lower head than --

MR. SAWYER: I verify that it doesn't.

MR. MICHELSON: It does not. This picture is --

MR. SAWYER: That's an artistic license in the picture; that's not real.

MR. MICHELSON: That's quite a bit of license in this case.

MR. SAWYER: It's not real.

MR. MICHELSON: But it isn't real?

MR. SAWYER: No, it's not real. If anything there is a slightly less dimension between the core plate and the lower head in ABWR because the ABWR has a disk lower head.

LOCA response, we show no core uncovery for any pipe break in our analysis. No core heatup for what we call the nominal case.

Now, to make Appendix K compliance for us as simple as possible we have simplified the process of converting the safer application to the ABWR, and as part of that process we conservatively trip all the pumps, all the RIPs at time zero even though that by itself is an accident.

So when we report peak clad temperatures for these various LOCAs what we're really doing is, doing minor variations on all pump trip events as opposed to reporting peak clad temperatures that would occur due to any uncovery.

MR. CATTON: And in support of this you're going to take a look at the RIP and the possibility that you will blow it out?

MR. SAWYER: Not as part of our design basis evaluation.

MR. CATTON: No, but you were going to do that. If the number is really low that supports the two-inch break in the bottom.

MR. SAWYER: Oh, the two-inch break is part of our design basis. We have certainly analyzed that. That's not the limiting break, however.

If there are no questions let me proceed on to Chapter 17, Quality Assurance. Let me preface this remark by saying, I am not a quality assurance expert.

MR. CARROLL: Have you ever met one?

MR. SAWYER: No comment.

But let me at least run through what's in Chapter 17 and what our commitments to the Staff are in meeting our quality assurance requirements.

As you know, we have a project in Japan that we and two Japanese partners are performing what we call a common engineering for those units and the three of us are jointly responsible for that design.

[Slide]

MR. SAWYER: We have a process where each company has to formally review and approve each common engineering document. In other words, whether or not General Electric did the rudimentary work that formed the basis for the document or whether Itituchi or Toshiba did it, all three of us have to review and approve each document. The lead responsibility is assigned to one of the three companies and that includes drafting the document; internally processing it; obtaining review and all that kind of stuff.

Once the document is issued it is put on General Electric's master parts list and then at that point a General Electric engineer and General Electric will take responsibility for making sure that continued changes to that document are within our design change control system.

The change control process works the same way the original design process works which is, all three parties have to approve the change.

[Slide]

MR. SAWYER: We are committed to the QA procedures in the ABWR Organization Procedures Manual. GE, Itituchi and Toshiba meet our respective guidelines, regulatory guidelines for quality assurance. We have each reviewed the adequacy of each others QA program against each others guidelines. And we have, for example, reviewed Itituchi and Toshiba's process and we are convinced that it meets the requirements of Appendix B. I believe the NRC has audited not only us but Itituchi and Toshiba in this regard and has come to the same conclusion.

[Slide]

MR. SAWYER: We are responsible for the content of all the common engineering document. And the way we maintain continued assurance is by formal review and approval in the first place; by annual review of the QA program; and by maintaining a configuration control of over future changes.

MR. MICHELSON: My only comment and I think the committee would like to hear your response. My only comment on Chapter 17 is that this is a three party organization, three equal party organization, each party is doing a portion of the design work, although one party is assuming responsibility for the work it's not doing the work.

My rudimentary understanding of QA is that it's a program that controls the work being done by the people doing the work. And the SSAR does not contain nor even reference the quality control program that's being used by Itituchi or Toshiba.

I can understand that GE can say they're responsible, but I think I have to know what is the program actually being carried out by the people doing the work, and that is not the GE QA program, it's the Itituchi program or the Toshiba and none of this is described in the SSAR.

MR. SAWYER: Yes.

MR. MICHELSON: And the fact is, Itituchi and Toshiba aren't even mentioned. It just talks about some term like technical something or other.

MR. SAWYER: Technical associates.

MR. MICHELSON: Technical associates you call them.

MR. SAWYER: Yes.

MR. MICHELSON: I don't even know who they are. I don't know if it's just those two or whether you're talking about other organizations as well or just what.

MR. SAWYER: I think we can satisfy your concerns in this area by basically outlining in the text that for QA purposes with regard to ongoing design and knowledge of the design and maintenance of the design that we would treat Itituchi and Toshiba just as we would any other subvendor that we treat on our projects.

MR. MICHELSON: That means that your QA plan controls the subvendor or just that you have reviewed the subvendors program?

MR. SAWYER: Reviewed the subvendors program. Of course, a subvendors program is auditable.

MR. MICHELSON: Even that clarification would be important because then I could read your QA plan and know what you're expecting of your subcontractor. But that is not the way it's explained in the SSAR.

MR. SAWYER: You're right, the SSAR right now is silent on that point.

MR. MICHELSON: Yes, it sounds like three equal partners each doing their own thing but GE assuming responsibility.

MR. SAWYER: That's true. I don't think we have totally explained in the text that you have the ongoing maintenance role that GE wants to assume.

MR. CARROLL: This is for the long-term --

MR. SAWYER: That's right.

MR. CARROLL: -- for when you sell a domestic plan, for example.

MR. SAWYER: For example; that's correct.

MR. CARROLL: And you want to have certification that the QA program is acceptable for that.

MR. SAWYER: Right.

MR. CARROLL: And Toshiba and Itituchi aren't involved in that at all.

MR. SAWYER: Not necessarily.

MR. MICHELSON: But unfortunately, maybe their design --

MR. SAWYER: Unless we choose them to provide some of the equipment, yes.

MR. MICHELSON: The problem is, it may be their design that the domestic user is going to use. Nothing says that GE is going to redesign these things for domestic use.

MR. SAWYER: We have no intention of doing so.

MR. MICHELSON: That was my understanding.

MR. SAWYER: We basically intend to adopt the design that the three of us have agreed to.

MR. MICHELSON: Now, if you can assure end product quality by some kind of an auditing process alone, then that's great. But that's not defined in here either as your intention. So I assume that we're going to build a plant using a Toshiba design in part. Part of the work that was done --

MR. SAWYER: Part of the original design work would have been done by Toshiba, for example, that's correct.

MR. MICHELSON: And it may even be made in Japan and brought to this country, whatever. But it's going to be a design that GE did not do themselves.

MR. SAWYER: That we did not originally create; that's correct.

MR. MICHELSON: And it's the creation of design products that I think have to have a quality control program or quality assurance program.

MR. SPRAUL: We have looked at what GE has done in the past, they have gone in and audited and for a design control program it's described in their topical report of how they control the design. And their control of design and their design reviews assure even more of an independent review of the Japanese work than they have assured us of an independent review of their own work. Because internally the question of independence always comes up, but their design verification of the work that is being done in Japan and their auditing of the Japanese work in accordance with their QA program provides, I think, a pretty solid assurance that that design will be an acceptable design.

MR. MICHELSON: Well, I am quite sure you're right.

However, go back -- I invite you to go back and read the SSAR and see if it says what you're telling me now. I couldn't derive that from it.

MR. SPRAUL: We would not have derived it without going out there and see what's going on.

MR. MICHELSON: Well, the document that I have to review and the committee has to review is the SSAR and it should explain what you're going to do. If it doesn't, then it needs to be fixed.

MR. SPRAUL: Agreed.

MR. MICHELSON: Okay.

MR. CARROLL: How does this work under Part 52? Chapter 17 of a SSAR today talks about how the equipment was manufactured, operational phase, QA, all of that stuff. Where does that get into this process?

MR. SPRAUL: I'm sorry, I didn't introduce myself, I'm Jack Spraul of the NRC Staff in the Licensing Performance and Quality Evaluation Branch.

We have reviewed the SAR submittal and looking at it on a basis of certification of the design.

MR. CARROLL: Yes.

MR. SPRAUL: Now, operations is not covered.

MR. CARROLL: When does it get covered?

MR. SPRAUL: When an licensee or when an applicant comes in with an application to use this design and build a plant and operate it.

MR. CARROLL: And that same would be true of manufacturing and construction.

MR. SPRAUL: Yes, that's right.

MR. CARROLL: And startup testing.

MR. SPRAUL: Your typical PSAR, Preliminary Safety Analysis Report, covers quality assurance for construction. And then the FSAR normally covers quality assurance for your operations and they're significantly different programs and, you know, one-step licensing will all have to be described in one SAR.

But GE involvement in the development of the SARs for construction and operation, it will be there but we're not looking at that right now, we're looking at the design certification only.

MR. CARROLL: But ultimately there will be a more complete Chapter 17 at this point --

MR. SPRAUL: Oh, yes. When an applicant comes in and says, we're going to do it; then we will be looking at their program as well.

MR. KERR: One part of this activity that concerns me, it is a conventional wisdom among many of the people to whom I talk that GE is going to get out of the nuclear power business as soon as it gracefully can do so. And I am puzzled that one can expect the appropriate care and attention to this activity from an organization that has sort of committed itself to get out of the activity.

Is the Staff concerned about this or am I getting information that's invalid?

MR. SAWYER: May I comment first.

I would like to know where this conventional wisdom came from. For a company going out of business we're the only company that I know that has two units on order for a large plant; has a significant design activity for a 600 megawatt passive plant; and is the lead agency in the liquid metal program. So I suppose the conventional wisdom comes from some of our competitors, marketing organizations.

MR. KERR: Would the Staff comment and convince me that my information is invalid; I would be happy to be convinced?

MR. SCALETTI: Well, I cannot comment on the validity of your information. I have not heard it. The application is before us and as General Electric's application and we have it under review.

MR. MICHELSON: I think it is fair to say, though, that we have to review this certification process as if GE didn't exist thereafter simply because whoever does pick it up if it were to happen is only going to deal with what has been agreed to in the process of certification. Nothing beyond that, as I see it, is cast and concrete.

MR. SCALETTI: I guess if GE should somehow vanish from the picture the Staff would have to, I guess, determine later on the validity of the certification because General Electric holds all the documentation to the design.

MR. MICHELSON: Well, it may be a spinoff of the General Electric nuclear operations of some other company, for all I know. These things are not uncommon in the $25 billion range nowadays which far exceeds the worth of GE in this area. So it's not incredible that a change could happen.

I just wonder though, how do we judge? I think we have to be very careful to make sure that we have tied down what needs to be tied down in the certification process.

MR. SCALETTI: I guess it's something we would have to look at, at the time this action took place.

MR. MICHELSON: You mean you think that a certified design is only owned by one company then and that if it changes ownership, even though it doesn't change the name that it's a new deal?

MR. SCALETTI: Well, I'm not saying -- whoever purchased the company, whoever acquired the rights to the design had all the design documentation at their disposal to adequately to sell and to support the construction and operation of the design, then probably not. But clearly if General Electric just folded their doors and the design information with it, then clearly the design -- I would say the design certification would be invalid because no one would have the supporting design basis for the facility.

MR. REMICK: Mr. Sawyer, in the context of this mod that we're talking about, what are the major differences, if any -- I'm just talking about the systems and so forth described in what we've just been considering today -- what are the major differences, if any, between the certified design and the TEPCO plants; are there any?

MR. SAWYER: They are very minor. There are a few areas where we've had to make some minor changes to comply with NRC regulations, and they're so minor I can't even remember what they are.

You have to be aware that in the severe accident area we have proposed -- the four features we proposed and TEPCO hasn't decided what, if anything, to do about that in Japan. but we have committed to those in the U.S. and those are probably the most major of the differences that I can think of.

MR. MICHELSON: I thought that tangential turbine was quite a difference.

MR. SAWYER: Oh, that's right.

MR. MICHELSON: It certainly is not minor.

MR. SAWYER: No, that's right. We have a different turbine island.

MR. REMICK: But that wasn't in this module?

MR. SAWYER: That's not in this module.

MR. REMICK: My question was really, just this module.

MR. SAWYER: No, you jogged my memory. You're correct, the turbine island will be different for the U.S. application.

MR. MICHELSON: I think the control room location is a little different than Japan, too.

MR. SCALETTI: The Japanese design is a two-unit design with a shared control room. The U.S. design is a single design with the supporting facilities.

MR. SAWYER: I was thinking mostly of the safety envelope.

MR. MICHELSON: Yes. Now, we did bring up a discussion during the subcommittee which the full committee should be aware of and that is, I asked a question concerning how you're going to document these differences because the detailed design is only going to be done on the Japanese plant, as I understand it.

So how detailed the design is going to be performed in those areas where there are significant differences like the tangential turbine, because that has to change quite a few layout considerations.

And the answer that I -- well, let me ask General Electric if they could give the answer again that they gave the other day, where there are differences how will you document them for Staff review.

MR. SAWYER: In the case of the arrangements I think we are providing the Staff with the U.S. reference plot plan and turbine island arrangement and control room arrangement, for example.

If it's a detailed design difference, what we're doing in San Jose is we're basically reviewing those differences and if we agree that the difference needs to be there, then we are going to basically engineer it down to the point where we understand what its ramifications are, but not actually make document changes at this time.

MR. MICHELSON: Now, at this time means prior to certification?

MR. SAWYER: Prior to certification, correct.

MR. MICHELSON: So that where there are differences between the Japanese design and the U.S. design, their differences will not be detailed to the so-called essentially complete level.

MR. SAWYER: The question really depends upon what the difference is. If the difference is in something which was required for NRC review for certification, yes, then we provide you the new design. If the difference is in a detail that's below the level of what we had anticipated to provide for review, then we're just going to document the difference, keep track of it so that when we get an order we will be prepared to then execute the change.

MR. MICHELSON: In reading the design basis letter that the Staff sent to you people I find there are a number of areas in which they expect reasonably detailed information. I think major cable tray layout, that sort of thing was one of those areas.

I assume, of course, you could readily supply that to the Staff where the Japanese plants the same. But where the Japanese plants are different are you going to still supply the level of information defined in the licensing basis letter?

MR. SAWYER: Yes.

MR. MICHELSON: So you will detail on locating the cable trays and so forth?

MR. SAWYER: Yes, our commitment is to supply what we agreed to supply in the licensing basis.

MR. MICHELSON: So I can use that basis letter as a feeling --

MR. SAWYER: That's right.

MR. MICHELSON: -- even where there are differences that will be detailed to that level.

MR. SAWYER: That's right.

MR. MICHELSON: Thank you.

MR. REMICK: Is that it for GE?

