ORIGINAL ACRST-1999

OFFICIAL TRANSCRIPT OF PROCEEDINGS

Agency: Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards

Title: Meeting of the Subcommittee on ABB-CE
Standard Plant Designs

Docket No.

LOCATION: Bethesda, Maryland

DATE: Wednesday, March 9, 1994 **PAGES:** 299 - 543

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UNITED STATES NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

DATE: March 9, 1994

The contents of this transcript of the proceedings of the United States Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards, (date) March 9, 1994, as Reported herein, are a record of the discussions recorded at the meeting held on the above date.

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Meeting of the Subcommittee on

ABB-CE Standard Plant Designs

U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Conference Room P-110
Bethesda, Maryland

Wednesday, March 9, 1994

The above-entitled proceedings commenced at 8:30 a.m., pursuant to notice, Jay Carroll, chairman of the subcommittee, presiding.

PRESENT FOR THE ACRS SUBCOMMITTEE:

J. Carroll	P. Davis
T. Kress	W. Lindblad
C. Michelson	R. Seale
W. Shack	E. Wilkins
C. Wylie	D. Coe

PARTICIPANTS:

On Behalf of ABB-CE

S. Ritterbusch T. Crom
D. Finnicum C. Brinkman
S. Stamm E. Siegmann
L. Gerdes J. Trotter
F. Carpentino M. Cross
R. Schneider J. Longo
T. Oswald

On Behalf of NRC/NRR

M. Franovich R. Li
D. Terao J. Guo
B. Palla R. Architzel
J. Lyons N. Saltos
A. El-Bassioni

PROCEEDINGS

[8:30 a.m.]

MR. CARROLL: Let's reconvene our subcommittee on ABB-CE standard plant design. The plan today is to begin by picking up the remaining sections of Chapters 2 and 3 that we didn't get to yesterday and then Pete, we're going to talk about PRA. We're the same cast of characters as yesterday except Ivan, of course, is not here and we are joined by Bill Shack.

MR. MICHELSON: Good trade.

MR. CARROLL: Good trade; yeah. So -- you're going to make the presentation; you're on.

MR. CROM: Okay. I'm Tom Crom from Duke Engineering and Services. I am currently the engineering manager for the System 80+ project from Duke Engineering and Services. I have been with Duke Power since 1976 when I started on the design at Catawba Nuclear Station, and stayed on Catawba Nuclear Station until about 1985 when I started on this particular project, and I've been on this project since 1985.

This morning I want to start with inservice testing of pumps and valves, then get into flood protection and then, finally, talk about high energy line breaks. And my discussion on high energy line breaks is pretty much going to be where the lines are located, particularly outside a containment, so that we can talk about any interaction that may be with high energy line breaks.

As far as what I'm going to talk about on inservice testing, safety-related pumps and valves, I'm going to tell you basically, generally what we did in the program in CESSAR-DC. We have a very extensive table that has all pumps and valves that are in the standard plant, safety-related pumps and valves listed along with the test requirements. I'm going to talk about what we have in that table, the provisions we have for testing pumps and valves, and finally I'm going to go over the numerous COL applicants, because there are a lot of COL applicant-type items that are in this particular chapter.

[Slide.]

MR. CROM: My first slide here -- and the reason I have it is we spent a lot of time with the staff going over, particularly in valves, what the safety function is. We went through almost every valve that we have on our PNIDs and determined what the safety function was, determined what the design bases were, to first determine if it belonged in the IST program.

For example, there are some valves and stuff that are in the CVCS system that are considered in the safety class pressure boundary but have no safety function. And those particular valves are not included in the program and at the same time, there are some valves that may not have a pressure or may not have a leak-type requirement for their safety function but may have a stroke requirement. And we went through the definition of safety-related in each one of these cases, in determining whether the valve belonged in the program or what its safety function and what test did it need to have.

The major portion of what we did is in table 3.9-15 where list every safety-related pump and valve in the standard design. We developed this plan utilizing the ASME/ANSI OMa-1988 for Part 1 for Relief Valves, Part 6 for Pumps, and Part 10 for Valves other than relief valves.

Of course, also in there, as far as leak rate testing and test requirements also 10 CFR 50 Appendix J, testing for containment isolation valves is also included when we were developing the plan.

The types of headings we have in this table, for pumps we have -- for each pump we give the pump name, the ASME safety class, whether safety class 1, 2, or 3, the test parameter that is required to be done, listing those there, differential pressure, static suction pressure, operating suction pressure, flow and speed for a pump, the test frequency, whether it needs to be done quarterly at cold shutdown or refueling. For pumps, we were able to do quarterly testing. We've reviewed all our pumps and we're doing quarterly testing on all pumps.

We also provide in that table the test configuration for various test connections and stuff to ensure that the design is such that we can do these tests, and also list where the pump is located on the figure number within CESSAR-DC so it can be found.

As far as valves --

MR. CARROLL: Do I remember wrong? I thought vibration was also on that list.

MR. CROM: Yes, it is. It is. I believe it's -- I'm trying to remember if that's included on the table or whether it's just discussed in the SAR. The vibration is included as part of the inservice testing.

As far as valves, we list each valve number, valve name, valve type, whether it's a globe, gate, check valve, whether -- what type of actuator, whether it has an actuator, whether it's an EMO, whether it's an air-operated valve, hydraulic and so forth. Of course, the ASME safety class whether it's safety class 1, 2, or 3. The ASME code category, either A, B, C, D; A being that it's late-type requirements, either a pressure isolation valve, containment isolation valve, or a temperature isolation valve; B that it requires a stroke test; C that it's a check valve; and D is an explosive valve, which we have no explosive valves in the plant. There is no D category of valves. And of course there can be a combination of these, depending on what the actual valve function is.

We also list the valves that are -- function if they are pressure isolation valves -- in particular we're talking about the RCS pressure isolation valve -- a temperature isolation valve, which is a new requirement the staff asked us to add, with some of the problems of valves where there is a high temperature on one side actually leaking into a system and causing system problems and heat up. We listed those in the appropriate test to ensure that we don't have leakage or are able to detect leakage, and of course the containment isolation valves are listed.

We also put the required testing -- you know, whether it requires a stroke test, a leak test, whether it requires valve position verification, reverse flow stroke time, failure position, whether they can do a failure position if it has an air operated valve that fails to open or close or whether it requires a bench test if it's a relief valve.

We also have the test frequency which is either quarterly, refueling or cold shutdown and one thing I'm going to talk about a little bit later, the number of valves that we have to test either at refueling or cold shutdown, the number has reduced significantly compared to current plans.

MR. DAVIS: Excuse me; is the valves in your reactor cavity flooding system -- are they considered safety-related?

MR. CROM: Yes.

MR. DAVIS: And what's the testing frequency of those?

MR. CROM: I don't recall right offhand, but we can look -- I can look up in the table what it is. Sandy, you know --

MR. RITTERBUSCH: Yeah; at refueling outages.

MR. DAVIS: You can't test them during operation, I guess, can you?

MR. CROM: Yes -- I should -- yeah; the reason for that now, I recall, is you have to go inside down into the hold up volume and you have to shut a manual valve before you can do the stroke test or else you'll be flooding into the hold up volume.

MR. LINDBLAD: Mr. Crom, are there requirements to test after maintenance is done?

MR. CROM: Yes. There is.

MR. LINDBLAD: And does that apply across the board or just on certain valves?

MR. CROM: That's across the board.

MR. MICHELSON: Now in case of -- a few of the valves, the steam-driven auxiliary feedwater for instance -

-

MR. CROM: Yes.

MR. MICHELSON: -- you have the possibility that if you are experience that type rupture you have to isolate under blowdown conditions. How do you differentiate that from your full differential to open or full differential at closure?

MR. CROM: That is --

MR. MICHELSON: There is a difference.

MR. CROM: Yes. And we got a COL action item. That's one that I have listed, that the COL applicant has to determine the design bases conditions, specify what those differential pressures are. He then has to do a type test on the valves in the shop and do then a --

MR. MICHELSON: Where is -- is that in the staff's SER, then?

MR. CROM: I don't know whether it's in the staff -- it is in our --

MR. MICHELSON: It is in here somewhere?

MR. CROM: In our SER.

MR. MICHELSON: Maybe if you find the reference I'll read it.

MR. CROM: I've got it on my slide. It is in 3.9-6. There's a whole series on -- for pumps, valves --

MR. MICHELSON: I did find the blowdown tests in that discussion, though, but maybe I'm --

MR. CROM: It asks for a different -- it states the differential pressure, the differential pressure that has to close down.

MR. MICHELSON: But that doesn't do it, as you probably well know from the valve testing that was done there. The difference between opening under static, even though it's full differential conditions, and opening under dynamic conditions because of the differences in hydraulics around the disk as it's rising and so forth. The fact is on some tests it took a larger torque after the valve was cracked open, even, than during the full closure portion of the cycle.

MR. CROM: Yeah.

MR. MICHELSON: So you have to do that type of test, and I couldn't find on the table how you identify it. There are only a handful of valves that need that, but alternatively you could make it a COL action. I don't know -- but you'd probably know today which valves they'd have to be. Obviously auxiliary feedwater is one of them.

MR. DAVIS: Emergency.

MR. MICHELSON: Pardon me; emergency feedwater in this case.

MR. CROM: It's considered under the design basis and the COL applicant would determine the design basis.

MR. MICHELSON: Yeah, you can pass the buck on if you wish, although you shouldn't have to because you already know today, in most cases, which valves they are.

MR. CARROLL: You are saying you know which valves they are --

MR. MICHELSON: Yeah.

MR. CROM: That's correct.

MR. CARROLL: -- that the testing would be specific to the valve manufacturer.

MR. MICHELSON: Yeah; I don't mind that part. At least I thought, though, in this testing table I would have seen identified with a footnote or an asterisk or something that that particular valve will have to have more than just a normal breakopen --

MS. LI: This is Renee Li from Mechanical Engineering. The table 3.9-15, the intent is only to address the inservice testing and in the SAR. That's section 3.9 --

MR. MICHELSON: That is my point; it's inservice testing. Even for inservice testing now, you obviously can't break the pipe and do the test.

MS. LI: Right.

MR. MICHELSON: But you have to look at the test results during inservice testing differently than you do for a valve that doesn't have to open or isolate under blowdown conditions. And you somehow should flag in the table that this particular valve has got some unique requirements and you've got to go back to look at the prototypical test that was done back at the laboratory and then try to relate that to the test result you're getting to verify it will still work.

MS. LI: I agree with that. I am trying to point out that in the SAR, under the design and the qualification for pumps, MOV check valve, POV, they have detailed description of how a valve or a pump should be qualified. And in there, the manufacturer will perform ranges of different differential pressure and those --

MR. MICHELSON: That's the one that I was trying to find. Maybe you could tell me what section to read.

MS. LI: Okay.

MR. MICHELSON: I've probably just had too much to read.

MS. LI: For pumps --

MR. CARROLL: No; valves.

MR. MICHELSON: No, let's just do valves.

MS. LI: Okay; for valves --

MR. CROM: It's in several sections. There's one --

MS. LI: The format is MOV, check valve, POV. So if you look at --

MR. MICHELSON: Well, let's just do the MOV just as an example.

MS. LI: Okay. MOV, 3.9-6.2.1.1.

MR. MICHELSON: 9-6.2.1.1; okay. Okay. I'll read that and see if that does what I need.

MS. LI: Okay. And then there will be similar sections for check valves and other types.

MR. MICHELSON: Yeah; right. But this problem is kind of unique to the isolation --

MR. CROM: Yeah; okay.

MR. MICHELSON: Thank you.

MR. CROM: Going on, we also list the -- give the test configuration. Again, all this to ensure that the valves are testable, we can do the test, and also the CESSAR-DC figure the valve is located on.

I will note here that through our program and plan that we had -- took no exemptions to the code. We went through -- and that was one of the main reasons to do it, ensure the design did not require any exemptions to the code.

Some of the design provisions we include in the design to aid in IST, we include full flow testing for all safety-related pumps, that they can be tested quarterly; we have the capability to measure NPSH throughout the pump operating range. When I talk about redundance and separation of systems, a lot of current plants may have a common test line, recirc line or something, that the way the configuration of the system is, they can't test it quarterly because they put both divisions out simultaneously. We do not have any situations like that in this plant so that we are able to not take any exemptions on testing pumps, that we can't test them all quarterly.

MR. MICHELSON: That is under full flow condition?

MR. CROM: That's --

MR. MICHELSON: Or design flow condition.

MR. CROM: That's correct. We have full flow conditions -- have the capability of testing all pumps at full flow; all safety-related pumps.

As far as testing safety-related valves, again, we have ensured that we do not have to take any exemptions to the code and we have the capability to test all valves. We also, again, are able to test a large majority of valves quarterly because of our redundancy in separation and the system design, such that we do not have to test as many valves as current plants at cold shutdown and refueling. And there still is valves -- I mean, you can't test your RCS pressure isolation valves quarterly, and there isn't a way that you're going to be able to design that to do that.

MR. CARROLL: What progress is being made to update the inservice testing of valves by ASME to reflect

89-10 kind of considerations?

MR. CROM: I'm not sure I can answer that. Todd, do you know the answer to that?

[No response]

MR. CARROLL: All right.

MR. CROM: I'm now going to go through the various COL applicant items that we do have in CESSAR-DC. First of all, the actual IST -- full IST program is a COL action item. We have provided to him considerably the starting point, what he needs, and the table, based on the frequency and the type of test that had to be done for all safety-related valves and pumps. But he still has to develop, you know, his test procedures, his test schedules. Of course, the test frequencies are pretty well already set for him from what we have done in CESSAR-DC and also has to determine his baseline preservice test program, which we're going to talk a little bit more on some of the other COL action items.

This first one is for pumps and this is pretty much, Mr. Michelson, what you were talking about on valves if the COL applicant establishes the baseline pump design qualification. Basically all the flow head NPSH requirements, speed, vibration, that the manufacturer has to qualify the particular pump to, and also then is required to have the manufacturer do the type test on each size, type and model of pump.

The COL applicant also has to ensure that the pumps specified is not susceptible to minimum flow or inadequate thrust bearing capacity, ensuring pretty much the min-flow problems we've seen on some of the pumps. He has to ensure that the pump that's actually procured does not have a problem with min-flow, that we have adequate recirculation flow.

COL applicant has to develop the pump -- this assembly program for all safety-related pumps based on the historical pump performance, the pump components performance in non-intrusive test results. That is a new item that was in the SECY letters, looking at pump dis-assembly. And that is something that the COL applicant will have to do based on his performance.

COL applicant items that we have for MOVs and POVs -- Mr. Michelson, this is the one that I think you were looking for, that the COL applicant establishes, again, the baseline valve design qualification testing, determine the fluid flow, the differential pressure, including pipe break, system pressure, fluid temperature, ambient temperature, minimum voltage/pneumatic or hydraulic pressure and minimum and maximum stroke time requirements. He also would have the action to ensure the manufacturer does the appropriate type test for each size, type, and manufacture, and he also will have to then relate that to his pre-operational testing and what he has to do on each of the MOVs and POVs.

MR. MICHELSON: Is it clear to you from the SAR that the COL holder will have to look at even low energy lines from the viewpoint of the ability of valves to isolate? Because low energy lines have valves that, if you should rupture the pipe, it's a much smaller rupture situation of course, but they still have to operate under those conditions. And depending on the valve and the size of the operator, it may or may not. Generally, it's not a problem, but it's been found to be a problem already when people start looking closely.

MR. CROM: We have another -- when we get to flood, we have another COL applicant item, you know, that basically, when it comes to moderate energy line breaks or even with high energy line breaks, since we're not fixing the pipe routing, you know, for the certification stage, that the COL applicant has to ensure, you know, from a flood standpoint or an interaction standpoint, that he can isolate those portions of the system that need to be.

MR. MICHELSON: When looking at the possible break sizes, of course traditionally we look at the pipes which at low energy end aren't really too much of a problem. However, you have to look at heat exchanges where you can have a serious problem because two ruptures can be far bigger leaks sometimes than a critical crack.

MR. CROM: One other thing I want to demonstr -- was going to show, I think, in my flood presentation, is there may not be a need to isolate.

MR. MICHELSON: Yeah; if you can also show -- yeah. It all depends upon whether there's a need or not.

MR. CROM: Yes.

MR. MICHELSON: Clearly; yes. And the COL holder is going to do that. Now, somewhere there is a prescription that says he's going to do that.

MR. CROM: Yes. That's in the flood section.

MR. MICHELSON: And that's in the flood section.

MR. CROM: Yes.

MR. MICHELSON: That's in Chapter 9 or --

MR. CROM: That's in Chapter 3.

MR. MICHELSON: Chapter 3.

MR. CROM: Chapter 3, Section 3.4.

MR. MICHELSON: 3.4; okay. That's the one I read. Thank you.

MR. CROM: Continuing with MOVs and POVs, the COL applicant ensures that the MOV specified for each applicant is not susceptible to pressure locking and thermal binding, and the COL applicant will --

MR. CARROLL: That's easy to say. How does he do that?

MR. CROM: That would have to be done during the type test by the manufacturer and the layouts of the piping. You have to ensure that they were not -- based on past problems, ensure that it doesn't have that type of problem.

MR. MICHELSON: There was a day that people when people that understood this problem, that was 30 years ago, had bonnet leakoffs. Is the bonnet leakoff still there and they don't use it, or are they now going back and using it? You used to have a little --

MR. CROM: Right.

MR. MICHELSON: -- on the bonnet, leakoff the pressure to whichever side you wanted it to leak to.

MR. CROM: Uh-huh.

MR. MICHELSON: But those things people got tired of and they just kind of capped them off and I suspect the manufacturers might even be leaving them off now.

MR. CROM: Yeah; most of them are.