MR. MICHELSON: I think that's it for GE unless there are questions from the committee.

MR. REMICK: Anything further from the Staff?

MR. SCALETTI: No further comments.

MR. REMICK: Chet?

MR. SIESS: Are we supposed to write a letter?

MR. REMICK: Yes. We plan to and that's the next topic.

MR. SIESS: This is equivalent of what in terms of the old construction --

MR. MICHELSON: Nothing we've ever done before I guess.

MR. CARROLL: We're breaking new ground, Chet.

MR. REMICK: Incidentally, I believe the recommendation of the subcommittee is that it be a letter to the Staff at this point. We're not ready to write a letter on the module.

MR. SIESS: A letter to the Staff.

MR. MICHELSON: To the EDO.

MR. SIESS: Have we gotten formally any Staff safety evaluation report other than the thing I found somewhere?

MR. MICHELSON: That's it.

MR. SIESS: That's it.

MR. MICHELSON: So far. I mean, you will get more of that as more material comes in.

MR. SIESS: At some point in our lives are we going to get something equivalent to a Staff safety evaluation report of a license application.

MR. MICHELSON: Yes, what you would call, I think, the final draft, yes.

MR. WARD: This is a piece of it.

MR. MICHELSON: It's a piece of it.

MR. SIESS: Yes, but we can't write a letter in pieces.

MR. REMICK: That's why it's a letter to the Staff just giving our advice --

MR. SIESS: This is a letter advising the Staff on concerns.

MR. MICHELSON: Yes.

MR. WARD: Well, I think Carl is proposing that we write a letter.

MR. MICHELSON: To the Staff; not to the Commission.

MR. SIESS: At some point we're going to write a letter to the Commission to satisfy the requirements of law.

MR. MICHELSON: Yes.

MR. SIESS: That would be one letter.

MR. REMICK: And that first letter is predicted to be at the end of 1990?

MR. MICHELSON: That letter on the final safety evaluation report is scheduled by the end of 1990 assuming everything is on schedule.

MR. SIESS: This is the procedure: there will be at that point in the process an ACRS letter required just like there is one required for CPs and OLs, but this will be the equivalent -- it's not a CP or an OL.

MR. MICHELSON: This will not be the letter you have in mind yet.

MR. SIESS: But I mean, when we write that letter it is still somewhere in between a CP and an OL.

MR. MICHELSON: Yes, it's an FDA letter.

And I would envision that we write letters on the modules as they're proposed with draft safety evaluation reports, and you have module 1 now.

MR. SIESS: These are letters to the Staff.

MR. MICHELSON: These will be letters only to the Staff. There would be at least four modules plus a wrap-up module.

MR. SIESS: Basically things that we have no problem with in the Staff's draft here. If that same material appears in their final draft we won't reopen it.

MR. MICHELSON: Well, I don't know that we have committed because the problem is that even on individual sections there are open items on the section, and it gets awfully complicated to say which part we cast in concrete and which part we don't. Therefore, I would recommend that we not cast any of them in concrete. They're all preliminary comments until the end of 1990.

I certainly think the Staff would address them as they see fit, and if we like the addressing nothing more needs to be said. If we don't, then we would remind them again at that time.

There's quite a bit of discussion that has got to go with it. I can't read it in 10 minutes.

MR. REMICK: We will decide that. But before I want to thank both the GE people and the Staff for a very concise presentation of the information. I think on both sides you have things that we've asked that you need to supply us with. I trust that you made notes of those or will find it in the transcript.

I also want to compliment our subcommittee chairman for doing a very thorough job and I'm sure helped by the subcommittee meeting. But I want to thank you all.

[Whereupon, at 1:50 p.m. the meeting was adjourned to reconvene at the call of the chair.]

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: ACRS 355th General Meeting

DOCKET NUMBER:

PLACE OF PROCEEDING: Bethesda, Maryland

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.

Marilynn Nations

MARILYNN NATIONS
Official Reporter
Ann Riley & Associates, Ltd.

ACRS COMMITTEE

REVIEW

OF

GE ADVANCED BOILING WATER

REACTOR DESIGN

NOVEMBER 17, 1989

GE NUCLEAR ENERGY

ACRS REVIEW OF
GE ABWR

ITEMS TO BE COVERED

4.5 REACTOR MATERIALS

4.6 FUNCTIONAL DESIGN OF FINE MOTION CONTROL ROD DRIVE SYSTEM

5.2.1 COMPLIANCE WITH 10 CFR 50 55A

5.2.2 OVERPRESSURE PROTECTION

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.5 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION

5.3.1 REACTOR VESSEL MATERIALS

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.3 REACTOR VESSEL INTEGRITY

5.4.1 REACTOR RECIRCULATION SYSTEM

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

11/17/89 CWD

ACRS REVIEW OF
GE ABWR

ITEMS TO BE COVERED (CONT'D)

5.4.8 REACTOR WATER CLEANUP SYSTEM

6.1.1 METALLIC MATERIALS

6.1.2 PROTECTIVE COATING SYSTEMS (PAINTS) - ORGANIC MATERIAL

6.3 EMERGENCY CORE COOLING SYSTEM

CHAP 17 QUALITY ASSURANCE

ACRS REVIEW OF
GE ABWR

REACTOR MATERIALS

GENERAL

MATERIALS COMPLY WITH APPLICABLE CODES, REGULATIONS AND GUIDES

MATERIALS IN CONTACT WITH COOLANT INCORPORATE ALL THE EXPERIENCE AND DEVELOPMENT HISTORY TO ASSURE COMPATIBILITY WITH BWR WATER CHEMISTRY

MATERIALS LIMITED TO THOSE DEMONSTRATED BY SUCCESSFUL EXPERIENCE

PRESSURE VESSEL STEEL INCLUDES LOW INITIAL NDT COMBINED WITH CONTROL OF CONSTITUENTS THAT AFFECT RADIATION EMBRITTLEMENT

PROCESS CONTROLS EMPLOYED TO ASSURE THAT MATERIAL PROPERTIES ARE NOT DEGRADED DURING CONSTRUCTION

DESIGN RULES APPLIED TO ENHANCE MATERIALS PERFORMANCE

REACTOR MATERIALS (CONT'D)

MATERIALS USED IN CONTACT WITH NUCLEAR GRADE 304 AND 316 STAINLESS STEEL

- LOW CARBON TO AVOID WELD SENSITIZATION
- NITROGEN CONTROL FOR STRENGTH
- GRAIN SIZE CONTROL
- SOLUTION HEAT TREATMENT USED WHERE EXPOSED TO SENSITIZING OPERATIONS

WELDING MATERIAL FOR STAINLESS STEEL

- CONTROL AS DEPOSITED FERRITE
- CONTROL COMPOSITION

STAINLESS STEEL CASTINGS

- LOW CARBON (CF3) GRADES USED
- FERRITE CONTROLLED
- SOLUTION HEAT TREATED

REACTOR MATERIALS (CONT'D)

XM 19

- AUSTENITIC STAINLESS WITH HIGHER STRENGTH
- USED FOR FASTENERS AND OTHER SPECIAL APPLICATIONS
- EACH LOT TESTED FOR STRESS CORROSION RESISTANCE
- SPECIAL STRESS AND FLUENCE LIMITS WHEN USED AS THREADED FASTENER

ALLOY 600

- USED WHERE HIGHER STRENGTH IS REQUIRED OR WHERE THERMAL EXPANSION MATCH WITH CARBON OR LOW ALLOY STEEL IS REQUIRED
- AVOID CREVICED WELDS
  - WHERE AVOIDING CREVICED WELDS NOT POSSIBLE USE STABILIZED MATERIAL
- APPLY STRESS RULES
- SOLUTION ANNEALED OR SOLUTION ANNEAL PLUS SPECIAL HEAT TREATMENT

REACTOR MATERIAL (CONT'D)

CARBON STEEL

- REQUIRE INTRINSICALLY TOUGH GRADES
- APPLY SPECIAL FATIGUE DESIGN RULES

LOW ALLOY STEEL

- APPLY SPECIAL CHEMISTRY CONTROLS FOR HIGH FLUENCE ZONES - COPPER, PHOSPHOROUS, NICKEL
- SPECIFY LOW TRANSITION TEMPERATURES -38F TO -20F

SPECIAL REQUIREMENTS

- USE HIGH PURITY MATERIAL IN HIGH FLUENCE APPLICATIONS, I.E. CONTROL BLADES
- USE LOW COBALT MATERIALS IN INTERNALS
- USE COBALT FREE WEAR MATERIALS

REACTOR MATERIALS (CONT'D)

SUMMARY

- MATERIALS BASED ON SUCCESSFUL EXPERIENCE
- MATERIALS CONTROLLED, PROCESSING CONTROLLED, DESIGN CONTROLLED
- COMPLY WITH REGULATIONS, CODES AND GUIDES
  - CRD SYSTEM
    - MATERIALS PROPERTIES EQUIVALENT TO ASME CODE SECTION II (REG. GUIDE 1.85) WITH ADDED LIMITS
    - REG. GUIDE 1.31 (FERRITE CONTROL)
    - REG. GUIDE 1.44 (SENSITIZED SS)
    - REG. GUIDE 1.37 (CLEANING)
  - INTERNALS
    - ASME II AND III
    - REG. GUIDE 1.31, 1.44, 1.85

FINE MOTION CONTROL ROD DRIVE SYSTEM

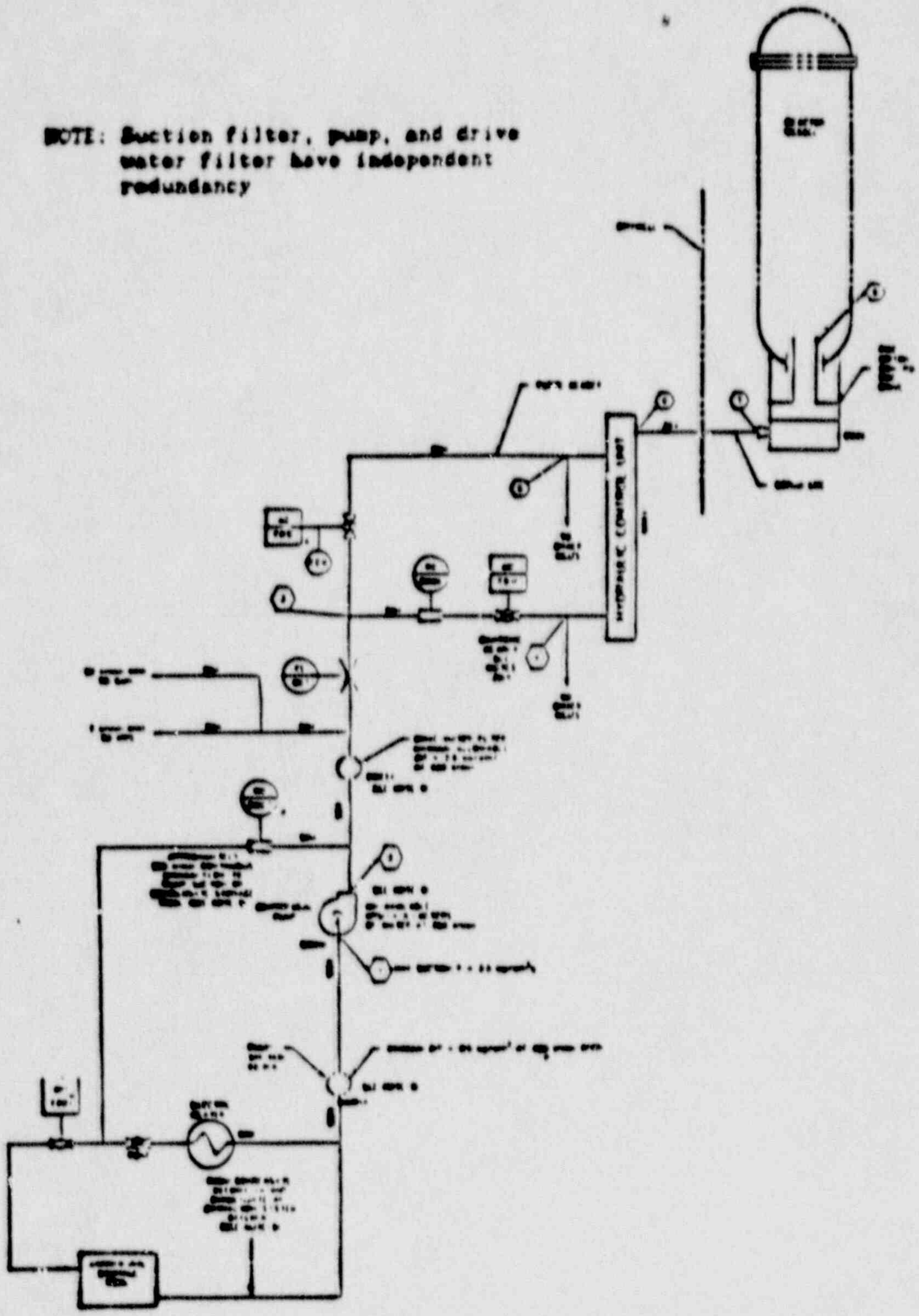
OVERVIEW

- FINE MOTION CONTROL ROD DRIVE HAS REDUNDANT MEANS OF INSERTION
  - HYDRAULIC SCRAM
  - ELECTRIC MOTOR DRIVEN INSERTION
- FINE MOTION
  - 18 MM STEP CAPABILITY
  - ALLOWS AUTOMATED STARTUP
  - FACILITATES LOAD FOLLOWING
- TWO FMCRDS PER HYDRAULIC CONTROL UNIT
  - EUROPEAN EXPERIENCE BASIS
  - SEPARATION MAINTAINED

ACRS REVIEW OF
GE ABWR

FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)

NOTE: Suction filter, pump, and drive water filter have independent redundancy



11/17/89 CWD 4.6-2

FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)

SYSTEM DESCRIPTION

- HYDRAULIC CONTROL UNIT INCLUDES
  - GAS BOTTLE
  - ACCUMULATOR
  - SCRAM SOLENOID VALVES
  - SCRAM VALVES
- CHARGING PUMPS (2)
  - ONE OPERATING, ONE SPARE
  - NON-SAFETY
- SCRAM LINES
  - ONE PER DRIVE
- REDUNDANT PROTECTION AGAINST CONTROL ROD EJECTION
  - BRAKE ON FMCRD
  - CHECK VALVE IN FMCRD

FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)

SYSTEM DESCRIPTION (CONT'D)

ELECTRIC SYSTEM

- STEPPING MOTOR
- POWER SUPPLY
- CONTROL LOGIC

SIGNIFICANT FEATURES

- ALLOWS FUNCTIONAL TESTING DURING OPERATION
- ALLOWS INDEPENDENT TESTING OF SCRAM CHANNELS AND ROD MOTION
- ROD PATTERN CONTROL SYSTEM MINIMIZES ROD WORTH - AVOID WITHDRAWAL ERROR
- ELIMINATES ROD DROP ACCIDENT
- NO SCRAM DISCHARGE VOLUME
  - REMOVES RADIATION SOURCE
  - ELIMINATES POTENTIAL COMMON MODE FAILURE
  - INTERNAL SHOOTOUT PROTECTION IN THE EVENT OF HOUSING WELD FAILURE

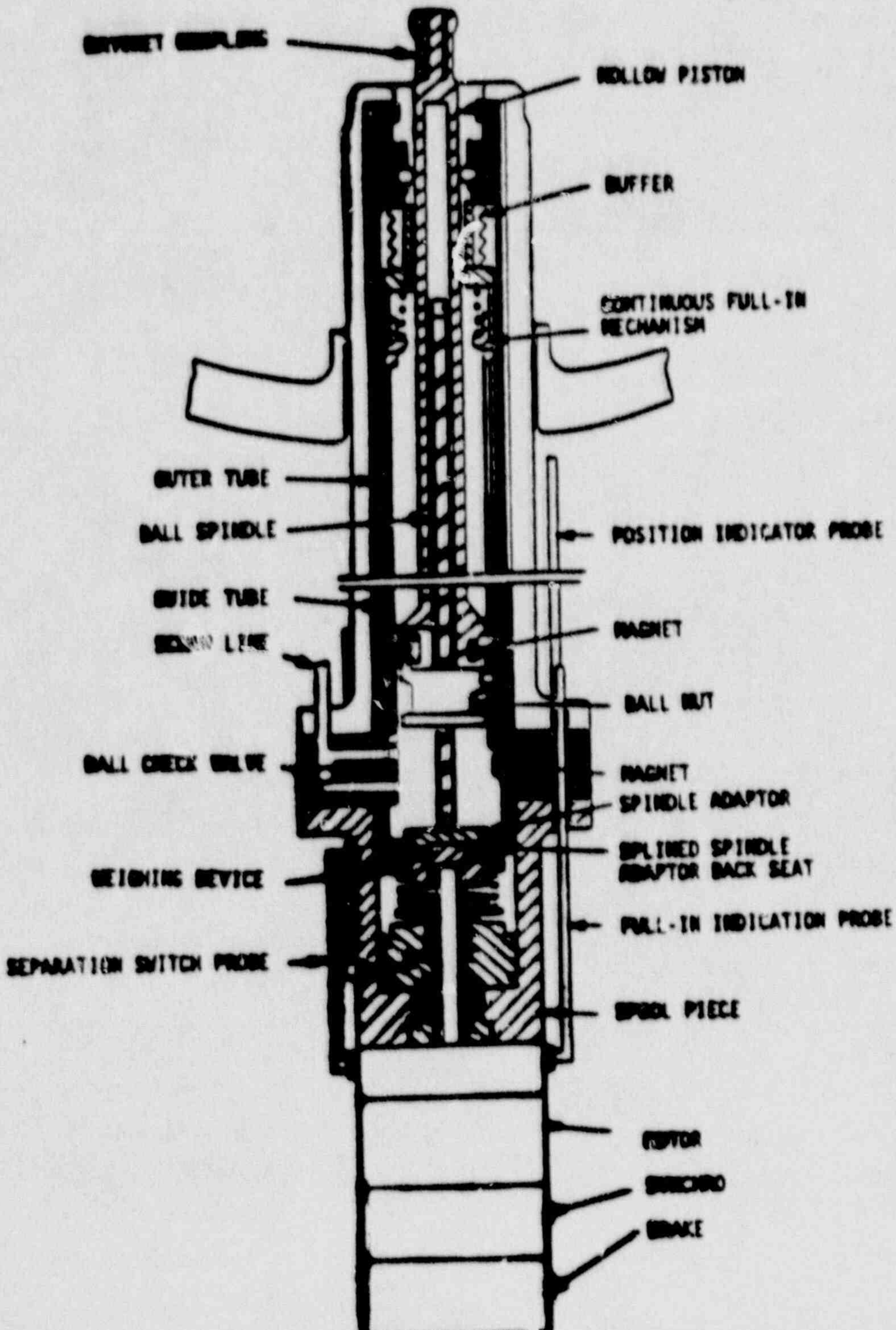
FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)

FMCRD DESCRIPTION

- FMCRD ORIGINATED IN EUROPE
  - NEARLY 2700 DRIVES IN SERVICE
  - OVER 15000 DRIVE YEARS OF EXPERIENCE
- FURTHER DEVELOPED BY GE, HITACHI AND TOSHIBA
  - SEISMIC TESTING
  - LIFE TESTING
- CONTROL ROD POSITIVELY COUPLED TO DRIVE - SEPARATION SWITCHES DETECT FAILURE OF ROD TO WITHDRAW
  - ELIMINATED CONCERN FOR ROD DROP
  - SEPARATION SWITCHES REDUNDANT AND CLIE
  - ALLOWS ELIMINATION OF VELOCITY LIMITER
- LOW MAINTENANCE/LOW EXPOSURE BASED ON EXPERIENCE

ACRS REVIEW OF
GE ABWR

FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)



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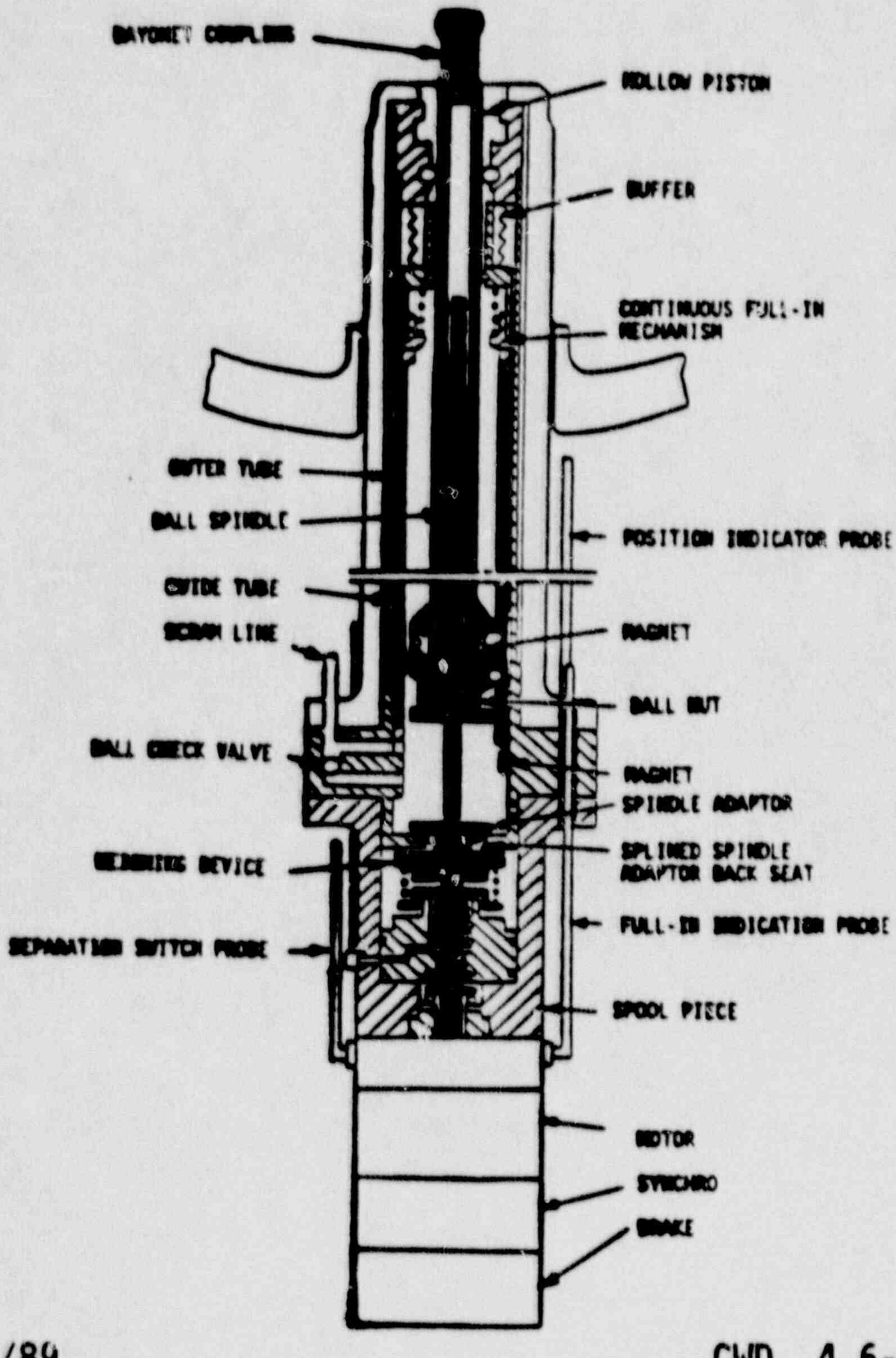
ACRS REVIEW OF
GE ABWR

FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)



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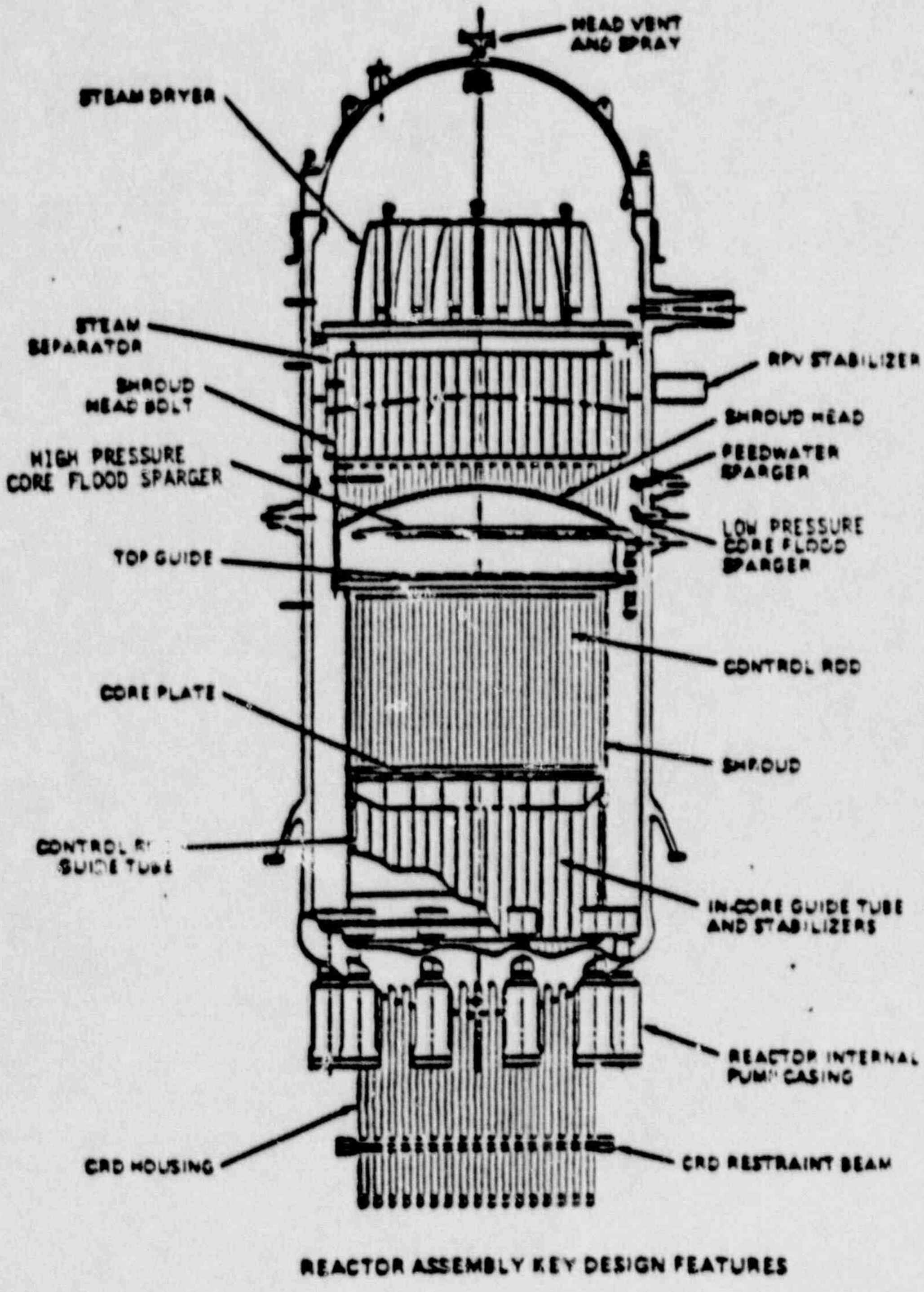
ACRS REVIEW OF
GE ABWR