MR. MICHELSON: Yeah. But that's what solved that problem when they knew about it. It took 30 years to rediscover it.

MR. CROM: Okay. Finally we have a COL applicant who will periodically test the MOVs per Generic Letter 89-10 paragraphs D and J and POVs to demonstrate continuing capability for design basic conditions.

MR. MICHELSON: When you say the 89-10, are you talking about the latest supplements as well? Does that kind of go without saying, or --

MR. CROM: Yes.

MR. MICHELSON: Because there have been supplements all the way up until very recently.

MR. CROM: That's correct.

MR. MICHELSON: Supplement 6 I guess is the last one out right now.

MR. CARROLL: Which struck me as a very prescriptive supplement.

MR. MICHELSON: Well, you've got to -- eventually you get prescriptive because that's one way people finally understand the message. I think the industry generally agreed with it. It was just -- so everybody agreed, they made it more prescriptive.

MR. CROM: Okay. For check valves, we have a COL app can establish again the baseline design qualification testing. Basically here he has to establish the required operating cycles to be experienced by the valve, numbers of operating cycles, duration; the severe transient loadings, again, pipe break and waterhammer; sealing and leakage requirements; operating medium temperature and gradients, and vibratory loadings. Those types of things. And then also ensure that the type test is done with the valve.

MR. CARROLL: What's the significance of vibratory loadings? What does that mean?

MR. CROM: I think basically vibratory loadings on the pump discharge.

MR. CARROLL: We're talking check valves.

MR. CROM: Check valves.

MR. MICHELSON: It's a flooder.

MR. CROM: Yeah.

MR. CARROLL: Okay.

MR. MICHELSON: One of the problems on high energy lines is if you break a line upstream of a check valve it experiences a sudden and violent reversal of flow. Is there any requirement that the designer now verify that what he bought will stay in place and act as an isolation valve when you experience this sudden violent reversal of flow?

MR. CROM: One is in the IST program on check valves. You have to do a reverse flood test.

MR. MICHELSON: It won't tell you, I don't think -- no. But you don't do it under break conditions.

MR. CROM: Right.

MR. MICHELSON: You just do it --

MR. CROM: That would only be done --

MR. MICHELSON: That's not a laboratory testing of these under break conditions, that I'm aware of, at least.

MR. CARROLL: That looks like the second bulletin.

MR. CROM: But that would be the type test. That would have to be done in the shop, but then you would do a reverse flow test.

MR. MICHELSON: Is there a reverse flow test required for check valves under break conditions?

MR. CROM: No. There would be a -- if the check valve is required to withstand a break condition, there would have to be a type test done in the shop.

MR. MICHELSON: Is that said somewhere in Chapter 3?

MR. CROM: Yes. That would be under --

MR. CARROLL: Isn't that under the second bullet?

MR. CROM: That's the --

MR. MICHELSON: Well, no, I don't think -- it depends on how you interpret -- do they have in mind on pipe break the fact that if you've got a high energy pipe and you break upstream of the check valve, there's a sudden violent reversal flow. All you've got to do is stress analysis or a test.

MR. CROM: Todd, you want to handle this?

MR. OSWALD: This is Todd Oswald, Duke Engineering. In Section 3.9-6.2.3.1 there is a statement about the type testing. These design conditions include all required operating systems experience, environmental conditions, several transient loadings expected during the life of the valve such as waterhammer or pipe break.

MR. MICHELSON: But people don't normally think about that check valve under this reverse flow dynamics and -- but I guess we can interpret it to be included there, even though not said. You should have just said flow reversal, and everybody could pick up on it right away then.

MR. CARROLL: I think pipe break is the clue there.

MR. CROM: Yeah.

MR. CARROLL: I think pipe break --

MR. CROM: The only way the check valve can hold is on a reverse flow.

MR. MICHELSON: Yeah; that's right. Well, no. Check valves also experience violent operating conditions even in the forward direction --

MR. CROM: Oh, I agree.

MR. MICHELSON: -- and you have the blowdown flow through them and you don't want to tear them up then, either, because -- depending on what their subsequent function might have to be. But the reverse flow would be a nice word to throw in there to make sure there's no misunderstanding. But we'll assume it's understood.

MR. CROM: Okay.

MR. MICHELSON: 3.9-6 you said?

MR. CROM: Todd, can you just repeat that section?

MR. OSWALD: That's 3.9-6.2.3.1.

MR. MICHELSON: 2.3.1; thank you.

MR. CROM: We also have a COL applicant ensures that check valve application is the proper size, type, location, operation, as recommended by the manufacturer, since there has been a series of problems with check valves that are in the wrong orientation as recommended by the manufacturer. We have a COL applicant item to ensure that they are installed and the proper check valve for that particular applicant.

MR. MICHELSON: Is there a requirement to put the marker on the valve body yet?

MR. CROM: We do not state that in the SAR.

MR. MICHELSON: So that's the only -- that's one good way -- fairly good way, at least, of checking.

MR. CROM: Yes.

MR. MICHELSON: Not all valve bodies have the arrows on them.

MR. LINDBLAD: Or steam traps.

MR. MICHELSON: Yeah.

MR. CROM: We also have a COL applicant ensures the capability of nonintrusive testing, measurable flow through check valve. Again, this is in addition to the code, to the SECY letters that would be able to do nonintrusive testing.

MR. CARROLL: You are saying every safety-related check valve --

MR. CROM: No. Not every one. The ones that the COL applicant puts in his program as being critical.

MR. CARROLL: Okay. I wouldn't infer that from what you have on the slide.

MR. CROM: I believe the SAR has further explanation.

MR. DAVIS: There was an incident one time where an applicant or a plant operator was testing the two-train containment spray system and discovered that one of the check valves had been installed backwards.

MR. CROM: Uh-huh.

MR. DAVIS: They were using compressed air, which is the typical way to do it.

MR. CROM: Right.

MR. DAVIS: So they sent a crew down to turn the valve around, and they turned the wrong valve around. And so they ended up with no operable system.

MR. CROM: Yeah.

MR. DAVIS: And I frequently use this as an example where redundancy really hurt, because if they'd only had one system, they would have got the right valve.

MR. CROM: Right.

MR. DAVIS: But the point is that there were no arrows on these valves either, so this was a mistake that was made because they couldn't tell.

MR. CROM: Uh-huh.

MR. DAVIS: And it seems to me like this illustrates the fact that it would sure be advantageous to indicate on the valve which way the flow is going. I personally would like to see that kind of requirement. It would prevent this kind of a problem.

MR. CROM: Todd?

MR. OSWALD: This is Todd Oswald, Duke Engineering Services. Just one comment; when you're down there looking at the pipe, you've got to know which direction the flow is coming from. I mean, even if you have an arrow on the valve -- I mean, there's no question about it -- it's a good idea to have that. But you've also got to realize where your flow is coming from and going to when you're down there.

MR. DAVIS: We'll put an arrow on the pipe, too.

MR. OSWALD: Okay.

MR. CROM: By the way, Todd, even though he is our structural expert, he's done inservice testing at McGuire for several years and he's got a lot of good answers on that too.

MR. CARROLL: The other thing that needs arrows are flow leveling orifices.

MR. CROM: Yes.

MR. CARROLL: That has bit people more than once.

MR. CROM: I understand.

MR. CARROLL: Me too. I've been bit.

MR. CROM: And finally, for valves, we have a COL applicant that he develop the disassembly program for all safety-related valves, again based on the historical valve performance, the valve components' performance, and non-intrusive test results. That is everything I have, unless there's any further questions.

MR. LINDBLAD: I have a question or two.

MR. CROM: Sure.

MR. LINDBLAD: Mr. Crom, does the program require that the inservice testing be done before the mechanic touches the valve?

MR. CROM: No; I don't believe it does.

MR. LINDBLAD: So there isn't a requirement that as-found conditions be used?

MR. OSWALD: Can I address that? This is Todd Oswald, Duke Engineering Services. You have to get your as-found condition particularly on Appendix J-type valves. There is a requirement to have the as-found leakage. I don't think that we have specifically required the -- such as MOV that's not an Appendix J valve.

MR. LINDBLAD: Yeah. But there is a natural competition between the engineer wanting functionality of the pump and the valve to have a loose gland and the housekeeper who goes around and tightens up the glands.

MR. OSWALD: Uh-huh.

MR. LINDBLAD: How is that prevented in the IST program?

MR. CROM: I'm not sure I've got an answer. Do you know, Todd?

MR. OSWALD: I'm not sure I understand the question exactly. You mentioned -- you're talking about someone just arbitrarily going out and touching the valve?

MR. LINDBLAD: No. I guess I'm talking about -- I didn't see in the program that there is a requirement to -- in the touch program to be satisfied, that the gland does not leak under service conditions. The stem packing of either the pump or the valve. And so if you want to be sure you pass the test, you have a loose packing to get the valve stroking at the best time, and yet you don't want puddles on the floor, and so you want the tight packing then. But --

MR. OSWALD: I think that would be more appropriately addressed in the maintenance program on your valves. When you have your work request or whatever maintenance program you use, you have that valve identified and you don't do your inservice testing on it until after all the maintenance is done. Your QA and inservice testing is one of the last steps in your maintenance program.

MR. LINDBLAD: Yes. That's what I've seen, too.

But then --

MR. CARROLL: And then the mechanic comes back --

MR. LINDBLAD: -- comes in to service --

MR. CARROLL: -- takes another half turn on the packing valve.

MR. OSWALD: Well, he's wrong.

MR. CARROLL: But he is real.

MR. LINDBLAD: Yes.

MR. CARROLL: The rule is that anytime you do maintenance, you've got to repeat the testing, Bill I'm not sure that always happens in practice, but that's what you're supposed to do. And a classic example is packing.

MR. LINDBLAD: Yeah.

MR. MICHELSON: The biggest problem is, though, these valves just aren't that reproducible either. You can cycle them a few times, they'll work one way, if you leave them set for 30 days, they'll work quite differently. There's a range of uncertainty there that kind of overshadows much of this. It just -- I think you just have to put a lot of conservatism into the selection and adjustment of the valves and hope for the best, because you won't get it by precision.

MR. OSWALD: This is Todd Oswald, again; Duke Engineering. I guess that's why we have the quarterly testing, is to catch that.

MR. LINDBLAD: I was just writing a note to Mr. Coe a minute ago that he distributed an operation occurrence where a steam generator recently went dry because the MSIB had been adjusted cold when -- and tested cold, when it should have been adjusted hot and tested hot.

MR. DAVIS: This was McGuire?

MR. LINDBLAD: Yeah.

MR. CROM: I'm quite familiar with that one.

MR. LINDBLAD: These are the same people.

MR. MICHELSON: But it also makes a difference in how many cycles and what your history is just before you do your final calibration test on a motor operated valve, too.

MR. CROM: But again, that is the type of thing that, you know -- for example, the McGuire instance, it's not something that we can specify in the SAR now because that's a manufacturer's recommendation and it depends on what the manufacturer tells you, whether it should be done cold or whether it should be done high. And that would have to be put in the COL applicant procedures for adjusting that particular valve. McGuire was testing it inappropriately. The manufacturer said it should be tested hot and they were testing it cold, and it did not shut under hot conditions.

MR. MICHELSON: If you did all your testing like the manufacturers say you'd be in deep trouble, too, as we've found out.

MR. CROM: I agree.

MR. MICHELSON: So ultimately the owner has to understand what he's doing and do it right.

MR. DAVIS: How many valves are covered by this program?

MR. CROM: I don't know if I've done a count, ever. It's about -- that table is about that thick and the first page is pumps and the rest is valves. I don't know if I've ever done a count.

MR. CARROLL: Carl's got the table if you want to count, Pete.

MR. MICHELSON: It's a long table. Would you like to look at it?

MR. DAVIS: I'm not sure I can count that far.

MR. CARROLL: You can count the number on a page and multiply by the number of pages.

MR. MICHELSON: The critical valves are fairly small in number, probably of the order -- less than a hundred, perhaps, depending on --

MR. CROM: Yes.

MR. DAVIS: Can one crew do all of the valves quarterly?

MR. CROM: Todd, can you answer that since you --

MR. OSWALD: Yes. Those shouldn't be any problem. Well, when you say one crew, you have a test group normally that does it, which consists of probably about 10 technicians or so, and a staff of engineers.

One other thing; quick glancing, looks like there's about 80 pages and I counted one page had about 10 valves -- of course one page has more than that. So it looks like it would be in the neighborhood of 700, 800 valves in the program.

MR. MICHELSON: But not all of those gets this awfully detailed kind of examination.

MR. CROM: Yes. Some of them are just basic.

MR. MICHELSON: Most of them are just push the button and watch it go up and down.

MR. CROM: Yes. Some are just the stroke test and some are just checking it's in the proper position.

MR. MICHELSON: Yes. But a few of them are very critical.

MR. CROM: Okay. Are you ready for me to go on to flood?

MR. CARROLL: Are you ready?

[Slide.]

MR. CROM: Yeah; I'm ready. The first G slides that I have in the package we really covered yesterday. Lyle Girdes covered pretty much on external floods. So I'm going to skip over those and go right into our internal flood slides.

MR. MICHELSON: Before you do that, I had a couple of questions. I never manage to get back with the people to show me where these tunnels are. You've got a big tunnel going from the point of cooling water building over to the nuclear island, and you probably have other tunnels.

MR. CROM: Todd's going to pull out the --

MR. MICHELSON: Including the intake structure. There's -- the RSW has, someway, a tunnel going from there over to the nuclear island. Couldn't find much detail on these, but these are all potentially worrisome sources of water, particularly in the case of the component cooling water or the -- in the case of the service water; you don't want to have a pipebreak start draining the pond back into the --

MR. CROM: Well, we're going to cover that --

MR. MICHELSON: You're going to cover that now? Okay --

MR. CROM: -- and that's going to be a non-problem for this plant.

MR. MICHELSON: Okay, now -- but on the site floods, it's the same problem again. You've got to keep the water out of all these buildings and all these tunnels, unless you provide some other means of keeping them from getting burst.

MR. CROM: That's correct.

MR. MICHELSON: And get you in trouble.

MR. CROM: Steve, do you want to --

MR. STAMM: Yes. This is Steve Stamm, Stone Webster. As far as component cooling water type tunnel, the elevations are showing that the connections of the tunnel on the building drawings for the nuclear annex and the component cooling water --

MR. MICHELSON: For which one? Which one?

MR. STAMM: The figures are Figure 1.2-2 shows the connection to the annex, at plus one feet for the top of the tunnel.

MR. MICHELSON: Okay.

MR. STAMM: Figure 1.2-25 shows the connection to the component cooling water building at 3.5 feet below grade for the top of the tunnel.

MR. MICHELSON: What was the elevation then in the nuclear island?

MR. STAMM: 91-9. The elevation of the actual --

MR. MICHELSON: Part of the tunnel then is a vertical --

MR. STAMM: It comes a vertical pipe and then into the tunnel.

MR. MICHELSON: -- going downward.

MR. CROM: That's correct.

MR. MICHELSON: So that if you have a site flood, if it does get into the tunnel, then it goes all the way down and produces hydrostatic pressure at elevation. What was that again?

MR. OSWALD: Delta 70.

MR. MICHELSON: 70 is about grade, isn't it?

MR. OSWALD: No. 91-9.

MR. STAMM: 91-9 is the top.

MR. MICHELSON: So that's only about 11, 12 feet then?

MR. STAMM: Yes. Although not explicitly stated, and we talked about adding some words so that it makes it very clear, the penetration seals and the seals for the tunnels, we considered structural components and are covered by the design criteria in 3.8a, which covers hydrostatic pressure.

MR. MICHELSON: Yeah. That's okay. That's an important part of it. Now, the next question is if we were to experience a seal failure and you have no good way of testing these through the life of the plant very easily, you can inspect them; that's about the best you can do; but you have to design some kind of a limited leakage seal such as if you get a catastrophic failure during a flood you don't fill the whole building because there's almost an infinite source of water available to fill the building.

MR. CROM: Let me address the station service water before you --

MR. MICHELSON: You've got to design it so you can do damage control while you're trying to stop the leak.

MR. STAMM: The source of flooding in the component cooling water tunnels is only component cooling water plus ex --

MR. MICHELSON: No, no, no. This is site flood I was talking about.

MR. STAMM: Oh; from site flood.

MR. MICHELSON: If site flood gets into the tunnel

MR. CROM: Okay. Tom is going to --

MR. MICHELSON: Now you've got an infinite source of water to flood the building with.

MR. STAMM: Tom's going to cover that --

MR. MICHELSON: Site floods are pretty large.

MR. CROM: Let me understand you. When you say site flood --

MR. MICHELSON: Yes; outside.

MR. CROM: Okay. From outside.

MR. MICHELSON: Yes. The one that comes just one foot below grade.

MR. CROM: Right. Now, Steve, isn't there a requirement that the tunnel be watertight?

MR. MICHELSON: Has to be; yeah.

MR. CROM: Yeah. So that the water is not going to -- if the --

MR. MICHELSON: This is where you get into the question of when do you discount the single failure criteria during the --

MR. CROM: Well, the other thing -- well --

MR. MICHELSON: -- do we have a failure of one of our seals, and if we do, how do we control the damage before we fill the whole building with water?

MR. CROM: Let's look at the internal flood and I think that maybe --

MR. MICHELSON: This is not an internal flood. This is an external flood that gets in internally.

MR. CROM: Well, what I'm saying is, what I think is if you look at our internal flood protection, including the safety-related pumps --

MR. MICHELSON: Well, if you look at it internally you don't see it because the water isn't coming from a source internally. You usually have limited sources and you show how you handle all that, and that's good.

MR. CROM: Yeah.

MR. MICHELSON: But this is an infinite source of water external to the building that comes in through a hole.