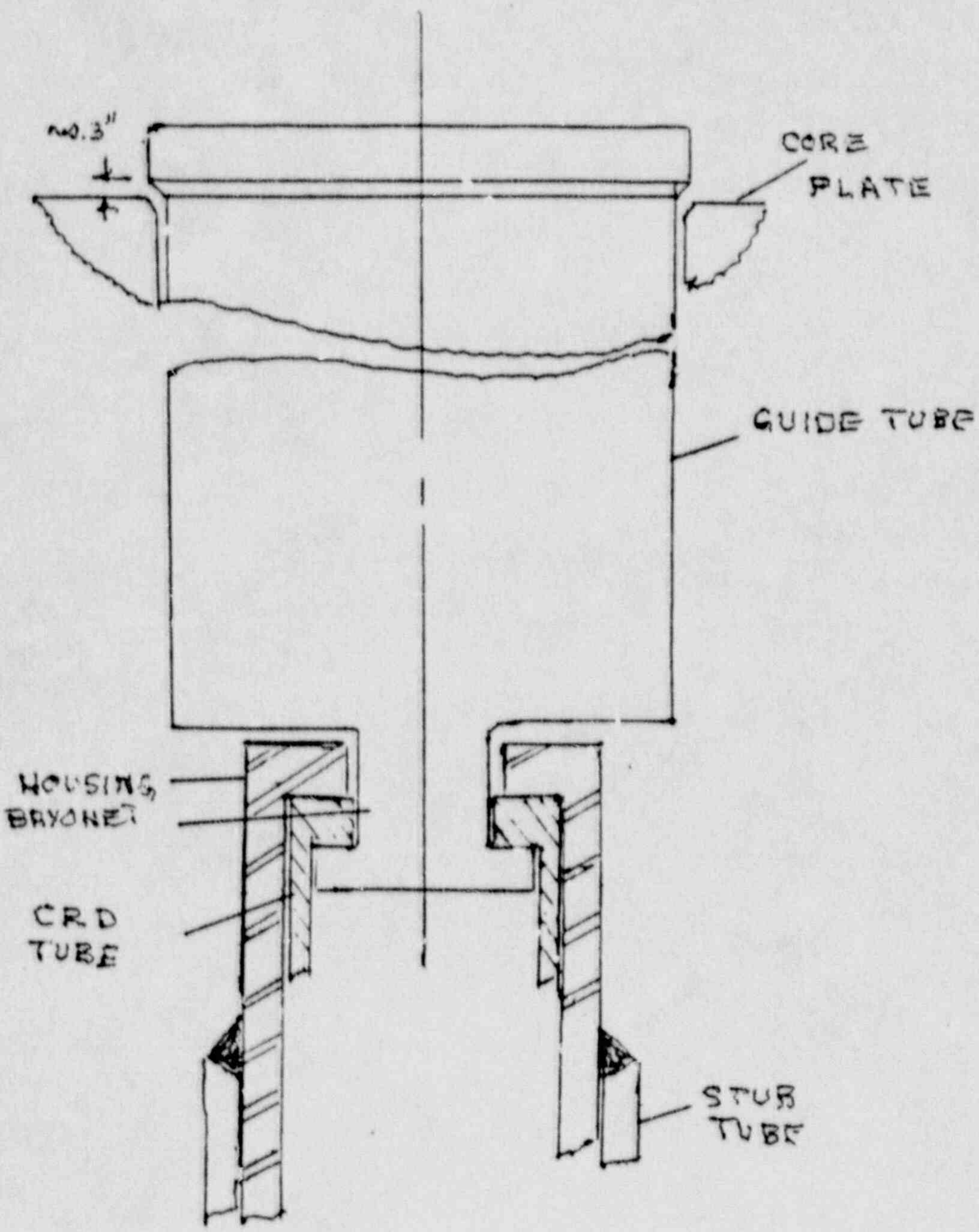
FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)



REACTOR ASSEMBLY KEY DESIGN FEATURES

ACRS REVIEW OF
GE ABWR

FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)



FINE MOTION CONTROL ROD DRIVE SYSTEM (CONT'D)

SUMMARY

- SYSTEM AND COMPONENTS BASED ON EXPERIENCE
- COMPONENTS FULLY TESTED
- PROVIDES
  - IMPROVED OPERABILITY
  - LOWER MAINTENANCE
  - LOWER EXPOSURE

ACRS REVIEW OF
GE ABWR

COMPLIANCE WITH 10CFR 50.55A - CODES AND STANDARDS

REACTOR COOLANT PRESSURE BOUNDARY

CLASSIFIED PER 10CFR 50.55A

MEET REQUIREMENTS OF ASME III CL 1

QUALITY GROUP A PER REG. GUIDE 1.26

ASME CODE DATES:

1986 EDITION

CODE CASES AS AUTHORIZED BY RGS 1.84 AND 1.85

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]
B1 Reactor Pressure Vessel System/ Fuel Assemblies					
1. Reactor vessel	1	C	A	B	I
2. Reactor vessel support skirt and stabilizer	1	C	A	B	I
3. Reactor vessel appurtenances -pressure retaining portions	1	C	A	B	I
4. Supports for CRD housing, in-core housing and recirculation internal pump	1	C	A	B	I
5. Reactor internal structures - safety related components including core support structures (See Subsection 3.9.5)	3	C	---	B	I
6. Reactor internal structures - non-safety related components (See Subsection 3.9.5)	N	C	---	---	---
7. Control rods	3	C	---	B	I
8. Power range detector hardware including startup range detector	3	C	---	B	I
9. Fuel assemblies	3	C	---	B	I

ACRS REVIEW OF
GE ABWR

OVER PRESSURE
PROTECTION

C. D. SAWYER

5.2.2 OVEPPRESSURE PROTECTION

- OVERPRESSURE PROTECTION CONFORMS TO 10CFR50, APPENDIX A, GDC15

- AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) INCLUDED WITH THREE LOW PRESSURE FLOODER SYSTEMS AND THREEHIGH PRESSURE FLOODER/FEED SYSTEMS (RCIC + TWO HPCF'S)

- THE ADS MAKES USE OF EIGHT OF 18 SAFETY RELIEF VALVES (SRV'S) OPERATED BY PNEUMATIC ACTUATORS

- SRV'S ARE DUAL FUNCTION:
  - SAFETY FUNCTION - LIFT AGAINST A SPRING FORCE
  - RELIEF FUNCTION - OPENED ON SIGNAL USING PNEUMATIC ACTUATOR

1375

- SRV'S LIMIT REACTOR PRESSURE TO 110% DESIGN PRESSURE PER ASME CODE SECTION III FOR MSIV CLOSURE WITH HIGH FLUX SCRAM

1150
55
1205

1250
125
1375

NUCLEAR SYSTEM SAFETY/RELIEF VALVE SETPOINTS

Set Pressures and Capacities

Number * of Valves	Spring Set Pressure (psig)	ASME Rated Capacity at 103% Spring Set Pressure (lb/hr each)	Relief Pressure Set Pressure** (psig)
1	1150	883,000	1090
1	1150	883,000	1100
4	1160	891,000	1110
4	1170	898,000	1120
4	1180	906,000	1130
4	1190	913,000	1140

---

* *Eight of the SRV's serve in the automatic depressurization function.*

** *Closing setpoint is 100 psi below opening setpoint.*

OVERPRESSURE EVALUATION

EVENT	PEAK VESSEL BOTTOM PRESSURE (PSIG)	PEAK STEAMLINE PRESSURE (PSIG)
MSIV CLOSURE	1242	1208
LOAD REJECTION	1205	1179
LOAD REJECTION W/O BYPASS	1247	1218
MSIV CLOSURE FLUX SCRAM (ASME EVENT*)	1274	
ATWS	1336	

*NO CREDIT FOR RELIEF FUNCTION

ACRS REVIEW OF

GE ABWR

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

MATERIALS ARE:

- CARBON STEEL
- STAINLESS STEEL
- LOW ALLOY STEEL
  - BOLTS AND RPV ONLY
- LIMITED USE OF PRECIPITATION HARDENED MATERIAL
  - VALVE SPINDLES/STEMS
- MATERIAL REQUIREMENTS DISCUSSED EARLIER APPLY
- PREFILMING OF STAINLESS EMPLOYED TO MINIMIZE RADIATION BUILDUP
- CONDENSER TUBES AND TUBE SHEET TITANIUM
- HYDROGEN ADDITION CONSIDERED TO MITIGATE IASCC

BWR WATER CHEMISTRY

	Concentrations* Parts Per Billion (ppb)					Conductivity		Electro-Chemical Corrosion Potential
	Iron	Copper	Chloride	Sulfate	Oxygen**	μS/cm at 25°C	pH at 25°C	V at 25°C
Condensate	<20	<2	<4	<4	<10	~0.15	~7	-----
Condensate Treatment Effluent and Feedwater	<2.2	<0.02	<0.4	<0.4	20-50	<0.059	~7	-----
Reactor Water								
(a) Normal Operation	<20	<1	<20	<20	**	<0.3	~7	<-0.23
(b) Shutdown	<20	<1	<20	<20	-	<1.2	~7	-----
(c) Hot Standby	<20	<1	<20	<20	<200	<0.3	~7	-----
(d) Depressurized	<20	<1	<20	<20	high (may be 1000 to 8000)	<1.2	5.6-8.6	----
Control Rod Drive Cooling Water	<2.2	<0.1	<0.4	<0.4	20-50	≤0.059	~7	-----

---

* *These limits should be met at least 90% of the time.*

** *Some revision of oxygen values may be established after hydrogen water chemistry has been established.*

REACTOR COOLANT PRESSURE BOUNDARY MATERIAL (CONT'D)

- CORROSION/EROSION RESTRAINT LOW ALLOY STEELS USED IN SUSCEPTIBLE LOCATIONS
- NON-METALLIC INSULATION APPLIED TO AUSTENITIC STAINLESS STEEL CONTROLLED TO AVOID SCC - MEETS RG 1.35
- FERRITE MATERIAL MEETS IMPACT REQUIREMENTS OF ASME
- WELD CONTROLS INCLUDE:
  - PREHEAT, INTERPASS TEMPERATURES
  - HYDROGEN CONTROL
  - SPECIAL WELDER QUALIFICATION IN AREAS OF LIMITED ACCESSIBILITY (RG 1.17)
  - HEAT INPUT CONTROL
- ELECTROSLAG WELDING NOT USED

REACTOR COOLANT PRESSURE BOUNDARY MATERIAL (CONT'D)

SUMMARY

- MATERIALS SELECTED BASED ON SUCCESSFUL EXPERIENCE
- CONTROLS APPLIED TO
  - PROCESSING
  - FABRICATION
  - CONTAMINANTS
  - WATER CHEMISTRY
- SPECIFIC AREAS OF ATTENTION
  - COBALT
  - SENSITAIZATION
  - DUCTILITY
  - IGSCC AND IASCC
- ALL CODES, STDS AND REGULATIONS COMPLIED WITH

ACRS REVIEW OF

GE ABWR

REACTOR COOLANT PRESSURE
BOUNDARY LEAKAGE DETECTION

LEAKAGE DETECTION SYSTEMS APPLIED WHRE REQUIRED IN THE PLANT

METHODS OF DETECTION INCLUDE TEMPERATURE, PRESSURE, RADIATION, FLOW

ACTIONS INCLUDE ALARM AND ISOLATION

SYSTEMS COVERED ARE:

- MAIN STEAMLINES
- HIGH PRESSURE CORE FLOODER
- RESIDUAL HEAT REMOVAL
- REACTOR WATER CLEANUP
- FEEDWATER
- COOLANT SYSTEMS WITHIN DRYWELL
- REACTOR PRESSURE VESSEL
- MISCELLANEOUS

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION (CONT'D)

DRYWELL

- SMALL UNIDENTIFIED LEAKS
  - PRIMARY METHOD SUMP PUMP ACTIVITY AND SUMP LEVEL
    - 1 GPM WITHIN 1 HOUR SENSITIZED
    - CONTINUOUS INDICATION/RECOTRDING IN CONTROL ROOM
    - NO ISOLATION TRIP - ALARM
  - SECONDARY DETECTION - PRESSURE AND TEMPERATURE IN DRYWELL
    - PRESSURE CAUSES ISOLATION
    - TEMPERATURE ALARM ONLY

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION (CONT'D)

DRYWELL (CONT'D)

- SMALL IDENTIFIED LEAKAGE
  - LIMIT 25 GPM TO ALARM
  - LEAKAGE PRIMARILY FROM VALVE STEMS
- VARIABLES MEASURED
  - HIGH DRYWELL TEMPERATURE
  - HIGH TEMPERATURE IN AREA OF STEAMLINE GUARD PIPES
  - HIGH SUMP FLOW
  - HIGH STEAMLINE FLOW RATE
  - HIGH DRYWELL PRESSURE
  - HIGH FISSION PRODUCT RADIATION
  - REACTOR VESSEL LOW WATER LEVEL
  - RPV HEAD SEAL DRAIN LINE HIGH TEMP
  - SPV DISCHARGE TEMP HIGH

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION (CONT'D)

- LEAKAGE EXTERNAL TO DRYWELL
  - AREAS COVERED INCLUDE
    - EQUIPMENT AREAS IN THE REACTOR BUILDING
    - MAIN STEAM TUNNEL
    - TURBINE BUILDING
  - MONITORED VARIABLES
    - WITHIN REACTOR BUILDING
      - STEAM LINE AND RCIC STEAMLINE HIGH FLOW
      - RPV LOW WATER LEVEL
      - HIGH FLOW RATE FROM SUMPS
      - HIGH EQUIPMENT SPACE TEMPERATURE - RCIC, RHR AND HOT PORTION OF RWCS
      - RCIC TURBINE EXHAUST HIGH DIAPHRAGM PRESSURE

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION (CONT'D)

- LEAKAGE EXTERNAL TO DRYWELL (CONT'D)
  - WITHIN REACTOR BUILDING (CONT'D)
    - HIGH DELTA FLOW -RWCS
    - HIGH RADIATION IN RBCW DISCHARGE LINES - RHR, RWCS, RIP AND FPC
    - RCIC LOW STEAM LINE PRESSURE
  - WITHIN STEAM TUNNEL
    - HIGH RADIATION
    - HIGH TUNNEL AIR TEMP
  - WITHIN TURBINE BUILDING
    - STEAM LINE LOW PRESSURE
    - LOW CONDENSOR VACUUM
    - HIGH AREA TEMPERATURE AROUND STEAM LINES

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION (CONT'D)

KEY FEATURES

- LEAKAGE TOTAL WELL WITHIN MAKEUP CAPABILITY OF RCIC SYSTEM (800 GPM)
- EXCEEDING LEAKAGE RATES RESULTS IN ORDERLY SHUTDOWN
- DIFFERENTIATION BETWEEN IDENTIFIED AND UNIDENTIFIED LEAKAGE
- SYSTEM TESTABILITY PROVIDED
- MEETS REG GUIDE 1.45

ACRS REVIEW OF
GE ABWR

REACTOR VESSEL MATERIALS

- MATERIAL PRIMARILY LOW ALLOY STEEL PLATE AND FORGING
  - PLATE ASME SA533 TYPE B CL1 FORGING ASME SA508 CL3
  - FINE GRAIN PRACTICE
  - VACUUM DEGASSED
  - LOW COPPER (.05%), PHOSPHOROUS (0.015%), NICKEL (1.2%)S IN BELTLINE
  - WELD METAL .08% AND .020% WITH NICKEL 1.29%
  - 100% UT EXAM PER ASME III DIV. 1
  - FRACTURE TOUGHNESS PER DIV 1
- STUDS, NUTS AND WASHERS ASME SA540 GRADE B23 OR B24.

REACTOR VESSEL MATERIALS (CONT'D)

NO WELDS IN HIGH FLUENCE ZONE - FORGINGS

PROCESSING MEETS ALL CODES, STANDARDS AND REGULATIONS

SURVEILLANCE SPECIMENS INCLUDED PER 10CFR 50 APP H AND ASTM E 185

ACRS REVIEW OF
GE ABWR

REACTOR PRESSURE VESSEL
PRESSURE TEMPERATURE LIMITS

CALCULATED PRESSURE/TEMPERATURE LIMITS CALCULATED BASED ON 10CFR 50 APPENDIX G

BOLT-UP TEMP 70 F

SHIFT IN RT_{NDT} CALCULATED PER REG GUIDE 1.99

- WELD METAL 28 F
- SHELL 8 F
- LOW SHIFT DUE TO MATERIAL AND LOW FLUENCE ($6 \times 10^{17} N/CM^2$)
- LOW FLUENCE DUE TO LARGE ANNULUS

EVALUATION OF MARGIN TO NON DUCTILE FAILURE

- WORST UPSET CASES
  - 1215 PSIG AT 528F (CURVE C)
  - 930 PSIG AT 250F (CURVE B) AFTER SCRAM

ACRS REVIEW OF
GE ABWR

REACTOR VESSEL INTEGRITY

INTEGRITY ASSURED BY

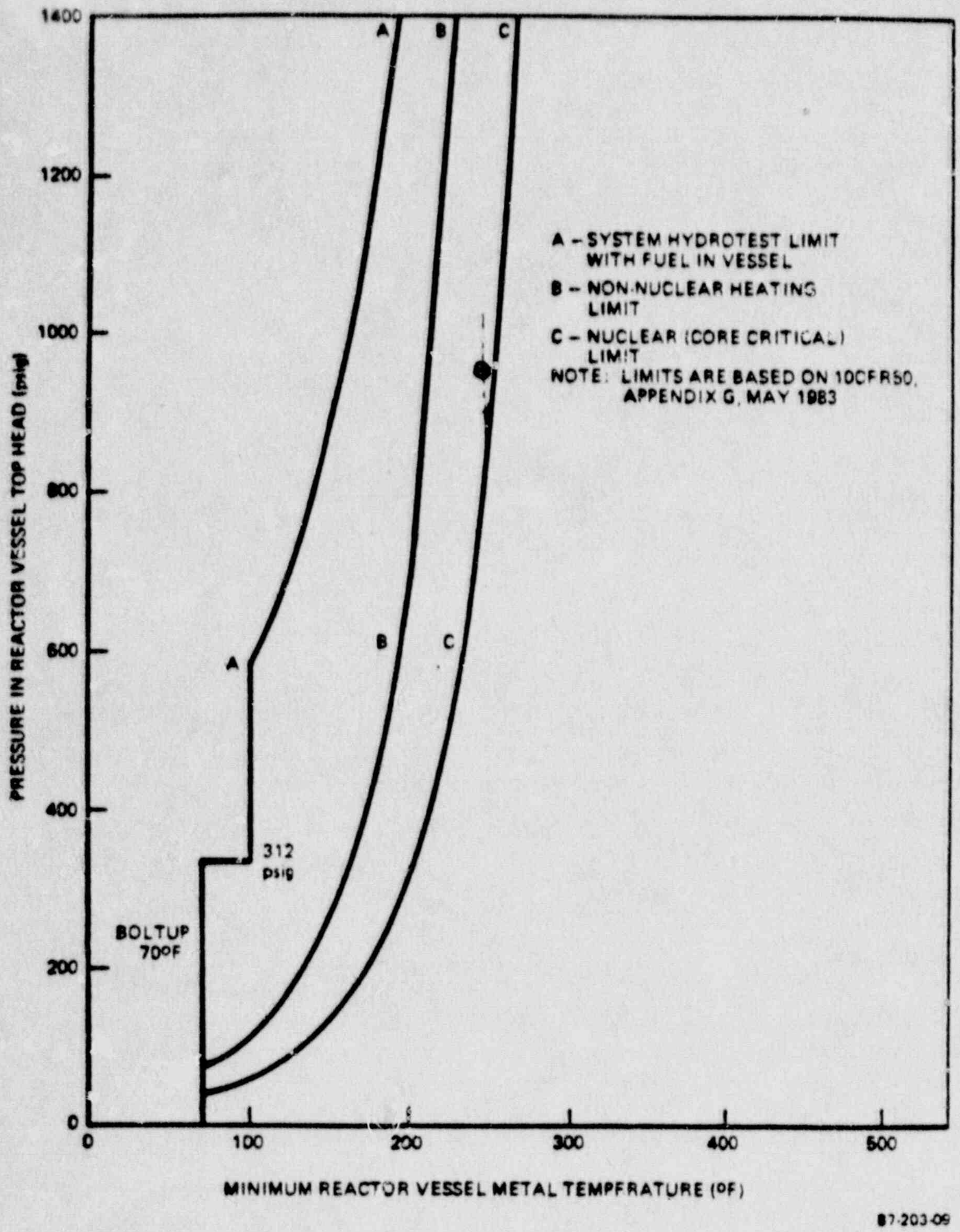
- MATERIAL CONTROL
- FABRICATION CONTROL
- OPERATIONAL MARGIN
- DESIGN PRACTICES
  - ASME III CL 1
  - CONSIDERS ALL OPERATING CONDITIONS
- IN-SERVICE INSPECTION
- SURVEILLANCE PROGRAM

REACTOR VESSEL INTEGRITY (CONT'D)

INSERVICE ANNEALING

- NOT REQUIRED
  - RT_{NDT} < 200 F
  - EVALUATION OF DBA AND EMERGENCY CORE COOLING SHOWS NO RISK OF BRITTLE FRACTURE

ACRS REVIEW OF
GE ABWR



MINIMUM TEMPERATURE REQUIRED VERSUS REACTOR PRESSURE

ACRS REVIEW OF
GE ABWR

REACTOR RECIRCULATION SYSTEM

OVERVIEW

- USES 10 INTERNAL PUMPS
  - NO LARGE PIPES
  - EUROPEAN EXPERIENCE
- ADJUSTABLE SPEED DRIVES
  - 70 - 100% LOAD FOLLOWING
  - SOLID STATE
- RECIRC PUMP TRIP
- MG SETS ON 6 RIPS ENHANCE COAST DOWN
  - THERMAL MARGIN

REACTOR RECIRCULATION SYSTEM (CONT'D)

RIP DESCRIPTION

DESIGN BASED ON EUROPEAN EXPERIENCE

WET MOTOR

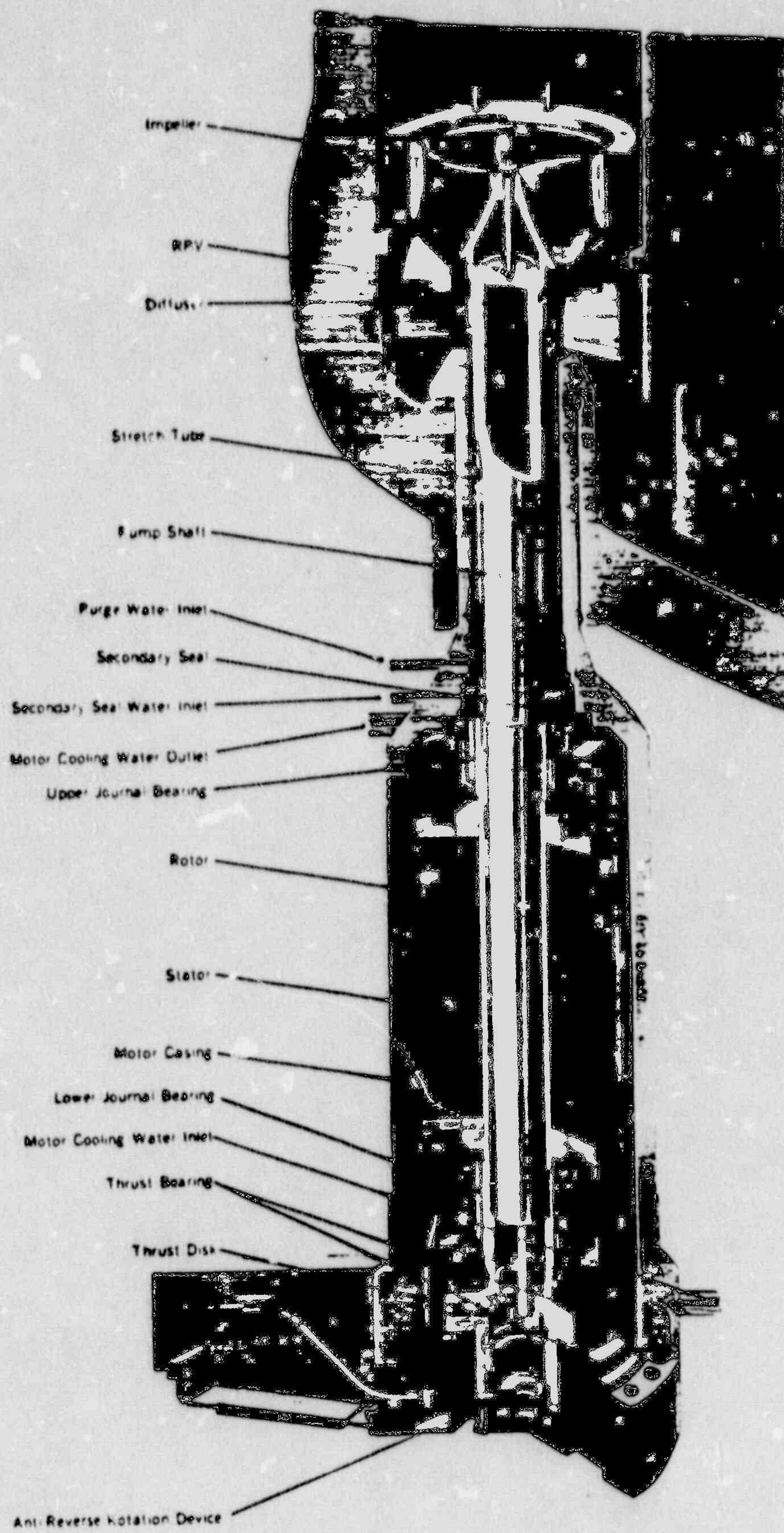
IMPROVEMENTS INCORPORATED BASED ON EXPERIENCE AND TEST

HIGH VOLTAGE

IMPROVED BEARINGS

INCLUDES BACKSEAT AND INFLATABLE SEAL FOR SERVICING

ACRS REVIEW OF
GE ABWR



REACTOR RECIRCULATION SYSTEM (CONT'D)

ABNORMAL EVENTS

FMEA DONE

INCLUDES MISSILE POTENTIAL

SHORT CIRCUIT

LOSS OF COOLING

CASING FAILURE

LOSS OF PURGE

NO SIGNIFICANT PROBLEM

EVEN COMPLETE FAILURE OF CASING TO VESSEL WELD CAN BE MADE UP BY RCIC + PLUS CRD

REACTOR RECIRCULATION SYSTEM (CONT'D)

OTHER FEATURES

- PURGE SYSTEM
  - MAINTAINS LOW CONTAMINATION
- FLOW MEASUREMENT SYSTEM
  - PUMP Δ P
  - CORE Δ P
- CAPABILITY FOR HIGH POWER WITH PUMPS OUT OF SERVICE

ACRS REVIEW OF
GE ABWR

RCICS

RHR

RWCS

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5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)

SYSTEM FUNCTION:

- o DELIVER REACTOR WATER MAKE UP DURING
  - ISOLATION TRANSIENTS WITH NO FEEDWATER
  - LOCA EVENTS
  - LOSS OF AC POWER EVENTS

- o PROVIDE SUFFICIENT FLOW TO
  - AVOID EMERGENCY SYSTEM INITIATION DURING TRANSIENTS
  - SUPPORT LOCA OBJECTIVE OF NO FUEL UNCOVERY

KEY DESIGN FEATURES:

- o ONE 800 GPM SYSTEM, DRIVEN BY TURBINE USING REACTOR STEAM

- o UPGRADED TO BE PART OF ECCS NETWORK WITH NO MAJOR CHANGES

- o PRIMARY SUCTION IS CONDENSATE STORAGE TANK (CST). BACKUP SUCTION IS SUPPRESSION POOL (S/P)