MR. CROM: Uh-huh.

MR. MICHELSON: And now, how do you plug the hole?

Well, the best thing to do is make sure you set a limited leakage seal such that you show you can do a damage control and get to the seal and do a damage control on it before it floods the building out.

MR. CROM: Right.

MR. MICHELSON: And that's --

MR. CROM: But that can be addressed -- what I'm saying is, if it is an external flood, and it's going internally into the building --

MR. MICHELSON: This is --

MR. CROM: -- that safety-related sump pump in level indication is going to detect it.

MR. MICHELSON: It depends on the rate of leakage of this seal --

MR. GUO: This is Jim Guo of --

MR. MICHELSON: -- and that's why they call it a limited leakage seal, so that you can handle it.

MR. GUO: I think the external flood has no way to go into the building because all the grade level below one foot from the flood level have no entrance, so there's no way to --

MR. MICHELSON: I'm sorry. I was just pointed out this vertical shaft here drops down, what, 10, 11 feet and then goes into the building. So you're 11 feet below the flood.

MR. GUO: Even if that's so, they have the seal at the --

MR. CROM: Well, that's what he's talking about. The failure of the seal.

MR. MICHELSON: That's what I'm talking about. All I'm saying, it has to be a limited leakage seal. Has to be -- and you can design seals so that even when the elastomers fail, the metal portions limit the leakage to a rate that you can handle, either with sump pumps or you can handle with damage control or something. I didn't find that requirement.

MR. CROM: Okay. I think we -- I understand what you're saying and Steve, do you think there needs to be -- we need to look at it, whether there needs to be any --

MR. STAMM: Todd just pointed out, you're absolutely right. The seal is designed so that we wouldn't expect a catastrophic failure. The other point that was just made to me was that the component cooling water seals are designed to prevent leakage in the event of a component cooling water pipe rupture in there, which would be much greater than we would get from an in-leakage, from an external in-leakage during a flood.

MR. MICHELSON: No, the source is pretty small. You've only got a few hundred feet of piping at most, and that's all the water --

MR. STAMM: Agreed. But the rate --

MR. MICHELSON: Sure you dump the infinite source of the site flood. You fill the building. You can't stop.

MR. STAMM: I was talking about rate as opposed to the total quantity.

MR. MICHELSON: Yeah; the rate would be greater in that case, but the damage control would be easier because of the very limited amount of water that you release. Here it's coming forever.

MR. STAMM: The other factor that needs to be considered and Tom probably is going to point this out, is the plant separation shows that small leakage, even significant leakage from external flood, will not affect both trains of the plant.

MR. MICHELSON: If seals don't -- if only one seal fails.

MR. STAMM: If only one seal fails; that's correct.

MR. MICHELSON: Does this tell me you've got limited leakage seals or not? You know, if you don't specify it as being limited leakage and limited to some value or one value is limited to the leak rate of the capability of the sump pumps.

MR. GUO: This is Jin Guo again. Even if --

MR. MICHELSON: It's not hard to do, by the way.

MR. STAMM: I understand.

MR. GUO: In case of leakage inside the reactivating subsphere, they have a sump and a pump that monitors 24 hours and they --

MR. MICHELSON: We're not even questioning that feature. We're questioning only what happens when the flood water gets into the building through a single failed seal.

MR. GUO: In case -- entrance of -- the seal fails, the water goes in the building, then you have a monitoring system and --

MR. MICHELSON: Right; right. Now, how fast does the water come into the building? How much is the leakage limited by the seal design?

MR. GUO: In any case, the leakage will not -- the leakage is not a big flood goes in.

MR. MICHELSON: Well, you better take a look at how people design seals. I've seen some seals that can leak literally hundreds of gallons a minute, and I've seen seals that only leak a few gallons a minute even after the elastomers fail.

MR. GUO: This leakage are monitoring 24 hours a day.

MR. MICHELSON: To monitor them -- that isn't the question.

MR. RITTERBUSCH: This is Stan Ritterbusch. Mr.

Michelson, we'll take a look at our statements in the SAR and make sure they say what we want them to say.

MR. MICHELSON: The nice thing to say is that the leakage is limited to the capacity of the sump pumps. That would be the nice -- I don't know if you can say that or not. In the case of GE, they couldn't quite say that. But that would be the ideal, is it's limited -- design a seal that limits the leakage upon a failure of the elastomers, limits the leakage to the capacity of the sump pumps; and you've got it made.

MR. CROM: Okay.

MR. CARROLL: All right. Let's move on.

MR. CROM: I'm going to start on the internal flood protection.

[Slide.]

MR. CROM: The first point I want to make is that station service water is totally located outside of the nuclear annex. The only place that station service water enters it into the component cooling water heat exchanger structure and the component cooling water heat exchanger structure is designed such that if you have a moderate energy line break of service water in that structure, it will not flood through the tunnels back into the nuclear annex. There is flood protection there.

MR. MICHELSON: One small question and I couldn't find, again, the answer. That is the piping between the service water pumping station and the component cooling water building.

MR. CROM: Yes.

MR. MICHELSON: That appears to be underground piping.

MR. CROM: That's correct.

MR. MICHELSON: Is it in a chase or is it in the ground?

MR. CROM: Right now it's buried.

MR. MICHELSON: It looked like it was buried. I couldn't verify that.

MR. CROM: That's correct. It's buried.

MR. MICHELSON: And it's buried, I guess, at least ten feet or so deep. You don't have a turbine missile problem then.

MR. CROM: That's correct.

MR. MICHELSON: Telephone poles would go pretty deep if you dropped them.

MR. CROM: The other point I wanted to make is that component cooling water and emergency feedwater systems are fully separated by division. These are not the only systems. Basically, what I'm saying is there's no cross-connects between the systems.

I've pointed these two systems out because current plants, when they look at their moderate energy line break, have found that in component cooling water and emergency, they normally have -- some plants have an open cross-connect and show that they lose both divisions on some of the breaks that they don't want.

MR. MICHELSON: You didn't specify, of course, much about the service water pumping station, since that's outside the scope of the certificate.

MR. CROM: That's outside the scope.

MR. MICHELSON: But I didn't find any interface requirement that says it shall have a divisional wall up to grade. Did I miss it?

MR. CROM: That is in the service water section.

MR. MICHELSON: It says it shall be separated, but it didn't say there was a divisional wall up to grade that would keep the water out and limit the water to just one side, which is what you have in the service water.

MR. CROM: Steve, do you recall what we had?

MR. STAMM: No. But in all likelihood, the designs that we were looking at, we actually had separate structures for the two trains.

MR. MICHELSON: The present plan here only shows one structure. If there were two structures, I wouldn't have asked the question.

MR. STAMM: But, obviously, we need to do that.

MR. CROM: I will check, but I think the ITAAC, which has an interface requirement, says that there has to be a divisional wall between the two.

MR. MICHELSON: Don't put it in the ITAACs. We've been assured repeatedly by the staff there are no new design requirements in ITAACs. Every design requirement --

MR. CARROLL: Put it in both places.

MR. MICHELSON: Both places, but don't just put it in ITAACs.

MR. CROM: I agree. It is in both. Let me check where it is.

MR. MICHELSON: We'll look for it later after you get a chance to look at it. Thank you.

MR. CROM: The next one let me go through and then I'm going to show a figure to point them all out. Of course, the divisional wall is our primary means of flood control in the nuclear annex. I'm going to point out that there are no doors provided up to Elevation 70 in the divisional wall. Also, the wall that we have around the two diesel generator rooms, again, there is no door until Elevation 70.

We will note there is no pipe or duct penetrations or any penetrations to --

MR. MICHELSON: There will be a note that says there are no penetrations of the all. Is that what you're saying?

MR. CROM: No. Let me put it this way. Right now we have -- in our design, we have no penetration through the wall. The pipes that we have routed, the duct we have routed, there is no penetration. In the SAR, we say if the COL later design puts a penetration through that wall, it has to be appropriately sealed and, your point, it would have to be qualified for the hydrostatic loading.

MR. MICHELSON: And/or pipe break loadings that go higher.

MR. CROM: That's correct. The reactor building subsphere is also separated in quadrants and those are also designated as flood barriers. There is a door in those walls, which is a flood door.

[Slide.]

MR. CROM: I will answer your question you had yesterday. We arbitrarily selected Elevation 70, but I've got a later slide that talks about analysis that demonstrates that that is an acceptable level.

MR. MICHELSON: Elevation 70 is what?

MR. CROM: Elevation 50 is, of course, the basement level. Elevation 70 is the first level above the -- or is the next level up.

MR. MICHELSON: It's 20 feet up from that.

MR. CROM: It's 20 feet, yes.

MR. MICHELSON: But that's not to grade.

MR. CROM: No, it is not to grade.

MR. CARROLL: My question was what's magic about Elevation 70.

MR. CROM: It was arbitrarily chosen, but we have done analysis which demonstrates that essentially we can take a break in every line, moderate energy line, simultaneously and will not flood above Elevation 70. We have flood barriers to provide separation between the electrical equipment and mechanical systems at the Elevation 50, the lowest elevation in the nuclear annex.

We also, around each of the emergency feedwater pump rooms, which is on Elevation 50, are compartmentalized and also have flood barriers around those.

MR. MICHELSON: Are you going to tell us later how you get the steam line down to the steam-driven feedwater?

MR. CROM: Steam line goes through a pipe tunnel from the main steam valve house down into the turbine-driven pump room.

MR. MICHELSON: I could find the tunnel on the drawings, but then the tunnel suddenly disappeared and I couldn't tell how you got from quite a ways away yet over to the room. In fact, you had to go somehow across the corridor to do it.

MR. CROM: It probably does not show up in detail.

Todd, do you want to answer that question?

MR. OSWALD: Todd Oswald, Nuclear Engineering. It crosses the corridor up at the main steam valve house at Elevation 106. It goes back down and the straight down into the top of the --

MR. MICHELSON: That's not he way I read the drawing. That was a problem, because those auxiliary feedwaters are out near the edge underneath the containment and the chase is quite a bit further in. This is that chase right there.

This is the chase that you're referring to, I assume.

MR. OSWALD: Yes.

MR. MICHELSON: It's not far down. Here's the room.

MR. OSWALD: No, that's the tanks.

MR. MICHELSON: Yes. That's the tanks, but the room is directly above the tanks.

MR. OSWALD: No. The main steam valve house is --

MR. MICHELSON: Let's just get the room.

MR. OSWALD: The room is right here.

MR. MICHELSON: Here it is.

MR. OSWALD: So your tunnel comes down right here.

MR. MICHELSON: You're staying there and it's not far to go. So that's also the vent shaft for the pressure.

MR. CROM: That's correct.

MR. MICHELSON: Have you done the pressure analysis on that room somewhere in the SAR?

MR. CROM: The pressure analysis -- Todd, that was done as part of the structural design, was it not? And it was also included in the equipment qualification in 311.

MR. MICHELSON: But is it written up somewhere in the codes you're going to use for the pressure analysis and the results?

MR. CROM: The person who did that is not here and I don't know the answer. Do you now, Todd?

MR. OSWALD: No. I'm not sure what code they used.

MR. MICHELSON: Could you get us the answer? If it isn't in the SAR, then can you send us a write-up on how you did your calculations, what code you used and what results you got?

MR. CROM: I'll ask CE. They were the ones that did it.

MR. OSWALD: I know what the result was. It was ten psi.

MR. MICHELSON: It sounds like it could be even higher than that, depending on which code you're using and what assumptions you're making. That's what we want to see.

These are very high pressures you get. So we have to ask the design of the doors to make sure that it really does vent the pressure up through the shaft and the whole pathway all the way to atmosphere. That I couldn't find anywhere. It becomes an important calculation and I'd like to see the results. I'm sure Ivan would, also, since he looked at them for ABWR very carefully.

After we look at them long enough and carefully enough, we finally realize what really happens.

MR. CROM: I think Mr. Mitchell was the one in your shop that performed that.

MR. MICHELSON: Is that a six-inch on this turbine or a four?

MR. CROM: Six-inch.

MR. MICHELSON: Six-inch. That's a substantial line, then.

MR. CARROLL: How do we resolve this?

MR. RITTERBUSCH: We're going to caucus here and we're going to have an approach to how to fix this before the end of the day.

MR. CARROLL: Sounds good.

MR. MICHELSON: Sounds good.

MR. CROM: Steve, do you have something?

MR. STAMM: Yes, if I could go back for a second. Steve Stamm. Section 9.2.1.1.4, Item B, says the "SSWS pump structure shall provide physical barriers to maintain divisional separation of SSWS components."

MR. MICHELSON: That I already knew. Was that a question I had?

MR. CARROLL: You asked about the separation of the station service water.

MR. MICHELSON: I knew about it. It has a divisional wall right down the middle. The fact is it's not even divisional. Oh, the service water. I'm sorry.

MR. STAMM: Yes, the service water.

MR. MICHELSON: I'm thinking of the -- I'm sorry. The service water, then, does specify a divisional wall to grade.

MR. CARROLL: It implies that, I think.

MR. STAMM: It definitely implies that.

MR. MICHELSON: That was the requirement and I will find that.

[Slide.]

MR. CROM: I just wanted to throw up a figure here. I've had the word slides. When we talk about the divisional wall on Elevation 50, we're talking about this wall, all the way across. The quadrant wall is here and, as I said, there are two doors here and here which are flood doors. Each of the motor-driven pump rooms and the turbine-driven pump rooms, and, again, there are doors entering each of those rooms which are flood doors.

MR. CARROLL: Tell me about a flood door.

MR. CROM: I'm not sure I can answer that question. We're talking about what's in typical plants, the submarine type door for flood doors, yes.

MR. CARROLL: Are they self-closing?

MR. CROM: We do have those all sensored and alarmed in the control room.

MR. CARROLL: That was going to be my next question.

MR. MICHELSON: They are essentially, though, lugged down, aren't they?

MR. CROM: Yes.

MR. MICHELSON: You're talking about fairly large hydrostatic pressures.

MR. CROM: I don't think you can design a self-closing flood door, but we do have them sensored and alarmed in the control room.

MR. MICHELSON: You could, but --

MR. GUO: Jin Guo. The flood doors are pressure doors and they have sensors in a central fire station. They're monitored 24 hours a day. So it's guaranteed the doors close.

MR. CARROLL: There was a statement in the FSER that made it sound like there's an operator station monitoring this thing 24 hours a day. I don't believe that. I can find it. Did you read the same thing?

MR. MICHELSON: I don't recall that one, no. Clarify something for me. What you show here is the vertical divisional wall is under the subsphere.

MR. CROM: Yes.

MR. MICHELSON: In the subsphere area. It only goes that far, whereas the main divisional wall, the horizontal one, actually goes all the way up through the building to what elevation?

MR. CROM: The divisional wall and even the quadrant walls go --

MR. MICHELSON: Let's just talk about the divisional wall first.

MR. CROM: The divisional wall goes all the way up to the building, yes.

MR. MICHELSON: All the way to the topmost floor?

MR. CROM: Yes. We don't designate that as a flood barrier.

MR. MICHELSON: No, no.

MR. CROM: Up above Elevation 50. It goes all the way through.

MR. MICHELSON: It's identified as the divisional wall.

MR. CROM: That's correct.

MR. MICHELSON: Now, the subsphere division into four quadrants is only through the subsphere area.

MR. CROM: That's correct, because you have containment above that.

MR. MICHELSON: And above that you've got nothing.

MR. CROM: That's correct.

MR. MICHELSON: That's right.

MR. GUO: There is a statement in the ITAAC Table 2.1.1-1 that says the flood door shown in Figure 2.1.1-1 through 2.1.1-12 has a sensor with open and closed status at the central fire station.

MR. CROM: The central fire station, by the way, is in the control room.

MR. CARROLL: That's not what my point was. I'll find it.

MR. CROM: Okay.

MR. CARROLL: Move on.

MR. CROM: The only other one I had, of course, is the flood walls around each of the diesel generator rooms, which have no doors in this particular elevation. They enter actually in Elevation 70 with stairs going down.

MR. MICHELSON: In the drawings at Elevation 70, it just shows an open doorway into that compartment. There really must be doors on it or something, aren't there? I'm looking at, in particular, 1.2-5A.

MR. CROM: At Elevation 70?

MR. MICHELSON: At Elevation 70, yes.

MR. CROM: Are you talking about into the diesel generator rooms?

MR. MICHELSON: Yes.

MR. CROM: Yes. There are two doors on either end, required for life safety.

MR. MICHELSON: They're not shown on the drawing, is that the idea, or am I -- you do show doors on this drawing, but not in this case.

MR. CROM: There are doors shown on that.

MR. MICHELSON: I'm looking at it right there. That's a door, sure, but how about that? What's that?

MR. CROM: There's a door on each side. You have to have two doors for --

MR. MICHELSON: You're actually showing a door on each side and then kind of a door about a third of the way across the wall.

MR. CROM: There is an equipment door also on that elevation.

MR. MICHELSON: Yes, but it doesn't show a door there. That's all I was asking. Is that an open doorway or has it --

MR. CROM: No.

MR. MICHELSON: -- got doors on it? You couldn't tell from the drawing, that's for sure.

MR. LINDBLAD: Mr. Crom, could you put up the slide of Level 2?

MR. CROM: Level 2?

MR. LINDBLAD: The 70-foot elevation.

MR. CROM: The 70-foot elevation, sure.

[Slide.]

MR. LINDBLAD: It shows the remote shutdown room being directly below the control room.

MR. CROM: That's right here.

MR. LINDBLAD: Adjacent to a stairwell.

MR. CROM: Yes.

MR. LINDBLAD: Have you looked at that for firemen trying to put out a fire in the control room with water coming out of the control room? Will it cascade down the stairs into the shutdown room?

MR. DAVIS: The PRA says you can't have a fire in the control room.