- o AUTOMATIC SUCTION TRANSFER TO S/P WITH MANUAL OVERRIDE

- o CONDENSATE STORAGE TANK LEVEL SWITCHES ARE SEISMICALLY INSTALLED

- o AUTO RESTART CAPABILITY ON LOW WATER LEVEL 2 REOCCURRENCE FOLLOWING HIGH WATER LEVEL 8 TRIP

- o SYSTEM INITIATION /OPERATION DOES NOT REQUIRE AC POWER

- o CAPABLE OF HIGHER TURBINE EXHAUST BACK PRESSURE OPERATION (50PSIG) TO SUPPORT SMALL BREAK LOCA MITIGATION

- o BYPASS START FEATURE INCLUDED

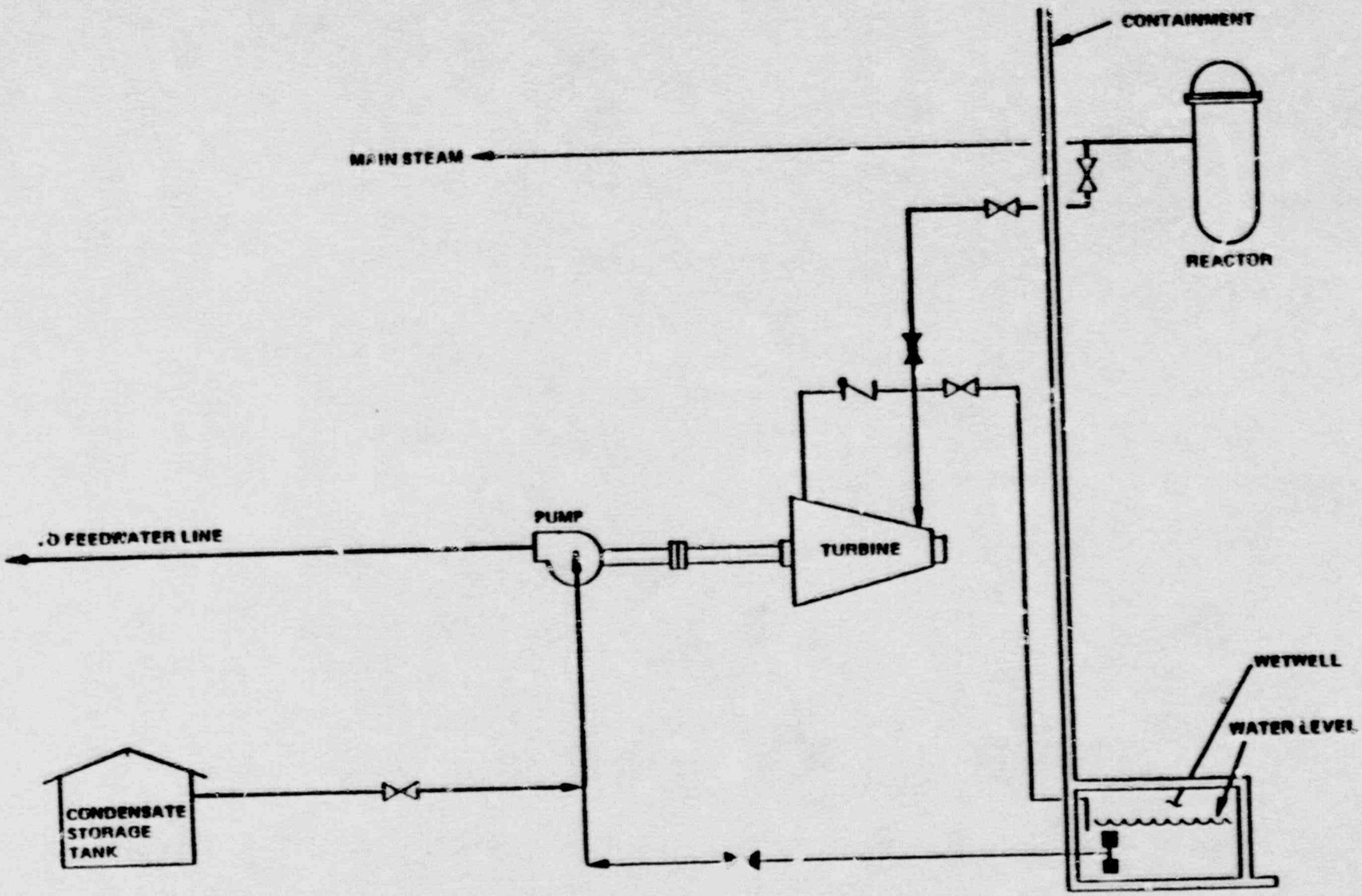
5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) (CONT'D)

PERFORMANCE EVALUATION:

- PREVENTS WATER LEVEL FROM DROPPING BELOW LEVEL 1.5 WHICH AVOIDS INITIATION OF HPCF FOLLOWING ABNORMAL TRANSIENTS.
- PREVENTS WATER LEVEL FROM DROPPING BELOW LEVEL 1 WHICH AVOIDS INITIATION OF ADS FOLLOWING A SMALL BREAK LOCA ($\leq$1 INCH)
- >0.97 RELIABILITY

SUMMARY:

- RCIC MEETS ALL DESIGN BASES AND IS CAPABLE OF MITIGATING ABNORMAL TRANSIENTS AND SMALL BREAK LOCAs



Reactor Core Isolation Cooling (RCIC) Schematic Flow Diagram

5.4.7 RHR SYSTEM

LOW PRESSURE FLOODER MODE

DESIGN BASE REQUIREMENT

- o AS PART OF ECCS MAINTAIN FUEL CLADDING TEMPERATURE LIMITS

- o MAINTAIN SUPPRESSION POOL TEMPERATURE UNDER 207°F

- o ACHIEVE THIS WITH
  - N-1; ONE LOOP FAILED
  - LOSS OF OFFSITE AC POWER SOURCES

DESCRIPTION

- o AUTOMATIC PUMP START FOR
  - HIGH DRYWELL PRESSURE

    OR
  - LOW LEVEL 1 REACTOR WATER

  RUNS AT MINIMUM FLOW

- o AUTOMATIC FLOODER INJECTION WHEN REACTOR DEPRESSURIZED

- o 4200 GPM AT 40 PSID (COMPARE TO 5000-7000 FOR BWR-6)

- o 30 MINUTES BEFORE OPERATOR ACTION NEEDED

PERFORMANCE

- o FOR FEEDWATER LINE BREAK DBA (CONTAINMENT)
  - S/P LIMITED TO 207°F

- o FOR HPCF LINE BREAK DBA (CORE)
  - NO CORE UNCOVERY

5.4.7 RHR SYSTEM

SHUTDOWN COOLING MODE

EMERGENCY OPERATION (RG 1.139):

DESIGN BASE REQUIREMENTS

- o COOL RPV TO 212°F WITHIN 36 HOURS
- o N-1; ONE LOOP FAILED

DESCRIPTION

- o DEPRESSURIZE TO 135 PSIG OR LESS
- o FLOW PATH
  - -RPV SUCTION
  - -RHR PUMP
  - -RHR HEAT EXCHANGER
  - -RPV RETURN
- o MANUAL INITIATION

PERFORMANCE

- o COOL TO 212°F IN LESS THAN 12 HOURS

NORMAL OPERATION:

DESIGN BASE REQUIREMENTS

- o COOL RPV TO 140°F WITHIN 24 HOURS
- o THREE LOOPS OPERATIONAL

DESCRIPTION

- o DEPRESSURIZE AND FLOW PATH AS ABOVE

PERFORMANCE

- o COOLING ACCOMPLISHED WITH APPROXIMATELY 7% MARGIN FROM HEAT EXCHANGER FULL HEAT LOAD CAPACITY

5.4.7 RHR SYSTEM

SUPPRESSION POOL COOLING MODE

DESIGN BASE REQUIREMENT

- o COOL S/P AFTER RPV DEPRESSURIZATION
- o COOL S/P PERIODICALLY FOR SRV WARMING

DESCRIPTION

- o MANUAL INITIATION
- o FLOW PATH
  - -S/P SUCTION
  - -RHR PUMP
  - -RHR HEAT EXCHANGER
  - -S/P RETURN
- o ONE TO THREE LOOPS AVAILABLE

PERFORMANCE

- o 50% OF LOCA CONDITION HEAT LOAD FOR EACH HEAT EXCHANGER
- o COOLING CAPABILITY MORE THAN SUFFICIENT FOR SHUTDOWN COOLING REQUIREMENTS

5.4.7 RHR SYSTEM

CONTAINMENT COOLING MODE

DRYWELL SPRAY

DESIGN BASE

- o PROVIDE STEAM CONDENSATION AFTER LOCA AS A BACKUP
  -EASE CONTAINMENT THERMAL ENVIRONMENT

DESCRIPTION

- o TWO LOOPS HAVE D/W SPRAY; COMMON SPRAY HEADER
- o MANUAL INITIATION
- o HIGH D/W PRESSURE NECESSARY TO OPERATE
- o 88% OF RHR RATED FLOW

PERFORMANCE

- o LONG-TERM D/W TEMPERATURE REDUCED

WETWELL SPRAY

DESIGN BASE

- o CONDENSE STEAM FROM D/W TO W/W BYPASS LEAKAGE

DESCRIPTION

- o TWO LOOPS HAVE W/W SPRAY; COMMON SPRAY HEADER
- o MANUAL INITIATION
- o 12% OF RHR RATED FLOW

PERFORMANCE

- o KEEPS W/W PRESSURE BELOW DESIGN VALUE

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5.4.7 RHR SYSTEM

FUEL POOL ASSIST COOLING

DESIGN BASE

- o ASSIST FPC IF MORE THAN A 35% CORE FUEL BATCH IS REMOVED DURING REFUELING

- o RHR AND FPC REMOVE DECAY HEAT OF
  - 100% CORE +
  - 4 PREVIOUS 35% CORE FUEL BATCHES +
  - 5TH PREVIOUS OUTAGE 30% CORE FUEL BATCH

DESCRIPTION

- o TWO 200% LOOPS

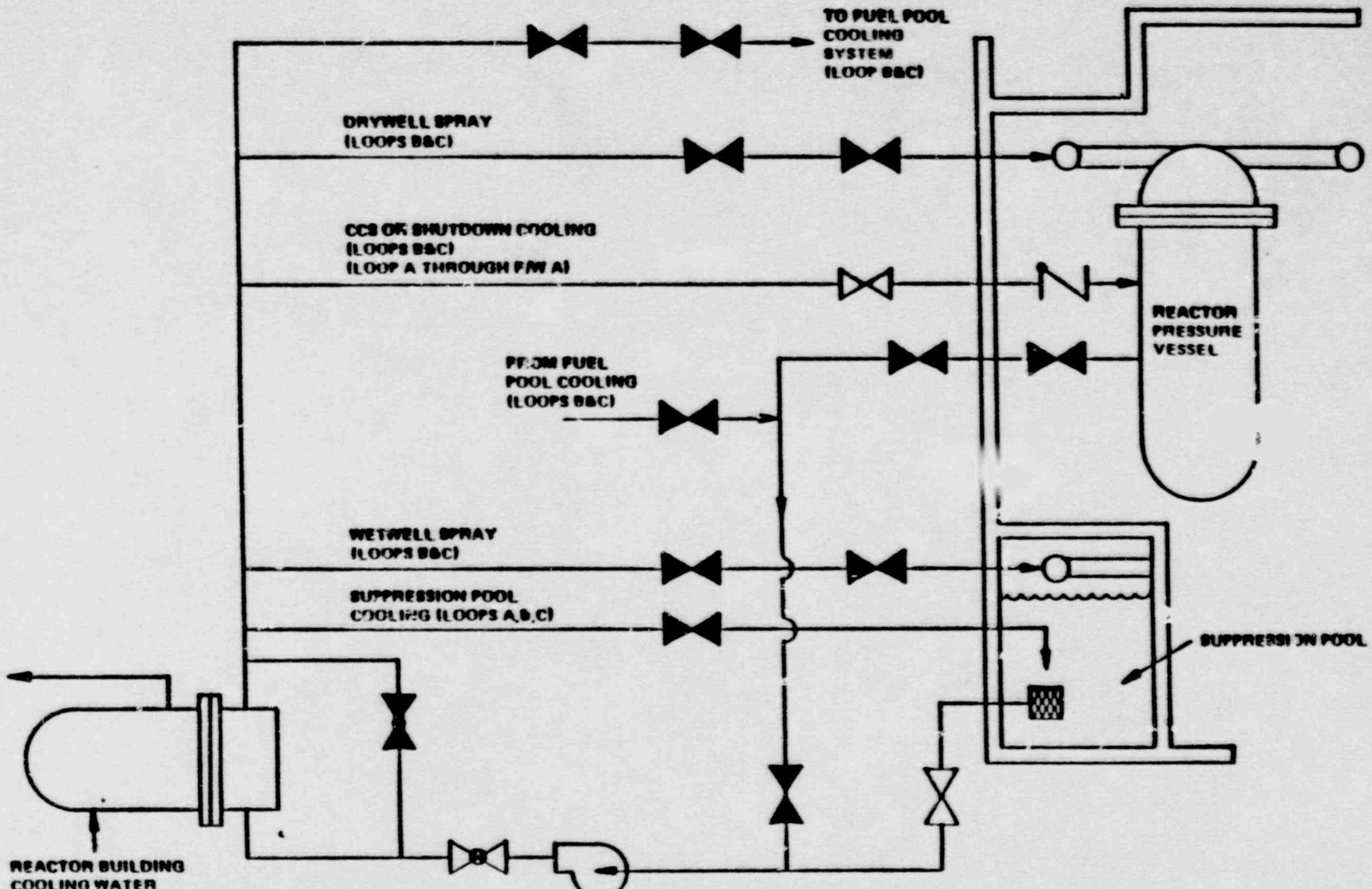
- o MANUAL INITIATION

- o FLOW PATH
  - SKIMMER SURGE TANK
  - RHR PUMP
  - RHR HEAT EXCHANGER
  - FUEL POOL DISTRIBUTION SPARGER

PERFORMANCE

- o ONE LOOP CAN REMOVE APPROXIMATELY TWICE THE REQUIRED HEAT LOAD

RHR FULLY SATISFIES ALL REQUIREMENTS



Residual Heat Removal System (RHR)

5.4.8 REACTOR WATER CLEANUP SYSTEM

SYSTEM FUNCTION:

- MAINTAIN REACTOR WATER QUALITY WITHIN SPECIFIED LIMITS WHILE MINIMIZING HEAT LOSSES

- DISCHARGE EXCESS REACTOR WATER DURING STARTUP SHUTDOWN AND HOT STANDBY CONDITIONS TO RADWASTE

- PROVIDE RPV HEAD SPRAY IF REQUIRED FOR FASTER COOLDOWN

5.4.8 REACTOR WATER CLEANUP SYSTEM

SYSTEM DESCRIPTION

- RATED AT 2% OF REACTOR FEEDWATER FLOW
- PUMPS IN COLD LEG DOWNSTREAM OF THE NON-REGENERATIVE HEAT EXCHANGERS
  - SEAL-LESS MOTOR DESIGN MORE RELIABLE
  - COLD LEG PRODUCES LOWER RADIATION SHOULD MAINTENANCE BE REQUIRED
- RETURN FLOW TO FEEDWATER VIA RHR AND RCIC
- 1X2% REGENERATIVE HEAT EXCHANGER
- 2X1% NON-REGENERATIVE HEAT EXCHANGERS
- 2X1% SEAL-LESS PUMPS
- 2X1% FILTERS/DEMINERALIZERS
- BACKWASH EQUIPMENT SHARED WITH FUEL POOL COOLING CLEANUP SYSTEM
- FILTER-DEMINERALIZER OPERATION FULLY AUTOMATED

5.4.8 REACTOR WATER CLEANUP SYSTEM

DESIGN BASES:

- o PROVIDE CLEANUP FLOW AT A MAXIMUM OF 2% OF FEEDWATER
  - MEET WATER QUALITY REQUIREMENTS
  - PROVIDE REASONABLY RAPID CLEANUP AFTER WATER QUALITY UPSET (IMPLYING INTRUSION)
  - BALANCE WATER QUALITY AND HEAT REJECTION COSTS
  - EXCELLENT NONCONDENSER AND CONDENSATE TREATMENT DESIGN COUPLED WITH LARGER RWCU WILL PROVIDE IMPROVED WATER QUALITY
- o NON-SAFETY SYSTEM
- o PROVIDE CONTAINMENT ISOLATION WHICH LIMITS POTENTIAL FOR SIGNIFICANT RELEASE OF RADIOACTIVITY TO SECONDARY CONTAINMENT VS. RWCU
- o ISOLATE ON LEAK DETECTION, LOCA, OR SLCS ACTUATION

1

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

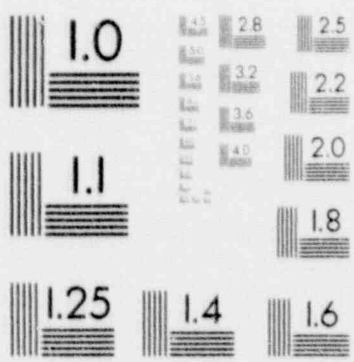
IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



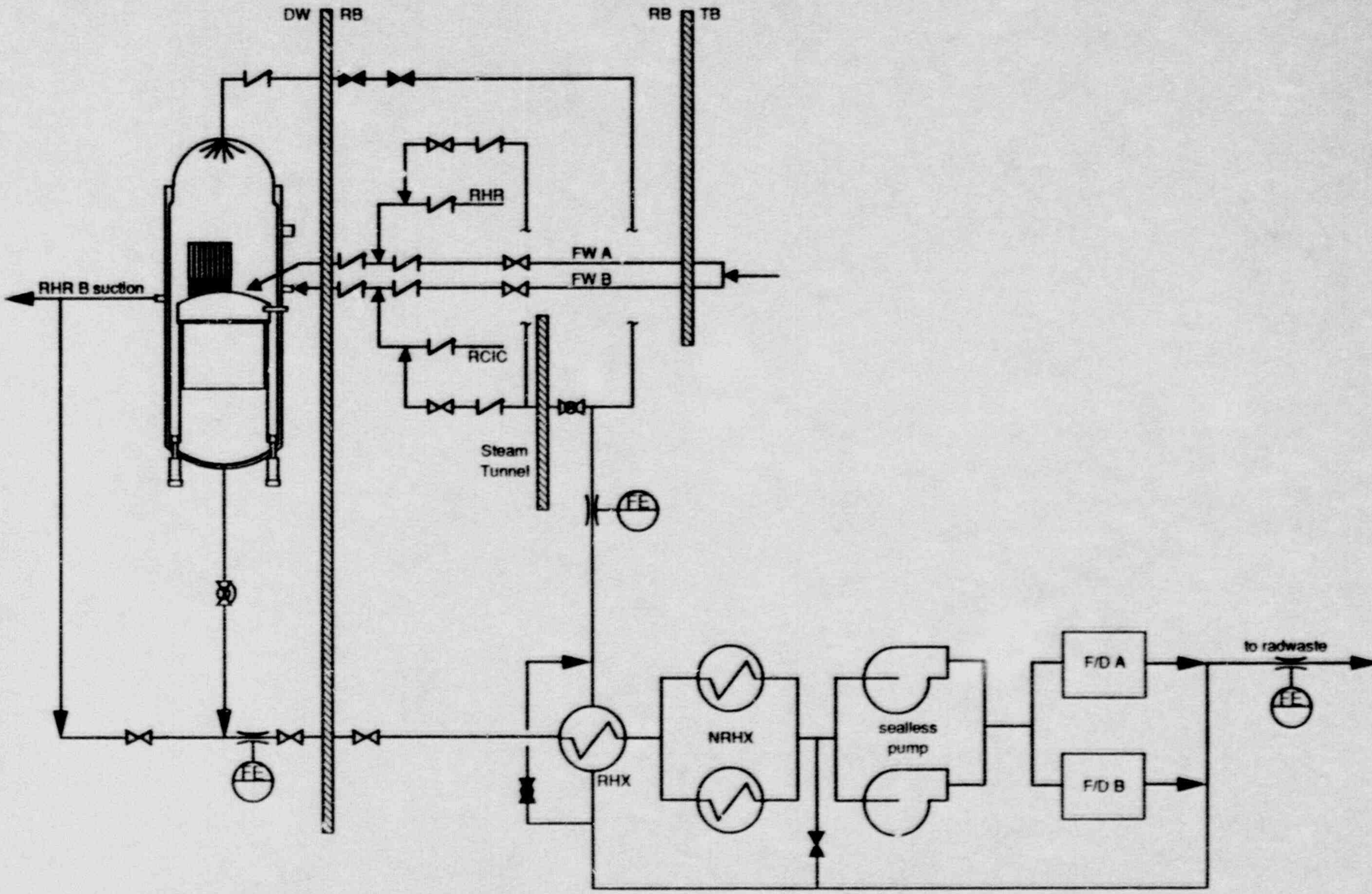
150mm

6″

SUMMARY

- INCREASED FLOW FROM 1% TO 2% OF RATED FEEDWATER IMPROVED SYSTEM RESPONSE TO PRIMARY SYSTEM INTRUSIONS
- SEALLESS PUMPS ELIMINATE SEAL LEAKAGE, IMPROVING PUMP RELIABILITY AND SYSTEM AVAILABILITY
- PUMPS IN COLD LEG MINIMIZES PERSONNEL EXPOSURE SHOULD MAINTENANCE BE REQUIRED

MEETS ALL SYSTEM REQUIREMENTS



Reactor Water Cleanup System

ACRS REVIEW OF
GE ABWR

ENGINEERED SAFETY FEATURES

METALLIC MATERIALS

MATERIAL SAME AS DISCUSSED EARLIER

PROCESSING CONTROLS SAME

FLUID PURE WATER

- EXCEPTION IS TO POTENTIAL BORON INJECTION THROUGH HPCF

ORGANIC MATERIALS - ENGINEERED SAFETY FEATURES

ORGANIC COATINGS

- CONTAINMENT LINES
- STRUCTURES
- EQUIPMENT
- EPOXY COATING QUALIFIED TO ANSI FOR LOCA ENVIRONMENT
  - MEET RG 1.54
  - EXCEPTIONS ARE SMALL ITEMS - NEGLIGIBLE IMPACT

OTHER ORGANIC MATERIALS

- MATERIALS CONSISTENT WITH EXPECTED ENVIRONMENT
- QUALIFIED FOR ENVIRONMENTAL CONDITIONS

ACRS REVIEW OF
GE ABWR

ECCS

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6.3 EMERGENCY CORE COOLING SYSTEMS

ABWR SAFETY SYSTEM FEATURES

- THREE COMPLETELY SEPARATE MECHANICAL & ELECTRICAL DIVISIONS FOR MOST IMPORTANT FUNCTIONS
  - CORE COOLING
  - SUPPRESSION POOL COOLING
  - SHUTDOWN COOLING

- AUTOMATION OF POST-LOCA POOL COOLING
  - HEAT EXCHANGERS ALWAYS IN THE LOOP

- ELIMINATION/TRANSFER OF COMPLEX MODES
  - STEAM CONDENSING
  - RPV HEAD SPRAY
  - CONTAINMENT FLOOD
  - REDUCED VALVES, PIPES BY ONE-THIRD

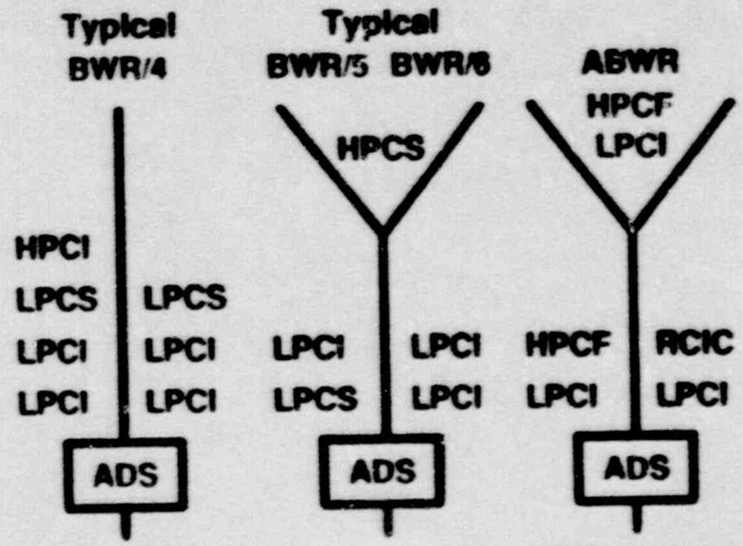
- SIGNIFICANT CAPACITY REDUCTION
  - REDUCED EQUIPMENT SIZES

- GREATLY REDUCED DUTY DURING TRANSIENTS
  - N-2 CAPABILITY AT HIGH PRESSURE

- IMPROVED SMALL BREAK RESPONSE
  - REDUCED NEEDS FOR ADS

- NO FUEL UNCOVERY FOR ANY PIPE BREAK

ECCS KEY PERFORMANCE FEATURES



	Typical BWR/4	Typical BWR/5 BWR/6	ABWR
High Pressure* Capacity	5000	1900	2800
Low Pressure** Capacity	42000	29000	19000
No. of Large Pipes Below Core	12	12	0
Peak Clad Temp (APP K), °F	1600	1100	No Uncovery
N-2 Capability	None	All but HPCS Break	All but LPCI (A) Break

* @1100 psi
** @ 100 psi

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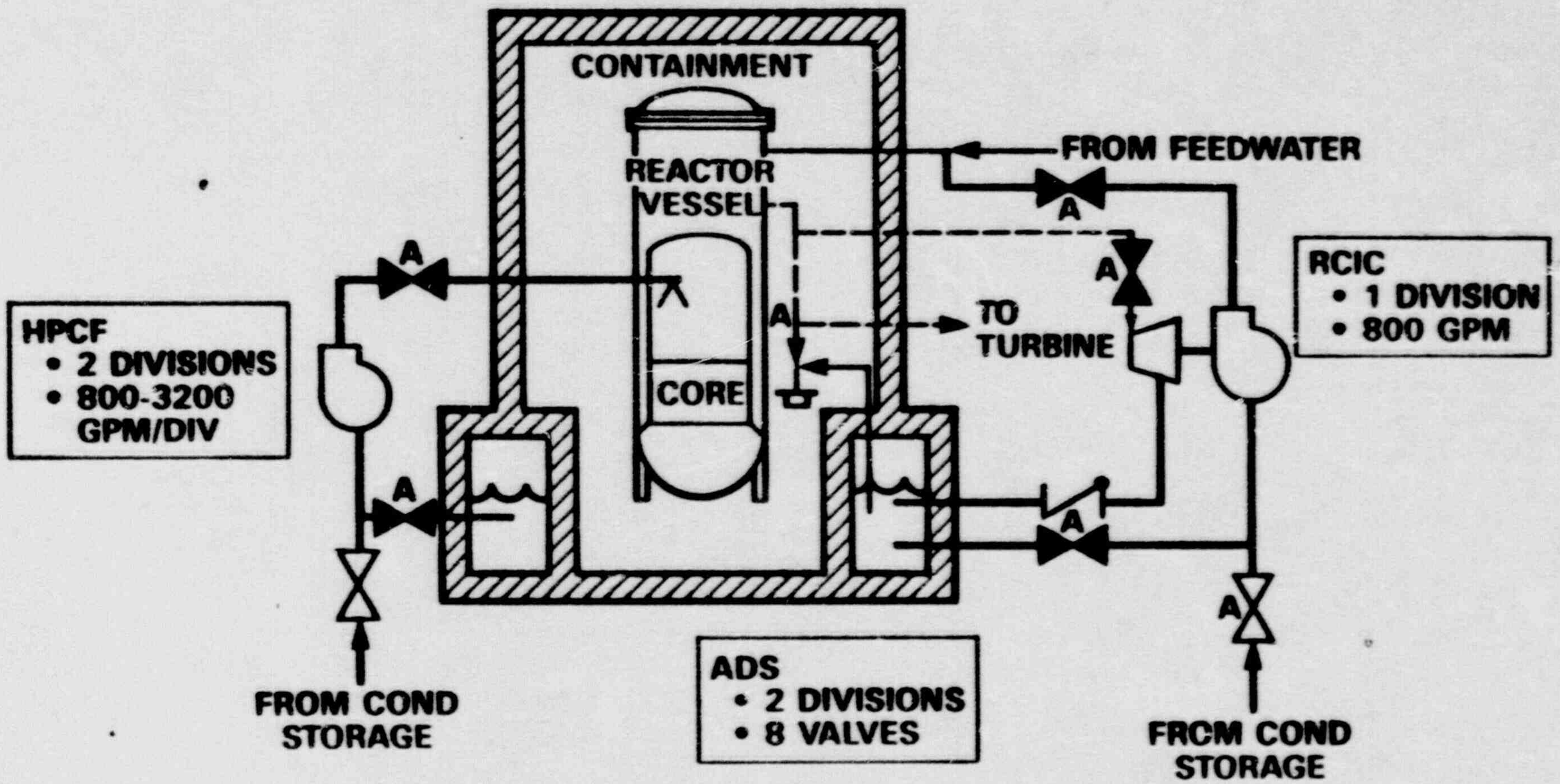
ABWR EMERGENCY CORE COOLING SYSTEMS