MR. LINDBLAD: But you can still have firemen in the control room, can't you?

MR. DAVIS: Well, I'm not sure what they'd be doing there if there was not a fire.

MR. CROM: Fires in the control room are going to be put out with extinguishers in the control room. Control room fires are -- the fire suppression is manual extinguishers. The other thing is we do have three-hour barriers, which we'll talk about when we get to fire protection, around that stairwell.

MR. LINDBLAD: I guess I was talking about internal flooding rather than the fire, but I was concerned about whether the remote shutdown room is occupied -- can be occupied under all conditions when the control room maybe challenged.

MR. CROM: Yes.

MR. LINDBLAD: You say there's no way for water to come down the stairwell.

MR. CROM: We do not have any automatic suppression for the control room. It is only done by manual fire suppression with manual extinguishers. So I don't see that there's a water source that could flood it. I've got other slides to talk about control room and the flood provisions we have on that.

MR. LINDBLAD: Could you answer my question? Is there an interconnecting stairway from --

MR. CROM: There is an interconnecting stairway. You do exit the control room and you go down that stairway.

MR. LINDBLAD: And while the remote shutdown room is not in the very lowest level of the building, it's --

MR. CROM: It's in Elevation 70.

MR. LINDBLAD: -- adjacent to the stairwell.

MR. CROM: That's correct.

MR. LINDBLAD: Thank you.

MR. CARROLL: I found the statement that I think needs clarifying in the staff FSER. I am on Page 19-220. It's talking about the flood doors and the alarms on them. It says that they're provided at a central fire alarm station, and that should go on and say which is located in the control room.

MR. FRANOVICH: This is Mike Franovich. If that's the case, we'll go ahead and clarify the FSER.

MR. CARROLL: And it also goes on and says "CA stated that the flood door open/closed status will be continuously monitored and manned 24 hours a day." That implies that it's manned, that there's some guy sitting here looking at these indicator lights, and that's not the case.

MR. GUO: The flood door is monitored in the central fire alarms, not the control room.

MR. CARROLL: Didn't I just hear that that is in the control room?

MR. GUO: No.

MR. MICHELSON: That's what I heard.

MR. FRANOVICH: This is not to be confused with the central alarm station for security.

MR. CARROLL: No.

MR. CROM: That's correct. The central fire alarm station is a panel that is located in the control room. It is not on the main control room console, but it is in the control room.

MR. CARROLL: And is there an operator sitting in front of it?

MR. CROM: No. There are operators in the control room that would be alerted.

MR. CARROLL: I think the wording needs a little cleaning.

MR. FRANOVICH: We'll go ahead and clarify that.

MR. SEALE: Is the panel alarmed or are there annunciators on the panel to call attention to a change in status?

MR. CROM: Yes.

[Slide.]

MR. CROM: I'm going to continue. We've already talked about the sensors. Again, at higher elevations, electrical equipment is elevated above the floor such that flooding events will not affect components.

MR. MICHELSON: That's always assuming that water comes from the bottom up from the floor and then it won't affect the component, it's on a pedestal. But how about water coming from the top down to the floor and the equipment is in the way?

MR. CROM: Of course, when we talk about fire, of course, we talk about the electrical equipment, that being NEMA-qualified for water sprays and interaction. The other thing is, of course, we --

MR. MICHELSON: Be careful, now. You're going to qualify it for water spray or you're going to qualify it just for --

MR. CROM: For drip-proof.

MR. MICHELSON: Drip-proof. That's not a water spray, of course.

MR. CROM: The other thing, of course, with the complete divisional separation, we still have the other division.

MR. MICHELSON: That does help, yes.

[Slide.]

MR. CROM: As far as the floor drainage system, they're separated by divisions in quadrants. What I'm saying is that the drain lines are physically not connected to each other going to the sumps. In the quadrants, each quadrant has a separate sump with two safety-related sump pumps.

MR. MICHELSON: When you get the water out of the sump, where does it go to?

MR. CROM: It goes into the rad waste building.

MR. MICHELSON: Does it go in as separate pipes into the rad waste building from each of the two divisions or is it headered?

MR. CROM: Of course, once it reaches into the rad waste building, it would be into a common header and back-flow devices are in that particular line.

MR. MICHELSON: You've provided how many back-flow devices? How many devices have to fail in order for back-flow to get both sides?

MR. CROM: I'd have to look at the diagram. I know what you're saying.

MR. MICHELSON: Gravity is what is driving it, unless these lines are high enough in the building, of course, where they're headered so they can't get a back-flow from one side to the other. If it's at a lower elevation, the water just goes through the header and right on into the sumps of the other division.

I guess there will be some words that cautions the owner and requires some amount of surveillance of check valves or whatever you're using. Design is the best way to solve the problem, of course, but you could solve it with valves, if you had to.

MR. CROM: Yes.

MR. MICHELSON: But I just didn't find any of this, but I probably didn't know where to look.

MR. CROM: We'll look at that and address that.

MR. MICHELSON: That is an open question.

MR. CARROLL: Are those check valves part of your check valve testing program?

MR. CROM: Only the safety-related ones. We have safety-related check valves in each of the sumps, which is my next bullet there. Those are included in there. The reverse flow test.

MR. MICHELSON: So the check valves on the sumps, even though they're non-safety sumps, are going to be safety-related.

MR. CROM: They are safety-related sumps.

MR. MICHELSON: They are?

MR. CROM: In each of the quadrants in the subsphere, they have two safety-related sump pumps powered from the diesel generators.

MR. MICHELSON: Safety-related means the QA on the piping, the whole thing.

MR. CROM: Yes.

MR. MICHELSON: All the way back to rad waste?

MR. CROM: No.

MR. MICHELSON: How far?

MR. CROM: Only to -- it's the pressure boundary that --

MR. MICHELSON: Like the check valve or somewhere.

MR. CROM: Yes.

MR. MICHELSON: And from there on, is it

seismically qualified?

MR. CROM: No. My next bullet was that the safety Class 3 check valves were provided to prevent back-flow, and that is when the drain lines actually enter into the sump. Of course, we've already talked about each subsphere quadrant has a sump and there's two safety-related sump pumps that are powered from the diesel generators in each of those.

Relating to some of the interaction problems that plants have seen in the control room, no water lines are routed above or through the control room or the computer room. We have that requirement.

MR. DAVIS: Excuse me. Let me ask you, if I could, about he diesel generator rooms.

MR. CARROLL: No drinking fountain?

MR. CROM: In our design, we have all the break rooms, kitchens and everything outside of the main control room. We intentionally did that, even though we violated an EPRI requirement.

MR. CARROLL: I know you did.

MR. CROM: That was the reason for it.

MR. CARROLL: Pete, I'm sorry I interrupted.

MR. DAVIS: The diesel generators are protected by a pre-action water spray system.

MR. CROM: That's correct.

MR. DAVIS: And the pre-action valve is automatically opened on detection of smoke.

MR. CROM: That's correct. Then the nozzle itself has to open on the heat sensitive.

MR. DAVIS: Now, this system is not Seismic 1. That's right?

MR. CROM: The standpipes are seismic, the Seismic 1. The piping, from an interactions standpoint, are Seismic Category 2. There's a shutoff valve on the standpipe and it's Seismic Category 1 all through there. Then we have a requirement that all suppression lines, from an interaction standpoint, be Seismic Category 2.

MR. MICHELSON: That means they don't fall down on vit.l equipment, but they can dump their contents.

MR. CROM: That's correct.

MR. DAVIS: Let me lead you through a scenario here and see what's wrong with it. If you have a seismic event, frequently there's a lot of dust generated. I'm postulating that that would cause this pre-action valve to open.

MR. CROM: Yes.

MR. DAVIS: Based on the detection of aerosols.

MR. CROM: Yes.

MR. DAVIS: And then the piping fails because it's not Seismic Category 1. It can be stranded pipe, which doesn't --

MR. CROM: No. We require it all to be welded pipe. We addressed that in that one question and we consider that to be seismically rugged. That's the way it's addressed in the PRA.

MR. MICHELSON: Seismic Category 2 isn't rugged from the viewpoint of contents.

MR. CROM: In the PRAs and all of the IPEs, it's considered if it's welded to be seismically rugged.

MR. MICHELSON: You mean the PRA people think pipe is seismically qualified even when it isn't.

MR. CROM: That's correct.

MR. MICHELSON: I didn't realize that.

MR. CROM: That's the way current IPEs are being done in current plants.

MR. LINDBLAD: If you're in the diesel room, wouldn't it be two over one?

MR. MICHELSON: Only from the viewpoint of the pipe falling down.

MR. LINDBLAD: Yes.

MR. MICHELSON: But not from the viewpoint of dumping contents.

MR. DAVIS: I'm concerned about spray from this system.

MR. LINDBLAD: I understand that, but I --

MR. MICHELSON: The pre-action valve is already open and now you've got a crack in the pipe where a joint broke or whatever and he's dumping the contents and it continues to dump it until you perform some kind of an isolation, which takes a while.

MR. CROM: Dave, do you want to address that?

MR. FINNICUM: The piping, as Tom said, is all welded piping and it's supported. We spoke with Dr. Kennedy about this and he says that that is seismically rugged. His first estimate is that i' you support it laterally, also, that it should have a HCLPF value in the .9g range.

Within the nuclear annex, we have a limited water source for the automatic sprinkler systems. The major water sources are outside the annex in separate buildings and are non-seismic and do not have the welded piping. So they would probably not be available for a spurious actuation of a fire system in a seismic event.

So in looking at the available inventory for the spray-down, from a flooding standpoint, we only get a depth of about four inches, I believe it was.

MR. MICHELSON: But where is the water coming from? It's coming from the ceiling, not from the floor. It's coming down on the equipment before it ever gets to the floor and that's the concern, what it's doing to the equipment in the process of coming down to the floor. So I don't think pedestal heights are meaningful at all.

The ability of the equipment to resist the spray is meaningful, but not the elevation of the equipment.

MR. CROM: Again, it's a limited source, as Dave said.

MR. MICHELSON: It's speculative as to how limited it is. That depends on what's happened to the rest of the fire protection system during the earthquake.

MR. CROM: I agree.

MR. MICHELSON: We just don't know. On the other hand, you can eventually understand that this is happening and get it isolated, but that takes time. First of all, in an earthquake, things are exciting, I'm sure, and maybe this is not high on their list of things to think about. I don't even know if you have a detection -- I guess you've got an alarm that says your pre-action valve opened, but you have nothing that says that --

MR. CROM: You also have a safety-related sump and sump pumps in that particular detection.

MR. MICHELSON: Then the water finally gets to the floor and gets into the sump. Is it alarmed or is it only monitoring how often --

MR. CROM: It would be alarmed on high level.

MR. MICHELSON: No, but it doesn't get to high level when it starts pumping. It just pumps the water that's coming in.

MR. CROM: I understand.

MR. MICHELSON: So you may not even know it's spraying in there until somebody walks in or whatever.

MR. CROM: The alarm would be on your pre-action valve.

MR. MICHELSON: Only on the pre-action.

MR. CROM: That's correct.

MR. LINDBLAD: Mr. Crom, when he says welded piping, was he talking about butt welding or socket welding?

MR. CROM: Socket welding. It could be either, depending on the pipe size.

MR. LINDBLAD: I am almost sure that the nozzles in the fire system would be socket welded.

MR. MICHELSON: Yes.

MR. CROM: Yes.

MR. MICHELSON: They'd have to be.

MR. LINDBLAD: Which isn't quite as rugged as butt welding systems.

MR. CARROLL: Why does every earthquake, I heard it yesterday again, result in fire systems letting go? Loma Prieta wiped out the whole United Airlines wing of San Francisco Airport because of fire systems letting to and I think Kennedy or Idriss yesterday mentioned --

MR. LINDBLAD: In the North Ridge recently, all large commercial buildings have provisions to evacuate the building on earthquake and nobody goes back in until a building inspector says it's safe. So as a result, a substantial amount of the damage in North Ridge is based on flooding of contents because a mechanic wouldn't go in and turn off the water.

MR. SEALE: And they had inadequate water to fight fires because of the bleed-down of the system.

MR. LINDBLAD: Yes.

MR. CARROLL: Real fires.

MR. MICHELSON: The real question here, I think, is simply why aren't we seismically qualifying at least within the diesel compartment and get rid of all this Mickey Mouse -- I think I almost heard it was, but not quite. What's wrong with just going the rest of the way? It's no big deal.

MR. LINDBLAD: I'm sure it is on the basis of two over one. I'm sure that the supports --

MR. MICHELSON: That keeps the big pieces from coming down.

MR. LINDBLAD: Yes.

MR. MICHELSON: I think there's hardly anything left but making it a seismically qualified system inside of the diesel compartment.

MR. CROM: Let us take that under advisement and get back to you with a response to that question.

MR. MICHELSON: And then I think you've got to begin to get --

MR. DAVIS: In this scenario, of course, the diesel generators would be trying to start because you would lose off-site power.

MR. CROM: Correct.

MR. DAVIS: So they would be trying to start at the same time they're being sprayed. I don't know how fragile the nozzles are on the fire protection system either.

MR. CROM: If your piping withstood the seismic event, not only would you have to have the pre-action valve opening -- and that would be the most likely, not only from the signal that you're talking about. The pre-action valves are only a flapper that's held in place and the seismic event could actually cause it to open. We have looked at that and agree that during a seismic event, they could open, even if we seismically qualified the lines.

MR. CARROLL: Could open, but would they stay open?

MR. CROM: They would stay open, because when -- all it is is a flapper held with a solenoid and if that flapper opens, it stays open.

MR. MICHELSON: It stays.

MR. CROM: But you still have to have the heat-sensitive spray nozzle to actuate, as well, because you've got redundancy there. I think the real question --

MR. CARROLL: And diversity.

MR. CROM: Yes.

MR. CARROLL: Right, Pete?

MR. MICHELSON: What do you mean diversity?

MR. DAVIS: I don't think diversity.

MR. CROM: The real question is whether the pipe ruptures and you're spraying it down. We've said it's seismically rugged and there's a question as to why not go further and make it all Seismic Category 1. We will take that under advisement and respond to it.

MR. DAVIS: The scenario is you get both diesels this way, because if one fails -- they're highly correlated because they're at the same elevation.

MR. CROM: I understand.

MR. DAVIS: Then your combustion turbine is not seismic, either.

MR. CROM: That is not. It is not seismic. However, we have a requirement, which Dave, I think, will talk about. There's a HCLPF requirement, even though it's not Seismic Category 1.

MR. DAVIS: To me, that's just as good.

MR. CROM: It must withstand a certain HCLPF as far as the seismic PRA is concerned.

MR. MICHELSON: You're talking about the combustion turbine.

MR. CROM: Yes.

MR. MICHELSON: How about all the wiring coming into the building and everything?

MR. CROM: I'll have to ask Dave to address that. Are you going to address that, Dave, in your PRA?

MR. CARROLL: Pete, you're cheating. You're moving into PRA space and we're trying to get rid of floods.

MR. DAVIS: I'm trying to move it into the important areas here.

MR. CARROLL: We'll get there.

MR. MICHELSON: I think you had an answer back here.

MR. FINNICUM: Would you like me to provide an answer?

MR. CARROLL: Sure, do it.

MR. FINNICUM: This is Dave Finnicum from ABB. In the PRA, based on information provided by EPRI in the URD, the seismic fragility of the combustion turbine was assumed to be about .36g, which is above the fragility of the off-site power source, which is about .12g. It's underground cabling from the alternate AC source into the building and this is assumed to be seismically rugged.

MR. MICHELSON: It is required to be or just assumed to be?

MR. FINNICUM: It was assumed to be.

MR. MICHELSON: Are you going to have a requirement, an interface requirement for the COL holder that they make it seismically rugged?

MR. FINNICUM: That has not been specified at this point.

MR. MICHELSON: Then I can't assume it is unless it's specified.

MR. CARROLL: We also discussed the fuel supply to the gas turbine yesterday and I don't think I got a complete answer on how good it is seismically.

MR. CROM: Again, the fuel supply is not a Seismic Category 1.

MR. CARROLL: I understand.

MR. CROM: Dave, can you address the fragilities and things like that of it?

MR. FINNICUM: That was included within the EPRI discussion on the seismic ruggedness of the AC source.

MR. MICHELSON: How did EPRI know that? Because that tank arrangement is a site-specific situation, too. The COL holder is going to design that fuel storage, I think. So how would you know ahead of time what the seismic ruggedness is unless you specify it?

MR. LINDBLAD: Are we talking about deterministic design basis accidents or the PRA now?

MR. MICHELSON: We're talking about the earthquake in the PRA.

MR. LINDBLAD: PRA.

MR. MICHELSON: Yes.

MR. LINDBLAD: So it seems to me that the PRA examiner inspects what the conditions are.

MR. MICHELSON: But what is he inspecting? We don't have any design or anything to inspect.

MR. LINDBLAD: The concept, then.

MR. MICHELSON: So you look at the -- you look at the requirements is what you look at, and I'm asking where are the requirements that say that it's going to be seismically rugged and whatever. I didn't find them, but you can point out where I should read and I'll read it.

MR. DAVIS: The COL will verify, after the plant is built, the seismic capacity.

MR. LINDBLAD: The assumptions of the PRA, yes.

MR. DAVIS: The seismic capacity of all this equipment If there's a problem at that point --

MR. LINDBLAD: Whether it's specified cr not.

MR. MICHELSON: If they have to meet the PRA requirements, then that's great. I didn't find that either. Is there something that says that they must verify and meet the PRA assumptions?

MR. CARROLL: That's the D-RAP and U-RAP.

MR. FINNICUM: This is Dave Finnicum, again. Yes. What it is is in Section 19.7.5.3, there is an area that specifically talks about the assumed fragility for the alternate AC source and it references the EPRI URD report that talked about the considerations on which they based their evaluation of the fragility of the AC source. That is there.

I believe based on discussion with the NRC, we have also added into the -- I believe it's 19.7.5.3 -- a discussion that talks about that the COL must perform a seismic walkdown for the plant to confirm the assumptions made in the seismic PRA.

MR. DAVIS: Right.

MR. MICHELSON: The assumption includes all the electrical breakers and controls and whatever it takes to make that gas turbine work.

MR. FINNICUM: This is correct.

MR. MICHELSON: So if there's a walkdown requirement and a verification of the PRA, that should take care of it.

MR. EL-BASSIONI: I'm El-Bassioni. I'm in the PRA Branch of NRR. Dr. Michelson, usually, in conventional PRAs, we do not consider cables in seismic PRAs, because we assume that they are flexible enough to take seismic events.

MR. MICHELSON: But structures are important, too, to house them.

MR. EL-BASSIONI: What?

MR. MICHELSON: The structures that house the cables are just as important and they are not necessarily flexible.

MR. EL-BASSIONI: Yes. For this reason, as you're saying, this is a basic assumption in PRA and we are highlighting key assumptions and the most significant insights to be included in the design control document and as Tier 1 or Tier 2. We are going to see to that. This assumption is highlighted.

MR. MICHELSON: At the time of COL licensing.

MR. CARROLL: Moving on.

[Slide.]

MR. CROM: Water lines to HVAC air conditioning units around the control room, and we're talking about the air conditioning for the control room itself, are contained in rooms and we have curbs around those particular rooms so that if there would be a moderate energy break, and these are small lines, that the flood would not go into the control room or the computer rooms, but be directed down around those into the lower elevations.

MR. MICHELSON: Now, none of those lines are in the control room, though.

MR. CROM: That's correct.

MR. MICHELSON: Or in the computer room.

MR. CROM: That's correct.

MR. MICHELSON: From your earlier statement.

MR. CROM: Yes.

MR. MICHELSON: And it can't get in there. Now, the rooms that they are located in are just for the air handling units.

MR. CROM: That's correct. The component cooling water heat exchanger structure, and there's one structure for each division, is -- I say it's divisionally separated, but there actually is -- it's divisionally separated because there are two structures.

It also is, I believe, separated -- each heat exchanger is separated within a division, such that they -- since we have two buildings, you can't -- a flood or break from service water or component cooling water in those particular buildings cannot effect both divisions.

Also, from a turbine building standpoint, we have one door that leads from the turbine building to the nuclear annex. This door is located at Elevation 130 plus six and that is at an elevation such that any flood that may occur in the turbine building, the turbine building flood will flood out, since we've got an aluminum-sided turbine building, onto the grade elevation at 91.9 before it ever reaches this door.

MR. MICHELSON: There must be some lines from the turbine building over to rad waste, aren't there?

MR. CROM: Yes. They're all located out of the building. They do not --

MR. MICHELSON: They apparently don't have to be processed.

MR. CROM: They do not run through the nuclear annex.

MR. MICHELSON: How do they get over to the rad waste building?

MR. CROM: Through pipe tunnels in the ground.

MR. MICHELSON: And they don't in any way connect to anything but the rad waste portion of --

MR. CROM: That's correct.

MR. MICHELSON: -- nuclear island.

MR. CROM: That's correct.

MR. MICHELSON: Looking backwards, how do you -- or are you going to get to how you -- floods in the rad waste building, how can they get into the nuclear island?

MR. CROM: There is a flood barrier in the pipe chase between the two.

MR. MICHELSON: Are you going to talk about it in a little bit?

MR. CROM: I don't have a slide, but the answer is that the pipe chase leading between --

MR. MICHELSON: How about doorways from the rad waste building, are they all above elevation, whatever it is, 91 or whatever?

MR. CROM: Do you recall, Todd?

MR. OSWALD: No. They are the -- one door we have I think is right at grade.

MR. MICHELSON: But there are no doors below grade.

MR. OSWALD: That's right. There are no doors below grade. They're one foot above grade, as we've committed to all the doors.

MR. MICHELSON: There are probably, undoubtedly, in fact, pipe penetrations below grade from rad waste over into the nuclear island.

MR. CROM: Yes.

MR. MICHELSON: Where will I read how those are going to be sealed for a flood now in the rad waste building, which can get there a number of different ways.

MR. CROM: You mean have we got a statement anywhere, and I'm not sure we do. I know the answer, but I'm not sure there's anything in the SAR. We'll have to look and make sure there is. The answer is there is a flood barrier in that penetration.

MR. MICHELSON: Any time you connect buildings together with umbilical cords, I think it behooves the safety analysts at least to look carefully at the umbilical cords to make sure they aren't common connectors to all buildings. I didn't find that kind of a connection anywhere.

MR. CROM: We will look for it and if it's not there, we'll add the words.

MR. MICHELSON: It's easy enough to -- I'm sure you're taking care of it all, but this makes it --

MR. CARROLL: I was surprised to hear one door. It wouldn't meet standards in California, at least. You have to have two ways out of any building.

MR. CROM: Todd?

MR. OSWALD: That's the one door into the rad waste building, but there's also doors -- the rad waste building doesn't cover that whole length of wall along the --

MR. CROM: We're talking the door from the nuclear annex to the rad waste building. We're not talking about doors in and out of the rad waste building.

MR. MICHELSON: Only above grade we were asking. There may be several doors further up, I don't know, but none below grade was the answer.

MR. OSWALD: No doors are below grade. The other egress doors, there are other egress doors, but not into the rad waste building.

MR. CARROLL: Fine.

[Slide.]

MR. CROM: We did perform an analysis. What we did -- and this is all in Chapter 19 under the flood PRA. We looked at the volumes of various large volume sources that were contained in one division or each division of the nuclear annex. We included one component cooling water division, including the external piping, the piping leading to the component cooling water heat exchanger structure, and the surge tank, the in-containment refueling water storage tank, one emergency feedwater system division, including the emergency feedwater storage tank, which is 350,000 gallons.

MR. MICHELSON: Is that a tank within that compartment?

MR. CROM: Yes.

MR. MICHELSON: It didn't show a tank in the compartment.

MR. CROM: The compartment itself is the tank.

MR. MICHELSON: That was my question. Is this a box, in other words?

MR. CROM: Yes.

MR. MICHELSON: And it fills the entire volume shown on the drawing.

MR. CROM: That's correct.

MR. MICHELSON: That must have pipes going off here and there, for nothing more than to keep the water clean.

MR. CROM: Yes. You have pipes coming from the condensate system.

MR. MICHELSON: And there's a fairly large volume, 350,000, I think you said.

MR. CROM: Yes. We also included the entire fire protection system, including the two external water supply tanks. I think each one of those tanks are 300,000 gallons. The chemical volume control system, including the external hold-up tank, the boric acid tank and the reactor makeup water tank.

If you look at the total water volumes of all those, even if they all simultaneously failed in one division, we have demonstrated through this analysis that that comes up to an equivalent volume of 385,521 cubic feet. Division 1 is 477,000 cubic feet up to Elevation 70 and Division 2 is 525,000 cubic feet, and that includes a very conservative analysis assuming that 50 percent of that space is occupied by equipment.

So what we have demonstrated there is that this Elevation 70, and you had asked a question about that Elevation 70, that we can take a flood of one division without going through a doorway or something up on Elevation 70.

MR. CARROLL: Now, if you put appropriate conservatism into the analysis, what's the answer?

MR. CROM: Can I get more conservative?

MR. MICHELSON: It has to be the same. You've already named the maximum possible water sources, but you haven't named all the ways water could get in. You just named the sources, the storage tanks, and I'm not sure that that's the only way water can get in and flood the building.

MR. CROM: You're talking about the external floods.

MR. MICHELSON: Yes. What's happening out in the yard or what's flowing back from the cooling pond or however, depending on how this whole thing is arranged. It depends on the event you want to name and your ability to prevent siphoning or back-flowing and things of that sort. Generally, that's not a problem --

MR. CROM: Traditionally it's always been the service water system and the make of that. We have that contained in the outside nuclear annex.

MR. MICHELSON: You've got a heat exchanger in between and you put it out in a separate building, and that's a big step in the right direction.

[Slide.]

MR. CROM: We do have a COL applicant item that he shall perform a flooding analysis associated with high and moderate energy line rupture analysis outside containment. I think that from what I've told you here, it's easily demonstratable that we can shut this plant down considering a single failure, particularly since they allow you to take credit for non-safety equipment in moderate energy line break analysis.

And with the combustion turbine, we could flood the whole division and be able to shut the plant down considering a single failure since we have redundancy in all our safety systems in the opposite division. Plus, with the combustion turbine, we would have redundancy on the off-site power.

MR. CARROLL: And you're saying with a single failure, meaning the 1E EDG and the --

MR. CROM: And the combustion turbine would be able to meet that particular single failure. However, he could --

MR. MICHELSON: How is the combustion turbine power brought into the two divisions?

MR. CROM: The actual power itself, the actual switchgear is located in the turbine building. We have a requirement that the -- and that's discussed in the fire protection section.

MR. MICHELSON: So the power is in a non-seismic building, then, which we talked a little more about, asked him about the duct work and everything, yeah, that's all seismic. You're coming into a non-seismic building with your power distribution.

MR. CROM: Let me answer this question, because we don't have to consider a seismic event here.

MR. MICHELSON: Okay.

MR. CROM: The X and Y buses, the cables are separated by the divisional wall, X being in Division 1 and Y being in Division 2.

MR. MICHELSON: Wait a minute. The turbine building doesn't have a divisional wall.

MR. CROM: No. I'm talking about once it enters the nuclear annex.

MR. MICHELSON: Okay. I was worried about outside.

MR. CROM: Now, what I'm saying is the flood in a nuclear annex, if it floods one division, will not flood into the turbine building.

MR. MICHELSON: Yes.

MR. CROM: So that the combustion turbine and the switchgear are protected, but also the power cables going to the two Class 1E buses are also separated, so that you would have power to those.

MR. MICHELSON: Is the power distribution drawing in the SSAR somewhere?

MR. CROM: Yes.

MR. MICHELSON: Because there you do worry about the effects of flooding the switchgear in one division. There are going to be some power cables flooded in one division.

MR. CROM: The Class 1E switchgear are located on Elevation 70 and they're quadratized.

MR. MICHELSON: And you're going to make sure that that doesn't interact back to the panel that you've put in the turbine building and cause you to lose both sides because you've got a fault that you can't clear.

MR. CROM: Yes. The location of all the switchgear is shown on the general arrangement drawings.

MR. MICHELSON: Okay. I'll look at it. Thank you. You do have to go back and think about that seismicity some more because I thought you were bringing it straight into the seismically qualified building and not into the turbine building from the combustion turbine.

MR. CROM: Remember that the turbine building is a somewhat Seismic Category 2 structure.

MR. MICHELSON: Somewhat, yes, and that's what they're going to verify in this whole explanation when it comes out. Not quite as good, I think, as the --

MR. CROM: To conclude the flood section, as far as ITAAC, the things that we have addressed to ensure that they are met in the final design as far as flood. We have provided those flood barriers, the ones I showed you on Elevation 50, they're in the ITAAC.

Structural load from flooding is an ITAAC item. It is considered that you have to consider the hydraulic forces due to flooding.

MR. LINDBLAD: Mr. Crom, are we talking internal and external now?

MR. CROM: That's correct.

MR. LINDBLAD: When we're talking flood protection.

MR. CROM: That's correct.

MR. LINDBLAD: Thank you.

MR. CROM: The sensors on the flood doors are also an ITAAC item in the structural -- nuclear annex structural ITAAC. The divisional and quadra separation of the floor drains are an ITAAC item in the equipment floor drain system ITAAC. Station service water located outside of the nuclear annex is essentially covered in the station service water ITAAC and the location of it.

The divisional separation of the systems is in every system ITAAC. You have to ensure that it is divisionally separated by the divisional wall. Safety Class

3 check valves to prevent the back-flow that I talked about in each of the sumps is also in the equipment and floor drain ITAAC. The reactor building subsphere and diesel generator rooms provide redundant safety Class 3 sump pumps which are powered from the diesel generators is also in the equipment and floor drain ITAAC.

MR. LINDBLAD: Most of this has been directed to the nuclear island requirements. Are there any yard requirements on flood protection? Are buried tanks permitted?

MR. CROM: I'm not following your question. Are buried tanks permitted, the answer is --

MR. LINDBLAD: In balance of plant, can one --

MR. CROM: Yes.

MR. LINDBLAD: -- bury a tank that would pop up with an external flood if it were empty?

MR. CROM: I don't know if any safety-related tanks, because the only safety-related tanks that would be in the yard is the diesel generator fuel oil, and that is in a structure.

MR. LINDBLAD: But there is no limitation on that.

MR. CROM: No.

MR. MICHELSON: It's a waterproof structure up to grade.

MR. CROM: That's correct.

MR. LINDBLAD: How is the hydrogen stored and the chlorine stored?

MR. CROM: The hydrogen and all those gases are a COL item. It's not part of the standard design. However, there are interface requirements that they be stored -- protected and stored a certain difference from safety-related structures, and the protection, I think, is what you're referring to.

Unless there are more questions on flood, I'm going to go into the high energy lines.

MR. CARROLL: Shall we take a break?

MR. MICHELSON: Before we do that, let me ask you one question. I was looking in the order in which these were and you've got a drawing back here which I asked a question on yesterday. I don't know how to identify the drawing, except that it says "Nuclear Island Structure Section AA."

MR. CROM: Yes. I know which one you're talking about.

MR. MICHELSON: It's the first one of those series.

MR. CROM: This one here.

MR. MICHELSON: Yes. Could you clarify for me this flood wall that keeps the electrical stuff out of the balance of the building flood or are you conceding that the water goes into the electrical area, but on only one side? Is that -- because there's no flood wall between the electrical and the mechanical area there.

MR. CARROLL: You're talking about which room?

MR. CROM: Let me make sure I understand you.

MR. CARROLL: Far right?

MR. MICHELSON: Far righthand corner.

MR. CARROLL: The vital instrument and equipment room.

MR. MICHELSON: Right.

MR. CROM: Yes. We showed that on the Level 1.

MR. MICHELSON: Is the idea that you left that flood, along with the flood that occurs in that side of the divisional wall?

MR. CROM: Let me pull up another slide here a minute.

MR. MICHELSON: If you do, then I guess you -- I see doorways --

MR. CROM: No. The answer is no and I'm looking for that particular slide, because we have a flood wall on Elevation 50.

[Slide.]

MR. CROM: Look at the plan view. This is the flood barrier I'm talking about right here all the way through around here. This is where your electrical is, your

--

MR. MICHELSON: That just happens to be on top of the pedestal on your drawings, so you can't see it.

MR. CROM: It does not show up in that view.

MR. MICHELSON: That's the one that keeps it out.

MR. CROM: Yes.

MR. MICHELSON: And that goes up to grade.

MR. CROM: This one goes up to -- this flood barrier goes up to Elevation 70. Now, these walls for external floods go all the way up to grade.

MR. MICHELSON: Yes. And the divisional wall goes all the way up to the building, all the way to the top.

MR. CROM: Correct.

MR. MICHELSON: But without any doors, it's -- I thought below grade had no doors on the divisional wall.

MR. CROM: Below grade?

MR. MICHELSON: Yes.

MR. CROM: No. We have doors at Elevation 70.

MR. MICHELSON: You do, okay.j

MR. CROM: And a divisional wall. This wall here.

MR. MICHELSON: But you have no sources that can get --

MR. CROM: That's correct.

MR. MICHELSON: -- above 70.

MR. CROM: That's correct.

MR. MICHELSON: Okay. So all of these have doors above 70, because I see some --

MR. CROM: That's correct.

MR. MICHELSON: I thought I saw some.

MR. CARROLL: Okay, Carl?

MR. MICHELSON: Yes.

MR. CROM: My high energy line is not going to take long, if you want to finish that. Then we can take a break and go on to PRA.

MR. MICHELSON: It may take a little while.

MR. CARROLL: Yes. I'm afraid it may take a little while.

MR. CROM: I'll let you decide.

MR. CARROLL: Let's recess and return at 10:30.

[Recess.]

MR. CARROLL: Let's reconvene.

Tom, do you want to continue?

MR. CROM: Yes. I was going to go on to high energy lines.

[Slide.]

MR. CROM: The first slide here is just to identify, when we talk about high energy lines, what systems we're talking about inside containment, and these are all listed in the SAR -- I don't recall exactly the table number -- in Chapter 3, and of course, we're talking about the main steam system, main feedwater system, steam generator blowdown, steam generator wetlayup and recirculation, reactor coolant system, safety depressurization system, the chemical volume control system, safety injection and emergency feedwater, and there's only portions of safety injection and emergency feedwater, and those are meshed to the pressure isolation valves. They are considered moderate energy lines from that based on usage factors.

MR. CARROLL: When are we going to talk about the safety depressurization system with respect to the steam condensation?

MR. CROM: That will be in Chapter 6, which is schedule on April 5th and 6th.

MR. CARROLL: So, you're going to be ready to tell us why -- some of them hopefully will be.

MR. CARROLL: Why steam condenses in water, cold water.

MR. CROM: That is in Chapter 6 and discussed in Chapter 6.

MR. CARROLL: And the sparger design?

MR. CROM: Yes.

MR. CARROLL: And the testing you've done in support of it?

MR. CROM: That's correct.

MR. CARROLL: Okay.

[Slide.]

MR. CROM: Now, my next slide is the listing of the systems located -- with high energy lines -- located outside of containment within the nuclear annex, and then I'm going to have slides that tell you where the locations of these lines are.