High Pressure



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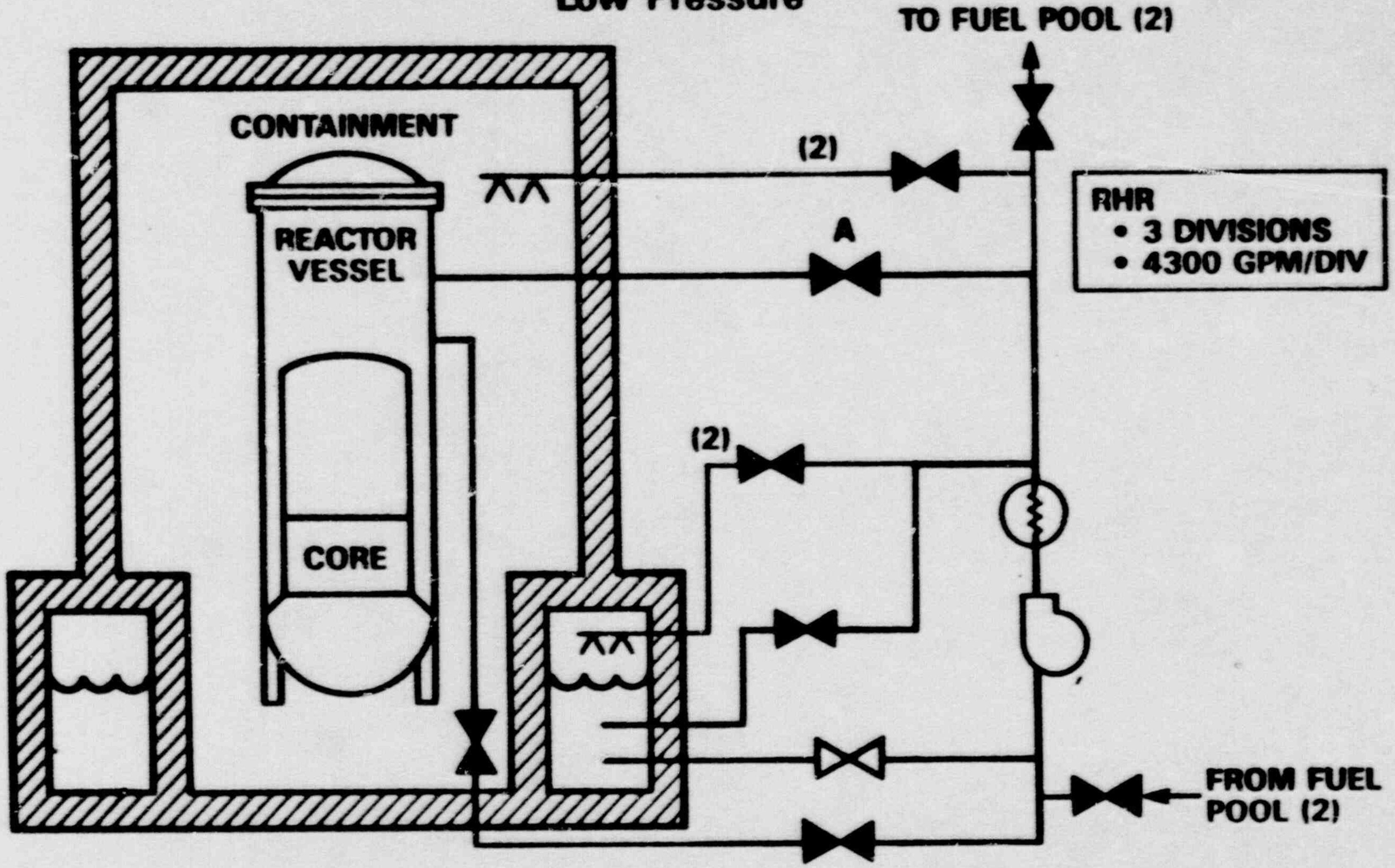
ABWR EMERGENCY CORE COOLING SYSTEMS

Low Pressure



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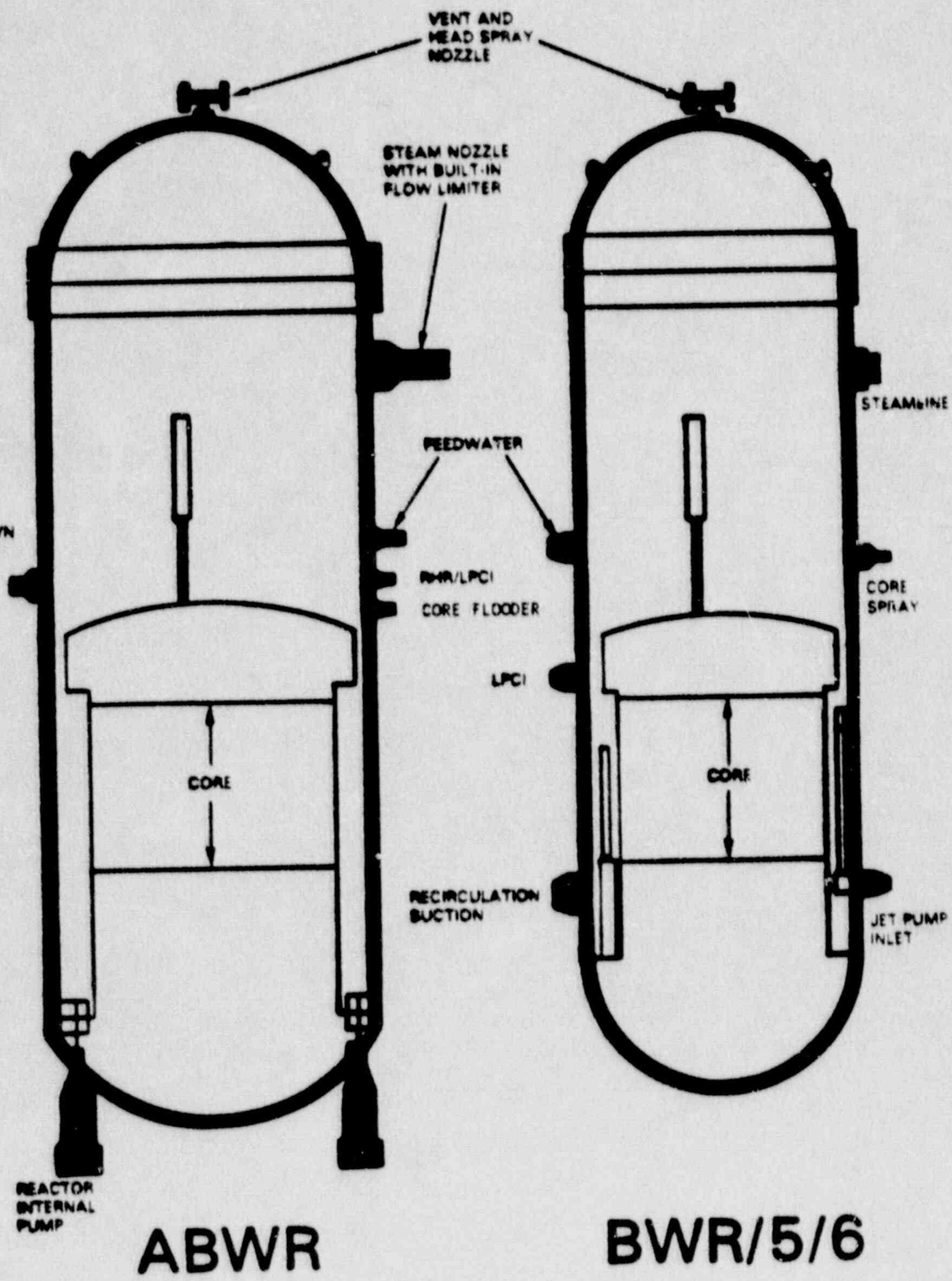
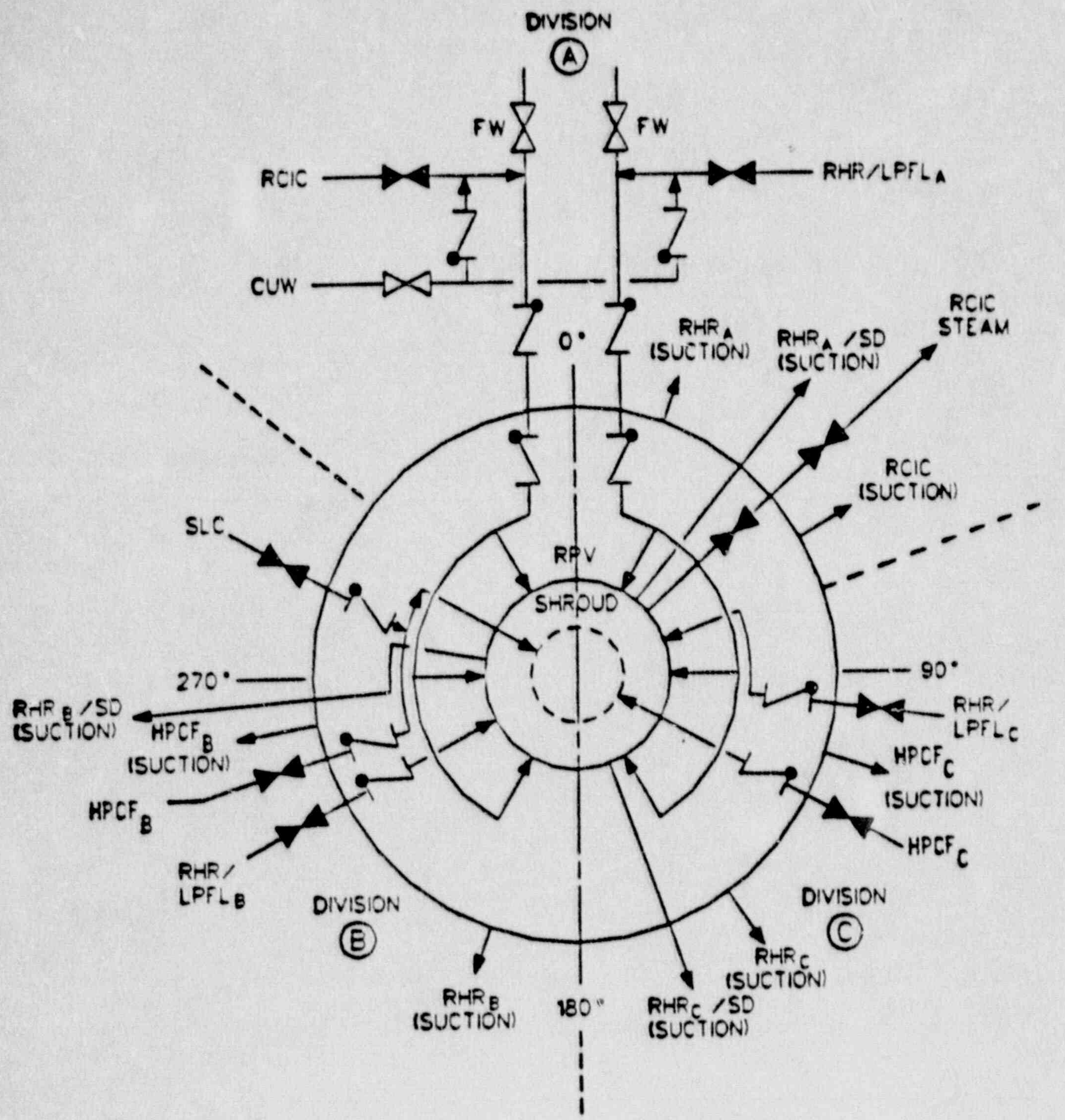


Figure 3. Vessel Nozzle Comparison



ECCS VESSEL NOZZLES (PLAN VIEW)

ABWR LOCA RESPONSE

- o NO CORE UNCOVERY FOR ANY PIPE BREAK
- o NO CORE HEATUP FOR NOMINAL CASE
- o MINIMAL CORE HEATUP OCCURS WHEN ALL RIPs ASSUMED TO TRIP AT TIME ZERO (ITSELF AN ACCIDENT). THIS PRODUCES A RAPID RECIRCULATION FLOW COASTDOWN.

ABWR
STANDARD PLANT

CHAPTER 17 QUALITY ASSURANCE

- o DESIGN RESPONSIBILITY

  - GE AND ITS TECHNICAL ASSOCIATES, HITACHI AND TOSHIBA, PERFORM JOINT "COMMON ENGINEERING" FOR 2 ABWR'S IN JAPAN.

    - THE 3 COMPANIES ARE RESPONSIBLE FOR ALL DESIGN UNDER THE JOINT EFFORT.

ABWR
STANDARD PLANT

CHAPTER 17 QUALITY ASSURANCE

o DESIGN QA PROCESS

- EACH COMPANY MUST FORMALLY REVIEW AND APPROVE EACH COMMON ENGINEERING DOCUMENT.
  - LEAD RESPONSIBILITY IS ASSIGNED FOR EACH DOCUMENT. THIS INCLUDES:
    - DRAFT DOCUMENT
    - INTERNALLY PROCESS & VERIFY
    - OBTAIN REVIEW BY OTHERS
    - RESOLVE COMMENTS
    - OBTAIN FORMAL APPROVAL
    - ISSUE AND MAINTAIN
    - CONTROL CHANGES

ABWR
STANDARD PLANT

CHAPTER 17 QUALITY ASSURANCE

- QUALITY ASSURANCE

  - GE AND ITS TECHNICAL ASSOCIATES, HITACHI AND TOSHIBA, ARE COMMITTED TO THE QA PROCEDURES IN THE "ABWR ORGANIZATION AND PROCEDURES MANUAL". THIS REQUIRES THAT:

    - GE, HITACHI, AND TOSHIBA MEET BOTH JEAG-4101-1981 AND 10CFR50, APPENDIX B.

    - EACH PARTY MAY INITIALLY REVIEW THE ADEQUACY OF THE OTHER'S QA PROGRAM.

    - EACH PARTY ANNUALLY REVIEWS THE IMPLEMENTATION OF THE OTHER'S QA.

ABWR
STANDARD PLANT

CHAPTER 17 QUALITY ASSURANCE

o ABWR CERTIFICATION QA

- GE WORKS TO AN NRC ACCEPTED QA PROGRAM

  - THE GE PROGRAM COMPLIES WITH ALL QUALITY RELATED REG. GUIDES IN EFFECT MARCH 31, 1987 OR NRC ACCEPTED ALTERNATE POSITIONS.

- GE IS RESPONSIBLE FOR THE CONTENT OF ALL COMMON ENGINEERING DOCUMENTS. QUALITY IS ASSURED BY:

  - FORMAL REVIEW AND APPROVAL OF EACH DOCUMENT.

  - ANNUAL REVIEW OF HITACHI AND TOSHIBA QA.