Of course, the systems we have listed are the main steam system, the main feedwater system, steam generator blowdown, emergency feedwater steam line to the turbine-driven pump, and the chemical volume control system. I will note those are small lines. They're two-inch lines in letdown and also on the pumps, the charging pumps going in there.

MR. CARROLL: Refresh our memory, Tom. What is the definition of a high energy system?

MR. CROM: A high energy system is any system that has a temperature over 200 degrees and --

MR. GUS: 275 psi.

MR. CROM: -- 275 psi. Also, you have a usage factor, and I'm trying to remember what that is, you know, how often it's used, and most of the safety systems, like safety injection, shutdown cooling, and emergency feedwater fall within the usage factor and considered moderate energy lines.

MR. MICHELSON: In looking at your SSAR, on page

3.6-3, you've got an additional wrinkle on the definition which maybe I just didn't remember has ever been there, and the staff can comment if they wish. In addition to the criteria of temperature and pressure, you've got a further criterion, and that is that it's pressurized above atmospheric pressure during normal plant operation. I never heard of that one, but that's a part of your criteria. It must also be pressurized during normal operation to be high energy, and I can't believe that. It's high energy when it's in operation and it's pressurized.

MR. CROM: Is that in the high energy section or is that in the moderate energy section? I know that's a true statement for moderate. It's really the usage factor on high energy lines.

MR. MICHELSON: I think you just got carried away or something.

MR. CROM: I don't recall that as being a definition in high energy. I know it is in moderate.

MR. MICHELSON: I had not heard of it before, but I thought maybe the staff had heard of it before, and something may have been added as a wrinkle. I don't know.

MR. RITTERBUSCH: This is Stan Ritterbusch. Tom Crom and I will take a look at those words and get it straightened out.

MR. MICHELSON: I think it's probably just an error.

[Slide.]

MR. CROM: I'm going to talk about the location of these lines.

The main steam, main feedwater steam generator blowdown, as you've probably seen -- we've shown previous slides, and I'll show another one here -- penetrate on each side of the containment or on each side of the divisional wall. They exit from the containment through the annulus through the shield wall, in-yard pipes, and then enter through the main steam valve house and then exit the main steam valve house through the yard, along piers, into the turbine building.

Just a quick refresher. Unfortunately, this slide does not show the lines, but we're talking, on each side, this being the main steam valve houses, the penetrations through the containment into the main steam valve house, and the main steam lines cross the yard into the turbine building.

MR. MICHELSON: Have you done the pressure calculations for the pipe break in the valve room?

MR. CROM: Yes, we have.

MR. MICHELSON: And what pressures are we dealing with?

MR. CROM: Todd, do you want to address that?

MR. OSWALD: This is Todd Oswald, Duke Engineering Services. We are looking at 10 psi on that one.

MR. MICHELSON: Now, again, I suspect there are various penetrations of that wall back into the nuclear island, probably, at least for conduits and control cables and I don't know what else, but I couldn't find any detail that says that the penetrations can take the 10 pounds, just like the wall, I'm sure, must handle it all right, but how about the penetrations of the wall? I couldn't find any doors, but somehow you've got to get in that area.

MR. CROM: There is a door.

MR. MICHELSON: There must be doors, and where is the pressure rating on the door, or where is it dealt with?

MR. CROM: Todd, do you have an answer to that?

MR. OSWALD: It's not specifically stated.

MR. CROM: The answer is that they will be qualified for the 10 psi.

MR. CARROLL: That's some kind of a door.

MR. MICHELSON: It will have be a big heavy steel submarine-type door. For 10 pounds pressure, you're going to have a real door, and 10 pounds sounds not unreasonable. It's even a fairly good wall.

MR. OSWALD: They're four-foot-thick walls, and actually, where the main steam line penetrates the walls, we had to go up to five-foot.

MR. MICHELSON: My one concern was the penetrations and what kind of sealant you have there and will it take the pressure? If not, where does the steam go back into? One concern is the diesel compartment is right next to the -- has a common wall, in part, I think.

MR. OSWALD: The diesel compartment is located down below grade, although it goes from elevation 50 up through elevation 70. So, the roof of the diesel is elevation 90, and you have the emergency feedwater tank directly below the main steam valve house. About elevation 104 is the top of -- the bottom of the main steam valve house. So, it wouldn't interfere with the diesel.

MR. MICHELSON: It shouldn't if there aren't any penetrations. If there are, then I don't know that -- I couldn't see enough of the detail to tell for sure. I assume not, but it's right in that neighborhood, at least. But I would expect, before we're done, to see some kind of design requirements on sealing up that room against these kinds of pressures, unless you can show you don't need the sealant.

MR. CARROLL: Where does the room vent to?

MR. CROM: The room vents out of louvers on each side of the valve house --

MR. CARROLL: To the outside.

MR. CROM: -- to the outside.

MR. MICHELSON: That's accounted for in the 10-pound calculation.

MR. CROM: That's correct.

MR. MICHELSON: It's just a small detail, but again, we would like to get an answer to it.

MR. CROM: Okay.

MR. CARROLL: Is the 10 pounds quasi-steady-state pressure, or is it the peak pressure you reach until the louvers open, and then does it drop to something much less than that?

MR. CROM: Fred Carpentino, can you answer that?

MR. CARPENTINO: That is a peak pressure, and the pressure would come down after it reached that peak, through the louvers.

MR. CARROLL: At what sort of a level?

MR. CARPENTINO: I don't remember how quickly or how low it dropped.

MR. MICHELSON: I suspect your louvers are opening --

MR. CARPENTINO: They are very large louvers.

MR. MICHELSON: -- not at 10 pounds. Are they a 10-pound-rated louver? They open at 10 pounds?

MR. CARPENTINO: The louvers are always pressurized.

MR. MICHELSON: Does it even have louvers on it?

MR. CARPENTINO: It's basically bird screens and things like that.

MR. MICHELSON: Okay. That's different.

MR. CARPENTINO: Yes.

MR. MICHELSON: There's nothing dynamic about that. It's always open.

MR. CARPENTINO: Yes.

MR. MICHELSON: Yes.

MR. CROM: Fred, before you sit down, you told me that you had an answer for the question on the break analysis.

MR. MICHELSON: Before we finish this one, though, I don't know if Ivan would want to see the calculations on the valve room as well as the emergency feedwater, but he was very much interested in that.

MR. CROM: Fred, you said you could answer that now.

MR. CARPENTINO: Yes.

You had asked earlier, Mr. Michelson, about the pressure in the emergency feed pump room and how that was calculated.

After thinking about it, our memory banks got back into sync, and the calculation for the pump room, per se, was done in a rather simple manner, by hand calculations, assuming the inflow to the room was from the six-inch steam line to drive the pump at critical flow conditions, at full pressure, and that the venting from the room was countercurrent to the inflow from the pipe, inside the pipe, on the outside of the pipe, within the pipe chase itself, back up to the main steam valve room.

So, that was done in a steady-state manner by hand calculations.

MR. MICHELSON: Can you send us a xerox copy of the hand calculation?

MR. CARPENTINO: We could do that.

MR. MICHELSON: I don't think we have to have anything fancy.

Now, you did calculations for the valve room, as well, for the case of the steam or feedwater line breaks.

MR. CARPENTINO: Right. That was done with a computer code.

MR. MICHELSON: Okay. Which code did you use for it?

MR. CARPENTINO: I believe we used our DDIF computer model, which is used for the subcompartment-type pressurization.

MR. MICHELSON: Is it written up anywhere in the SAR?

MR. CARPENTINO: It's referenced -- I think it's documented in a topical report, the number of which fails me right now.

MR. MICHELSON: We can get the topical report.

MR. CARPENTINO: I believe that's a matter of public record. It was reviewed the staff.

MR. MICHELSON: We can see it if it's referenced in the SSAR. I don't think there's any question of public record or not.

MR. CARPENTINO: Yes.

MR. MICHELSON: But the reference will be in there. In what chapter will I look for that?

MR. CARPENTINO: That will be in Chapter 6, referenced within 6.2, I believe.

MR. MICHELSON: Okay. That will do it. We'll ask for the reference.

MR. CARPENTINO: Okay.

MR. MICHELSON: Thank you.

MR. CROM: My next bullet I think we've already covered. The emergency feedwater steam line to the turbine driven pump is located in the main steam valve house, in the turbine driven pump rooms, and then is located -- routed through the vented chase between the two rooms.

Finally, the two lines on the chemical volume control system are located in a pipe chase after they penetrate the containment and then are routed through the pipe chase into the chemical volume control system area, which is a non-safety area.

I don't have much of a slide, but the pipe chase we're talking about, on the Division 2 side, is the one through here, CVCS area being in this area, and then on the upper elevations and then also in this elevation here.

MR. MICHELSON: What's the largest pipe size?

MR. CROM: Two inches.

MR. MICHELSON: Two-inch?

MR. CROM: Two inches.

MR. MICHELSON: What pressures did you get in that compartment when you broke the pipe?

MR. CROM: That line was not analyzed.

MR. MICHELSON: How well vented is the room?

MR. CROM: That's something which is very difficult. The reason it was not analyzed is you cannot do an analysis until you know what all your vent spaces are.

MR. MICHELSON: Is it a COL action item, then, to do the analysis?

MR. CROM: I'm not sure. Do you know, Todd, if that was a COL?

MR. OSWALD: This is Todd Oswald, Duke Engineering. Yes, there is a requirement to determine that pressure once the final duct-work and all of the penetrations into the room are determined.

MR. MICHELSON: That's a COL action item? It ought to be if it's not. It's something you can't do today, clearly.

MR. GERDES: Lyle Gerdes of ABB-CE.

I'm not sure, right now, in the SAR, if we've identified it as a COL action item. We have defined that that pipe chase will be designed for the pressures and temperatures for a pipe break. That would be done in the detailed design. Primarily, what that would define, then, is how much rebar, how much steel you need in those walls.

So, again, all of the detailed design has not been done. So, that would automatically become a COL action item when he does the detailed design.

MR. CROM: I believe there is a COL action item for all high energy line and moderate energy line breaks. I know there is one in flood, that the COL has to consider the effects of the flood and analyze all those, and also, in the EQ, there's also any effects from that, too.

MR. MICHELSON: A further extension of the question, then. Is there an ITAAC item that requires an inspection, you know, walk-down and so forth, to verify that these final calculations are realistic and that sort of thing?

MR. CROM: Can you answer that, Lyle?

MR. GERDES: Specifically for what you questioned, I don't believe the ITAAC identifies that. The ITAAC does identify that all the high energy line pipe breaks and protection from those pipe breaks, including pipe whip sprays, jet impingement, there will be analyses and a report relating to that.

MR. MICHELSON: But does the ITAAC require verification and examination of the report in a walk-down of the areas?

MR. GERDES: I do not believe it requires a walk-down. It does require a report.

MR. MICHELSON: What's the staff's position going to be in the case of this plant?

MR. TERAO: This is David Terao. There is no ITAAC that requires the walk-down, but that is specified in the SSAR.

MR. MICHELSON: Now, in the ABWR, there is an ITAAC requirement to do it.

MR. TERAO: Well, the details also in the ABWR were in the SAR. So, that's what we have also done on System 80+.

MR. MICHELSON: I was only making a statement. The ITAAC does require that it be verified and that a walk-down be performed.

MR. TERAO: That's correct. The ITAAC requires a general requirement to verify it, and the details are in the SAR.

MR. MICHELSON: Will this ITAAC have the same requirement?

MR. TERAO: Yes. Yes, it does.

MR. MICHELSON: Okay. Then we'll look for it when we review the ITAAC.

MR. TERAO: And the details are in the SAR.

MR. MICHELSON: Okay. I didn't find the details in the SAR. What details are you referring to?

MR. TERAO: The details as far as what would be included in the walk-down and what is included in the pipe break analysis report.

MR. MICHELSON: That's in the SAR?

MR. TERAO: That's in the SAR.

MR. MICHELSON: I found what's in the pipe break analysis report. I didn't find the walk-down part.

MR. TERAO: It was added in Amendment U or V. It was one of the later amendments.

MR. MICHELSON: I don't have those yet.

MR. CARROLL: Yes, you do.

MR. MICHELSON: Amendment T is the last one I got.

MR. CARROLL: Oh, no.

MR. COE: I had sent a large box.

MR. MICHELSON: I never got that box. You said you were going to send it me. It was about two weeks ago we chatted. I never got it, not before I left, at least, but I didn't leave until just a couple of days ago.

MR. CROM: That's all right. I didn't get my Amendment U until I just left.

MR. MICHELSON: Maybe that will clear it up, then. All right. We'll leave it for later.

MR. CROM: Okay.

MR. MICHELSON: All right. Thank you.

MR. CROM: That's all I have in my presentation. Any questions?

[No response.]

MR. CARROLL: We thank you.

Let's see. What's next, Pete? I guess we're going to move on into the PRA area. Are there some remarks you would like to make before we start here, Pete, since you've done a fairly detailed review?

MR. DAVIS: Yes, a couple of things.

I found this one to be the most comprehensively documented PRA that I've seen. When the UPS man delivered this to my door, I thought it was the entire SAR, and then I discovered it was only Chapter 19, all nine volumes of it.

MR. FINNICUM: And I was told to be terse.

MR. DAVIS: Let me just say a couple of things.

I thought it was a good PRA. I found a couple of things that I like very much.

One of them was a comparison of the as-built or, I should say, as-currently-designed plant versus the PRA-assumed design, because there have been some changes made since the PRA was finished, and that was very helpful, and this may have been because of the staff's requirements, I don't know.

You also had at least one operator-active commission, which is not commonly found in PRAs. This is the inadvertent entry into the feed-and-bleed mode by the operator. I don't know if there are any other acts of commission in there, but that one was certainly prominent.

One thing that troubled me a little bit -- I had trouble finding the results. I'm used to seeing those up front, but that wasn't the case in this PRA, and I had to dig around to find the results. It's always useful for a reviewer to know the results first, I think, so he can determine what's important and what isn't as he goes through the review. That's just an editorial comment.

One of the things that bothered me a little bit is that there are a number of assumptions made in the PRA that don't have any basis attached to them, and it's not clear that those assumptions are tied back into a design requirement anywhere.

I suspect a lot of the assumptions did come from the design requirements, but it's not so stated in many cases. I can give you a few examples as we move along.

It would have been helpful, I think, to show or at least refer to where this assumption can be validated.

One of them you make is there aren't any combustible materials in the electrical cabinets in the control room, and that's just an assumption that's stated, and that's used as part of your argument that you can't get a fire in the control room and that's not a contributing factor to the fire Core Damage Frequency.

In all other PRAs that look at fires that I'm aware of, the control room fire is usually the dominant contributor. So, it was a little bit unusual to see, in this case, that it was screened out, and I think you had some pretty good arguments for that, but that has to be reflected back into the design requirement.

[Slide.]

MR. FINNICUM: Let me just sign in, and then I'll tell you.

As I mentioned yesterday, my name is David Finnicum, with ABB, and I am the Task Manager for the System 80+ PRA.

One question I have -- have you received Amendment U and looked at that?

MR. COE: Yes.

MR. FINNICUM: That would be the large box.

MR. DAVIS: Yes, I think so. I couldn't read all nine volumes, I must confess.

MR. FINNICUM: Okay.

In Amendment U, in section 19.15.1, there is a new table that was added in Amendment U that was specifically added late to cover ongoing analyses.

What that table describes are the -- it's called the "Important Insights and Assumptions," and ABB-CE and the NRC staff had gone through the PRA and identified the important PRA insights in that and tied it back either to what we call Certified Design Material, which is ITAAC information, or Tier 2 information, which is information to be found in the SAR, or to COL action items.

The combustible material assumption that you specifically referenced that is specifically addressed in that table. It is tied back to design statements in -- I believe it's Chapter 7 of CESSAR-DC. We understand there were a number of assumptions.

MR. CARROLL: It seems to me that one of the points that was made during the discussion of Chapter 7 was that, given the kind of control room you have -- you don't have high voltages, you've got fiber -- that you just don't have the combustible loading that traditional control rooms have had. Is that not the case?

MR. FINNICUM: This is true, yes.

MR. MICHELSON: They certainly can still burn.

You've got a lot of wiring in the control room. You've got at least 110-volt equipment, probably, may even have some 220 in certain power supplies and so forth, I don't know, I haven't seen that level of detail, but you're not claiming it's all 24-volt inside the control room.

MR. FINNICUM: No, we're not. What we are claiming, as in the design, is that the materials used within that, other than the metal, would not sustain combustion.

MR. MICHELSON: What kind of wiring are you using for all your instrument wiring in the cabinets and whatever? What kind of insulation are you using on the wiring?

MR. RITTERBUSCH: We can't answer the details today. I do know that there was quite an extensive review, and the conclusion was that there was a very significantly reduced risk of fire in the control room.

MR. MICHELSON: I don't doubt there's a significantly reduced loading of combustibles, there's no doubt of that, but it's still combustible, and you still have to treat it accordingly.

MR. DAVIS: Well, the staff points out in the FSER that there are a lot of documents, of course, that are combustible and procedures and so forth.

MR. MICHELSON: Unless they've got some exotic insulation now, the wiring is, too. In fact, some of the instrument wiring is worse than the big power wiring.

MR. CARROLL: Some operators I've known have been known to burn procedures, too.

MR. FINNICUM: That was generally in the manager's office.

MR. MICHELSON: That's a good regulation to think about, isn't it?

MR. RITTERBUSCH: This is Stan Ritterbusch.

Mr. Michelson, I know you have a question. I'm not sure what the action is to resolve it. Do we owe you something on that issue?

MR. MICHELSON: I was only commenting on what Pete had read, I think, to the effect there wasn't any combustible, and apparently, there is combustibles in the control room, I think a significant amount, but it's nowhere near what it used to be.

MR. LINDBLAD: I think there's one issue about what is the definition of support combustion. Does it flame or does it smoke or does it smolder? I think that we've kind of identified that it probably won't ignite from --

MR. MICHELSON: This low-voltage stuff, unless they've gone to some of the -- there are some that won't ignite, but unless they've spent the money to get that for their low-voltage stuff, that's good stuff to burn. Wiring doesn't have to have much insulation, and it burns nicely unless you buy the --

MR. CARROLL: Yes, but in a manned area, you're talking about a local situation if you have some kind of a fault. I believe that people are going to be put that kind of fire out in a big hurry.

MR. WYLIE: There are insulations that, I'll say, won't burn.

MR. MICHELSON: There are, yes. That's why I asked which ones they're using for this plant.

MR. WYLIE: If they're using those, then they don't have a problem as a source from their wiring itself.

MR. MICHELSON: Precisely.

MR. WYLIE: Now, is that the case?

MR. MICHELSON: They didn't know yet.

MR. RITTERBUSCH: We can find some additional detail and provide it when we hit Chapter 9 in early April. I don't think we contend that fires are so small -- the likelihood of fire is so small that we won't have to ever evacuate the control room. We realize that we may. What we demonstrated is that there was adequate time so that transfer could be -- control could be transferred, and then they would exit and go down to the remote shutdown panel room.

MR. MICHELSON: Yes. I think all that is supposed to be factored into the PRA, including the probability of the failure, you know, the fire and so forth, and you've got to do it right. I don't know -- they may be using polyethylenes for all this 24-volt stuff. It's real good insulation. It just kind of burns a little bit.

MR. WYLIE: Well, there is the Raychem insulation systems that were developed for aircraft --

MR. MICHELSON: Yes, and they're goc

MR. WYLIE: -- and they are very good. don't really burn.

MR. MICHELSON: But I didn't find anywhere --

MR. WYLIE: The wires will fuse before the insulation will burn.

MR. MICHELSON: But I didn't find any commitment to that kind of insulation in the control room.

MR. RITTERBUSCH: We will describe our commitments at our next meeting.

MR. MICHELSON: And then you can judge fire accordingly.

MR. SALTOS: This is Nick Saltos from the NRC.

We looked at -- we did a scoping study, a risk study, considering a fire in the control room and how the functions would be transmitted to the remote shutdown panel and what functions would be accessed from the remote shutdown panel, and we found that there is enough hard-wire for those functions that the risk is --

MR. MICHELSON: I don't think anyone is questioning the design of the remote shutdown arrangement at all. That wasn't the point. The point was what the PRA is doing, and if they assume no combustibles in the control room, I think they've missed a point somewhere.

MR. SALTOS: But we assume that it can happen with a certain frequency, and we're trying to see how they can take care of it from the shutdown -- how the plant can be shut down from the remote shutdown panel.

[Slide.]

MR. FINNICUM: Briefly, what I want to talk about today is briefly to identify the objectives of the PRA, provide a brief description of the approach we used throughout the entire PRA and a brief description of the methodology, primarily discuss the results we have, and there are a couple of specific ACRS questions, issues that have been brought up in other areas that were pertinent to the PRA that I'm providing information on.

[Slide.]

MR. FINNICUM: Specifically, the objectives of the PRA performed for the System 80+ plant are we had to comply with the Severe Accident Policy Statement for providing a Level III PRA for an ALWR design; we have to demonstrate compliance with the EPRI ALWR Mean Core Damage Frequency Goal of 1 times 10 to the minus 5th events per year;

demonstrate compliance with a large release goal of 10 to the minus 6th events per year; and to demonstrate our containment performance and reliability. In addition, our internal objective was to use the PRA to support the design of the plant.

[Slide.]

MR. FINNICUM: The basic approach we used was to establish a baseline PRA for System 80+ using our currently certified design of the System 80. T is an NSSS design. We do have an operating plant, the Palo Verde plant, in operation, but the PRA was based on the certified NSSS design in CESSAR-F.

We used a balance of plant which was basically an amalgam of balance of plant for recent vintage CE plants. t was not strictly representative of any given CE plant. Basically, it was a BLP that would meet the interface requirements.

We then used the PRA as an evaluation tool for assessment of certain design changes, and through the process, the PRA that started out as a System 80 PRA evolved to the System 80+ PRA.

We prepared, then, the Level III PRA for the System 80+, and this included an evaluation of the external events.

[Slide.]

MR. FINNICUM: With the Level I portion of the analyses, the determination of Core Damage Frequency, we used pretty much standard methodology. We used the small event tree/large fault tree approach, which is used by most of the current PRA practitioners, with the exception, primarily, of Pickard, Lowe & Garrick.

Our front line system models address both the system component failures, we look at common cause faults, maintenance unavailability, operator actions, and we included a full support system model, including electrical power and component cooling.

The support system models were modeled to the same level of detail as the front line system models, and we solved this using the CAFTA code and performed full fault tree linking for the analysis.

MR. CARROLL: How would you characterize the approach that Pickles, Lox & Bagels uses on --

MR. FINNICUM: It is typically characterized as a large event tree/small fault tree.

MR. CARROLL: Okay.

[Slide.]

MR. FINNICUM: For the external events, the first step in the evaluation was basically a qualitative screening of external events. There is a very large listing that has been prepared by EPRI identifying many things.

We went through, grouped like events, things that had like effects, and looked at elements that could be grouped together as covered by existing events or things that could be covered as one event or things that, based on the standard site requirements, could be excluded. Such things as landslides, volcanoes were basically excluded from further quantitative analysis.

What we then did is identified events that were to be evaluated in more detail, either quantitatively or in more detail qualitatively.

The events we looked at were the tornado strikes, seismic events, which we used the seismic margin assessment for, and a scoping evaluation for internal fires and floods.

MR. SEALE: Could I confirm that you did verify that the hurricane problem was subsumed with the range of external assaults, if you will, that you had in this assessment?

MR. FINNICUM: Yes. It was subsumed within the tornado. The main impact was the wind velocity, and the tornadoes have larger wind velocities.

MR. SEALE: And some flooding.

MR. FINNICUM: Yes, some flooding. With the layout of our site, it should not produce significant external flood threat. So, the main threat was the wind loads.

MR. CARROLL: But for some sites, hurricanes could be different than tornadoes in that they do produce flooding, whereas a tornado typically doesn't.

MR. FINNICUM: Correct.

MR. CARROLL: Okay.

MR. EL-BASSIONI: This is El-Bassioni, the PRA branch of NRR.

Part of the post-certification tasks for the COL holder will be integrating site specifics. So, this would be addressed by the COL holder.

MR. CARROLL: We were just talking generally.

MR. LINDBLAD: Did I understand, then, on external floods, there is no PRA for external floods?

MR. FINNICUM: There is no PRA for external floods, based on the design we have of site requirements, where it's sloped away from it, and that the maximum flood level is one foot below grade.

MR. LINDBLAD: Mr. Michelson has this concern about umbilicals and penetrations that are supposedly sealed, but the seal, over a period of years, might fail, and one doesn't identify the seal to fail until there is water in the sump. That was not considered a major issue?

MR. FINNICUM: No, that was not considered.

MR. MICHELSON: They don't know where the seal is, and until you know that, you don't know whether it's a major issue or not.

[Slide.]

MR. FINNICUM: Shutdown risk was also identified in the PRA. The basic Level I analyses were performed for at-power --

MR. CARROLL: You deliberately skipped a slide, or we're out of order?

MR. FINNICUM: Oh, I'm sorry. I did not deliberately skip a slide. Yesterday I had pulled the seismic margin slides. If you'd like me to go back over those again --

MR. CARROLL: Are we all happy? Okay. Moving on.

MR. FINNICUM: Okay.

As I said, the main Level I PRA covered at-power events, which were considered to cover Modes 1, 2, and 3.

We also performed a shutdown risk analysis. In this case, basically had developed an outage profile, or Duke Engineering had developed an outage profile. This outage was divided into four plant operating states.

The first one was a Mode 4 or Mode 5 with normal inventory and Mode 6 with the IRWST full and refueling cavity empty. For these different conditions, it was all assumed to be at the equivalent in plant configuration.

The second operating state was what is called Mode 5R. In other words, it was in Mode 5, it was reduced inventory, and this includes the mid-loop operation.

The third plant operating state was Mode 6E. This is the case when we're in a refueling outage, with the IRWST empty, with the refueling cavity full, and the upper internals removed. In other words, the inventory from the IRWST has been transferred into the refueling cavity.

And finally, the fourth mode was Mode 6I. Again, the refueling cavity is full, with the IRWST empty but with the upper internals in place.

For each of these plant operating states, event trees were developed for four types of events. The first one was a loss of decay heat removal or DHR, the second was a small LOCA on drain-down events, third is fire, and fourth was the loss of offsite power.

MR. CARROLL: So, you're not considering in these event trees certain things. One that comes to mind are dropping of a heavy load damaging fuel.

MR. FINNICUM: We did not include an event tree for that event.

Eric, do you want to address more on that?

MR. SIEGMANN: My name is Eric Siegmann.

We did, as part of the review of shutdown risk, a look at dropping heavy items and considered, because of either procedures -- that is, paths of moving heavy items and such -- or because of the robustness of the piping, that dropping of heavy items was not risk significant, and that's discussed in Appendix 19.8A, which is a mechanistic risk assessment.

MR. CARROLL: Okay. What other kinds of things did you look at that aren't covered by event trees? That's one.

MR. FINNICUM: We have some back-up slides on the mechanistic analyses that were done. If you would like us to present that at this time, we could, or if you would like, we could hold it till later.

MR. CARROLL: Let's hold it till we get into shutdown risk.

MR. FINNICUM: Okay.

MR. RITTERBUSCH: We can certainly summarize by saying we looked at all of the issues identified by the staff and also documented in NUREG-1449. It was quite a thorough review.

MR. DAVIS: Did you look at seismic events during shutdown?

MR. FINNICUM: No, we did not.

[Slide.]

MR. FINNICUM: Within the shutdown risk, the initiating event frequencies that were used were taken from the Brookhaven National Lab 1991 study. We developed fault trees for each of the branch points in the event trees, and these were based on modifications of the Level I fault trees.

Within this, shutdown risk is heavily dependent upon operator actions. The human error probability that we used within the analysis were really developed for two basic response times. One was a 40-minute response time for reduced inventory conditions and, secondly, a two-hour response time for all other events.

MR. CARROLL: What does the 40 minutes mean? What happens in that time period?

MR. FINNICUM: Eric Siegmann will address that specifically.

MR. SIEGMANN: The 40 minutes represents the time to boil off the reactor coolant from the bottom of the hot leg to the active core and then heating up the active core until the onset of clad damage.

MR. DAVIS: This would be a mid-loop operation condition?

MR. SIEGMANN: Yes.

MR. CARROLL: Okay. And you're saying that the operators are capable of responding to this within 40 minutes?

MR. SIEGMANN: Yes, we are, because of the instrumentation available to the operator and alarms and such.

MR. CARROLL: How much margin is there?

MR. SIEGMANN: By the way, we calculated an error rate for the operator.

MR. DAVIS: Yes. There still is an error probability that he won't do it.

MR. SIEGMANN: Right.

MR. CARROLL: But assuming he does do it, does he just make it, or is there some margin here? You tell me it's 40 minutes until the bad things happen.

MR. SIEGMANN: It isn't a case that he does or doesn't make it. Well, actually, on each branch point, there is a case he does or doesn't make it. If he doesn't make it, that's basically failure just like failure of a mechanical thing. He doesn't half make it. But there's basically a probability that he does or doesn't make it.

MR. CARROLL: But a success path would be completed how much sooner than --

MR. FINNICUM: I think the timing was developed such that, if the operator was capable of performing the action, either start an injection or restore heat removal, within the 40 minutes, that he would make it. If it was longer than 40 minutes, then we would have seen onset of core damage, and at this point, one of the assumptions we made throughout the PRA is, once we had the onset of core damage, that was called core damage.

MR. CARROLL: Okay. I'm happy.

MR. DAVIS: I may be getting ahead of you here, but in your shutdown risk analysis, typically the containment will be open for part of that, if not most of it, and are you able to close the containment up if loss of offsite power occurs?

MR. CARROLL: Yes.

MR. FINNICUM: Eric Siegmann can address that.

MR. SIEGMANN: Yes, we are.

MR. DAVIS: How long does that take?

MR. SIEGMANN: We will be able to close the containment in one hour.

MR. DAVIS: One hour. Okay. Thank you.

MR. CARROLL: And you have considered all the things that might be -- all the cables and steam -- maybe not a steam generator but a coolant pump motor in the process of going through the equipment hatch. In one hour, you can clear all that out of the way and get the hatch closed.

MR. FINNICUM: Again, Eric.

MR. DAVIS: With the lights out.

MR. CARROLL: With the lights out.

MR. SIEGMANN: We do not plan to have any cabling run through the equipment hatch during shutdown modes, and as far as probabilistically, I haven't addressed, you know, the probability of having an RCP pump motor in the hatch at the time of the blackout or whatever, but when you're getting into multiple events of RCP replacement followed by station blackout, you're going to be well beyond design basis accidents.

MR. CARROLL: So, in other words, the hour assumes that there is nothing blocking the hatch at time zero.

MR. SIEGMANN: That's correct.

MR. CARROLL: What occupies the hour? What has to happen at time zero plus?

MR. SIEGMANN: By the way, at mid-loop, the containment will be closed.

MR. CARROLL: Always?

MR. DAVIS: That's required?

MR. SIEGMANN: Yes.

MR. CARROLL: Okay. So, at time zero, something has happened to suggest that we ought to get the equipment hatch closed. Some time is devoted to getting some people up there, and the rest is just operations that are required to swing the thing back in place and get four bolts in it?

MR. CROM: This is Tom Crom from Duke Engineering.

I wanted to mention that we have considered in the design, you know, a lot of the questions you've asked, that we do not have to run cables particularly for steam generator eddy current testing, the known types of tests that have to currently be done on plants, that we provide provisions that you can hook that type of equipment up inside containment rather than run the cables right through the equipment hatch.

We also have a design that says you can do most of your pre-staging of all your equipment, you know, moving of equipment in and that type of thing, before you would go to mid-loop operation, such that the equipment hatch can be in place, so it can be closed quickly.

We also have the motors on the trolleys powered off of the battery such that they will close, you know, during the station blackout event.

MR. FINNICUM: Does that answer your question?

MR. CARROLL: I guess so.

[Slide.]

MR. FINNICUM: For the Level II analyses, the process we followed was to define a set of plant damage states by defining plant damage state parameters and then grouping them to develop individual plant damage states based on parameter values, and we quantified the probability of getting a given plant damage state using the Level I information. We then developed our containment event tree and supporting logic models.

Like in the Level I analyses, we used what could be referred to as a small event tree/large fault tree approach. In NUREG-1150, they use rather large decomposition trees. I personally have always had trouble trying to follow them, so I moved to a methodology that I could follow a little easier, where we had a containment event tree that looked at the high level -- the containment failure -- the high level failures, early containment failure, late containment failure, major phenomena that affect the source term, and addresse. the details of the phenomenology in supporting logic models, and then we were able to solve the CET or to quantify the CET for each of our plant damage states.

From that, we were able to define a release class where each end point on the CET was considered to be a unique release class based on how it was defined originally. We could assign a probability to that end point based on the quantification process.

We did not do a full uncertainty analysis on this equivalent to that that was done for NUREG-1150. We basically propagated it through what would be considered mean probabilities. We did perform sensitivity analyses for selected parameters.

[Slide.]

MR. FINNICUM: For the Level III analyses, the risk measure that we were using is basically a dose at a half-mile from the reactor. This is based on the EPRI goal, which is that the probability of exceeding a release of 25 rem at a half-mile from the reactor would be less than 10 to the minus 6th.

We used the MACCS code for determining the dose at the distance. This basically used meteorological data for a bounding site, and this information was provided by EPRI via the URD.

We did not use demographic or population data for calculating the dose at distance, although it was input into the code.

Within the calculation, we assumed that there was no evacuation, and we calculated the cumulative -- complementary cumulative distribution function for the whole body dose at .5 miles and also at 300 meters from the reactor, and again, we performed a set of sensitivity analyses for selected issues.

MR. DAVIS: Let me ask you a couple of questions, if I may.

I couldn't tell for sure if your fire results had been used in the Level III analysis.

MR. FINNICUM: The fire and flood -- the fire, flood, and seismic margin analyses were not propagated into the Level II and III.

MR. DAVIS: Okay. So, these risk results don't include those contributions.

MR. FINNICUM: No, they do not.

MR. DAVIS: Secondly, I couldn't find anywhere a comparison of your results with the NRC safety goals. Is that something you did, or did the staff try to do that?

MR. FINNICUM: We did not specifically do that. The rationale was that the other safety goals are primarily risk to a population. The population risk requires inputting the population data, and what we really had was generic, for no specific site.

MR. DAVIS: Right.

MR. FINNICUM: That's why we did look only at dose at distance and figured that would give an indication.

There are several areas where there might be no change in risk from dose at distance but might have an impact on risk if you looked at some of the other measures, but in looking at what our probability of exceeding the dose at 25 rem at half-mile was, it was very low. We did not feel that we would see any significant change in the other risks.

MR. KRESS: Pete, if you look at their Core Damage Frequency and their Conditional Containment Failure Probabilities, then you can infer that they meet the safety goals.

MR. DAVIS: There is no question in my mind that it meets it. I was just surprised you didn't make the comparison, because that is one of the primary indices of the risk comparison.

MR. EL-BASSIONI: This is El-Bassioni, PRA branch in NRR.

We didn't press for that comparison, because we do not have a full scope PRA including -- as you know, we have done margins analysis, for example, for seismic, and most of the external events were not explicitly calculating Core Damage Frequency, for example, and they have mentioned, many of the analyses and external events were not propagated to Levels II and III. This is why we didn't press very hard for that. But eventually, if we are going to have a living PRA, this comparison will be done.

MR. DAVIS: Thank you.

MR. CARROLL: You said "if." What is the status of the living PRA with respect to the Advanced Light Water Reactors?

MR. EL-BASSIONI: The staff is still working about what's meant by a living PRA, and we hope that there will be a Commission paper within a few months.

MR. DAVIS: But on page 19-25 of the FSER, it states that the PRA is to be revised by the COL to account for site-specific information.

MR. CARROLL: But that doesn't make it living.

MR. DAVIS: As-built info, tech specs, operating procedures, design changes, failure rates, and human errors are to be updated.

MR. CARROLL: That's just an update. That's not a living PRA.

MR. DAVIS: I don't know what your definition of living is.

MR. CARROLL: Through the life of the plant.

MR. EL-BASSIONI: The PRA will evolved till the plant is operational. Then it will be a living PRA, but the staff did not define under what conditions this PRA will be updated. Are we going to have triggers for updating the PRA, or we're going to have periodic, and if it is periodic, what is the span of this period?

MR. CARROLL: Okay. And you're still working on that.

MR. EL-BASSIONI: Yes.

MR. CARROLL: Okay.

[Slide.]

MR. FINNICUM: The next area I want to talk about are the PRA results. The next two slides really go much together. I will talk about them together, but I will present them individually.

[Slide.]

MR. FINNICUM: What I am showing is the Core Damage Frequency contributions by initiating event for our

PRA.

What I show is the original PRA we did, the baseline, the Core Damage Frequency contributions. These were performed to a set of ground rules in effect essentially in 1986.

MR. CARROLL: Whose ground rules?

MR. FINNICUM: These are basically the EPRI ground rules.

MR. CARROLL: Okay.

MR. FINNICUM: And I will address that in conjunction with the slides.

We then upgraded that to a System 80+ configuration based on the design features using the same ground rules. This was performed shortly thereafter, and we have a comparison, and the bottom total for the original System 80 baseline -- we have a Core Damage Frequency of 8 times 10 to the minus 5th per year, and for the System 80+ PRA, performed to the same ground rules, we have a Core Damage Frequency of 6.7 times 10 to the minus 7th. This represents an improvement of about two orders of magnitude, and what we've looked at is how did we get there?

MR. CARROLL: Okay.

Now, going back to the original ground rules, what's Palo Verde's CDF?

MR. FINNICUM: I did not have that. We did not calculate a Palo Verde CDF.

MR. CARROLL: I would guess, if it's like the PRAs on a lot of PWRs, it's another order of magnitude -- well, maybe not. Yes, maybe another order of magnitude higher.

MR. DAVIS: Than what?

MR. CARROLL: Than the 8 times 10 to the minus 5th.

MR. DAVIS: I'd be surprised if it was that high.

MR. CARROLL: It would be higher, though, wouldn't it, Pete?

MR. DAVIS: No, I don't think so. Most of them are coming in around 10 to the minus 4.

MR. CARROLL: Okay.

MR. DAVIS: I don't think one has been done on Palo Verde except for the IPE submittal. I don't whether that's come in or not.

MR. FINNICUM: I'm not sure whether they finally submitted that.

MR. EL-BASSIONI: I don't have any number for Palo Verde.

MR. FINNICUM: They have submitted their IPE. I saw it recently. I don't recall the exact number, but I can get that for you.

MR. CARROLL: Okay. Go ahead.

MR. FINNICUM: There are, across the board, reductions in Core Damage Frequency, and we looked at what are the major design contributors. If you look, we have a fairly substantial reduction in large LOCA.

The things that really contribute to this -- we have the in-containment refueling water storage tank, which eliminates the need for the RAS, the changeover for recirculation. That has always been a problem in existing plants.

The other big item was the four-train ECCS, which provides a high level of redundancy.

Other major contributors also include the reliable power source, the two diesel generators, the stand-by combustion turbine, the grid connections, and the -- that's primarily the impact.

Likewise, for a medium LOCA, we again have a substantial reduction, and it's basically the same thing as the IRWST and the four-train ECCS.

For small-break LOCA, again a substantial reduction. In this case, the four-train ECCS is an important feature, also the four-train EFWS. We do need secondary side heat removal for small LOCA, and we have a very reliable emergency feedwater system.

For the other item, for secondary side break, again a substantial reduction, and again, it's the reliable ECCS, the reliable emergency feedwater system, and also the reliable power systems and the change in that, and likewise for the tube ruptures, and basically, the transients-related items, again it's the -- the impact is the four-train emergency feedwater system and the reliable power system, and for transients, the capability of a diverse means of cooling, the feed-and-bleed cooling capability provided by use of the depressurization valves, helps to make a substantial reduction in the transient.

Loss of offsite power -- again, a substantial reduction. This impact is primarily with the two diesel generators and the alternate AC source, highly reliable feedwater system. We do have six batteries, primarily the four division -- or the four channel batteries and two division batteries, and plus, with the turbine run-back feature and the two switchyards, make an impact on reducing the frequency for a loss of offsite power.

MR. CARROLL: Now, when you make these CDF numbers, do you consider the fact that, during the life of the plant, there are going to be periods of time when, say, one diesel generator is out of service?

MR. FINNICUM: We did include diesel generator maintenance elements in the fault tree models.

MR. CARROLL: And what did you assume for the frequency and duration of maintenance during operation?

MR. FINNICUM: During operation, what we assumed for maintenance unavailability is we looked at the test frequency for the diesels and the assumed failure rate for the diesels and calculated a probability of the diesel having failed a test, and we then assumed the full allowed outage time.

MR. CARROLL: Seventy-two hours or something like that?

MR. FINNICUM: Correct.

MR. CARROLL: Okay. All right. I see what you're doing.

MR. FINNICUM: Again, for ATWS, not quite as substantial reduction but still good reduction.

Inter-system LOCA, we did see a substantial reduction. This reduction is primarily due to a dedicated effort to eliminate interfacing system LOCAs. Primarily, we have high-pressure piping in areas that were subject to the inter-system LOCA problem seen on current plants.

Finally, we have vessel rupture. This is a WASH-1400 carryover. We include it because it was in WASH-1400, and I really can find no valid basis for excluding it at this time.

MR. FRANOVICH: You've probably got a better vessel, though.

MR. FINNICUM: I do have a better vessel, but I cannot statistically justify eliminating it.

MR. KRESS: On your interfacing systems LOCA, the original number comes out that low because of some assumed probability that your isolation valves will work. What did you use for that probability?

MR. FINNICUM: It was based on a calculation using a valve failure rate per hour.

MR. KRESS: That comes out of the standard valve failure rate.

MR. FINNICUM: Yes.

MR. KRESS: That's one of our sore points, using that particular value for interfacing system LOCAs, because those failure rates don't include the blowdown loads.

MR. FINNICUM: These are static valves, and we assume they failed open. Either you blew the disk out or you had a disk failure and did not --

MR. KRESS: Those are the static valves.

MR. FINNICUM: Yes. The primary risk areas are the --

MR. DAVIS: They're normally closed?

MR. FINNICUM: Yes.

MR. WYLIE: On your loss of offsite power numbers, if you assumed that your combustion turbine is not available, would the numbers come out the same?

MR. FINNICUM: No.

MR. WYLIE: What would they come out at?

MR. FINNICUM: I don't know whether I have that.

MR. CARROLL: You have a big section in the back here on --

MR. FINNICUM: Yes.

MR. CARROLL: -- on this issue, Charlie.

MR. WYLIE: Oh, okay.

MR. CARROLL: He's going to get to it, I think.

MR. WYLIE: All right.

MR. FINNICUM: It's in the slide package. I'm going to get to it.

MR. CARROLL: Let me ask about seal LOCA.

MR. FINNICUM: Okay.

MR. CARROLL: You have a dedicated seal injection pump that does not require cooling water or a positive displacement pump.

MR. FINNICUM: Correct.

MR. CARROLL: How long does it take -- now, is that manually placed in service under station blackout conditions?

MR. FINNICUM: It would be manually started from the control room.

MR. CARROLL: And is there valving to be done with it, too? My question is how long does it take to --

MR. FINNICUM: I cannot remember the details of valving, whether they have to open some specific valves or not. We can get back to you on that.

MR. CARROLL: Okay.

MR. FINNICUM: Just one second. Mike? Mike Cross?

MR. CROSS: Mike Cross from ABB.

There would be no valves that would have to be opened. It would be in parallel with the current charging valves that -- charging pumps -- excuse me -- that are with the chemical volume control system. So, you would have two centrifugal pumps, and in parallel with them would be a positive displacement pump.

MR. CARROLL: And all I've got to do is turn it on.

MR. CROSS: That is correct.

MR. CARROLL: So that looks like it's something you could do within 10 minutes. Is that correct?

MR. FINNICUM: Yes.

MR. CARROLL: Okay.

MR. DAVIS: I think, originally, that CE's contention was that those seals wouldn't fail anyway, and this was added, I think, mostly because of the staff's concern over this seal LOCA. Isn't that true?

MR. FINNICUM: This is true. It's still CE's contention that the seals would not fail under those conditions, but we do have the back-up pump.

MR. CARROLL: And it can be powered by either the diesels or the AAC.

MR. FINNICUM: Correct.

MR. CARROLL: Okay. Our interest in this is that GI-23 is still kicking around out there, and the last time the issues resolution guys from RES came down, they were arguing that, at least on a Westinghouse plant, unless you got the back-up system in service within -- I think it was 10 minutes, the front of hot water would reach the seals, and the ball-game was over, mechanistically, at least. You had a 440-gallon-a-minute leak on each pump that you couldn't recover from.

MR. FINNICUM: I understand.

MR. CARROLL: We sort of said, you know, tell us about this probabilistically, and well, no, that's the way it is. So, it sounds like you've come up with a solution to this.

When we met with the Germans and French last year, their solution is basically a little different. They don't -- the French have tried a pump, a turbine-driven pump, and found that it had some problems, because the steam generator pressure keeps changing, and it affects the performance of the pump.

I guess their approach now is a mechanical design to their reactor coolant pumps, so that, on loss of power and the pump stops, it comes down and back-seats so that you can't have seal leakage.

MR. RITTERBUSCH: This is Stan Ritterbusch. We were aware of that design option. We maintained that our original seal design was adequate and didn't need extra protection, and there was quite some debate about that. We have a different seal design than Westinghouse plants --

MR. CARROLL: I know.

MR. RITTERBUSCH: -- and the charging pump was simply an added margin of security.

MR. CARROLL: Well, you were getting bagged in with Westinghouse the last time they resolved GI-23, because you had not presented convincing information on that.

MR. RITTERBUSCH: We have always been in that bag. We have tried to get out, and unfortunately, we haven't succeeded yet.

MR. CARROLL: Okay.

[Slide.]

MR. FINNICUM: The second slide that I talked about -- what I present here is now, again, the Core Damage Frequency contribution by initiating event, comparing the System 80+ Core Damage Frequency as we performed the analysis under the original ground rules and the System 80+ Core Damage Frequency as performed under the current ground rules.

The bottom line is, in changing the ground rules, our Core Damage Frequency went from 6.7E to the minus 7th up to 1.7E to the minus 6th, and what were the ground rule changes? In the original analyses, check valve common cause failure was excluded, was assumed to be not credible. Additional thought and some additional information indicated that it may be a credible failure and should be included. So, we did include common cause failure for the check valves.

MR. CARROLL: Which check valves are important in this regard?

MR. FINNICUM: I included all check valves, the EFW, the safety injection, the containment spray, the RHR.

MR. CARROLL: So, you just waved a magic wand and --

MR. FINNICUM: That's it. If I've got to include them, I might as well include them all.

MR. CARROLL: -- and made them not function.

MR. FINNICUM: Yes.

MR. CARROLL: Okay.

MR. FINNICUM: The second major item is, in the original --

MR. CARROLL: Is that not function in the sense of not opening?

MR. FINNICUM: Not opening.

MR. CARROLL: Okay.

MR. FINNICUM: The second major item was, in the original analyses, a number of the operator error rates were calculated using the original EPRI human cognitive reliability model.

That model has fallen into disfavor, primarily because when you look at short action time or the short amount of time needed to perform an action and a relatively long period of time in which to perform the action, it produces some patently absurd results.

The decision made was to go back to the primary methodology using the Swain & Guttman handbook, the '84 version, and it produced -- it did increase a number of our operator error rates.

In addition, another change was made. This did not have a significant impact, but it was a change made.

For the valves inside containment, such as the safety depressurization valve, they are tested at an 18-month interval. We had been using the generic failure rate, and there was a concern that there may be a time-dependent element for the failure rate.

So, for the valves inside containment, they were tested at 18-month intervals. We back-calculated an hourly failure rate and recalculated a new demand failure rate based on the test envelope.

MR. DAVIS: On your loss of offsite power contribution, you used a number of 3.5E to the minus 2 as the initiating event, and I was wondering where that came from.

MR. FINNICUM: That number was provided by EPRI.

MR. DAVIS: Okay.

MR. FINNICUM: They had a calculation for the loss of offsite power, which really was defined to be a loss of offsite power such that actuation of the emergency diesels was required. It required loss of the main switchyard, loss of the secondary switchyard, and failure of the turbine generator to run back and pick up the hotel loads.

MR. DAVIS: That's not what your event tree says. Your event tree has this as the lead-in, and then you consider the switchyards and the turbine generator run-back.

MR. FINNICUM: That's included in the calculation of the initiating event frequency for loss of offsite power.

MR. DAVIS: If that's the case, it looks like you've taken credit for it twice. I'll have to check that again.

MR. FINNICUM: Let me see if I have a copy of my event trees here with me then.

MR. DAVIS: Let me see if I can help you find it.

MR. FINNICUM: I have a copy of the loss of offsite event tree. I do not have a slide for this. The events I show across the top are called loss of offsite power --

MR. DAVIS: Right.

MR. FINNICUM: -- then failure to deliver emergency feedwater, then failure of long-term decay heat removal, failure of safety depressurization for bleed, failure of safety injection for feed, and then failure of long-term containment heat removal.

MR. DAVIS: Are you looking at Figure 19.4.8-1?

MR. FINNICUM: I believe that's what it would be. Mine is not labeled as a figure. Let me come over there. Yes, that's the equivalent.

The calculation of this frequency for the loss of offsite power includes the element of loss of the site power, the original grid loss, the conditional probability that we would lose the second switchyard given the first one was gone, and then a conditional probability that we would fail to run back and pick up hotel load.

MR. DAVIS: If you go through that, you get a number of 5E to the minus 3. I'm still saying that your original loss of offsite power number is 3.5E to the minus 2. Why don't you go ahead, and I'll find where that is?

MR. FINNICUM: Okay. The calculation of the frequency for loss of offsite power is in Section 19.3, and there is a summary table right towards the end of the chapter, and that shows the input.

MR. DAVIS: My point was that number looks awful low, and I was wondering what kind of assumptions went with it, whether this is a sustained loss of offsite power, because you don't have a recovery factor in the event tree, and I couldn't find a definition.

MR. FINNICUM: The recovery factors -- the recovery of power are included as recovery actions against the individual cut sets. The 3.5E to the minus 2 was an EPRI number that they calculated a probability of losing offsite power, and I think it's for greater than 60 seconds or something like that, based on operating experience data.

MR. CARROLL: They have accumulated a hugh database --

MR. FINNICUM: Yes.

MR. CARROLL: -- from the industry.

MR. DAVIS: We did not do any additional evaluation of those numbers. We just used the EPRI calculation.

MR. DAVIS: Okay. Thank you.

MR. SIEGMANN: Eric Siegmann.

I'd like to point out, a single loss of the switchyard today, of a single switchyard, is considered like 8E to the minus 2 for a single switchyard based on EPRI data. So, it doesn't look particular low to me.

MR. DAVIS: Well, again, the 3.5E to the minus 2 was used as the entry event. Then you considered the loss of the other switchyard, and you gave that a .36 probability of failure.

MR. FINNICUM: Correct.

MR. DAVIS: That was considered later, not as part of the initiating event, and then your run-back unavailability was a 3-to-2 split, and you went on through the event tree.

MR. FINNICUM: Yes.

MR. DAVIS: I haven't looked at that EPRI database, but this number is considerably lower than most PRAs use. As you may recall, WASH-1400 used one loss of offsite power every five years, .2 instead of .035, and then that number gradually dropped down to about .1, and that seemed to be where most people were coming out.

MR. FINNICUM: The NRC had performed some evaluations several years ago, and their evaluation also came out with, I believe, a .07 value --

MR. DAVIS: Right.

MR. FINNICUM: -- in the calculation, and the EPRI value is --

MR. DAVIS: Above that.

MR. FINNICUM: Yes. We do have a sensitivity study talking about the sensitivity to the loss of offsite power frequency in there, and I believe we used a value of .15. Matter of fact, I'm getting to these, the sensitivity studies.

[Slide.]

MR. FINNICUM: For Level 1, we also did a number of sensitivity studies, and the base case that we're talking about was the 1.7E to the minus 6 as our Core Damage Frequency.

The first sensitivity study we did is we basically increased all of our operator error rates by a factor of 10. That's one order of magnitude. With that change, the Core Damage Frequency increased to 9E to the minus 6, an increase of about 5.

MR. CARROLL: So that says we don't have to spend all this money training operators or paying them as much as we do. They can screw up and it's no problem.

MR. FINNICUM: Probabilistically or realistically?

MR. DAVIS: They're one and the same.

MR. CARROLL: Catton should have been here to hear that.

MR. FINNICUM: The second sensitivity study we evaluated --

MR. CARROLL: These aren't cumulative. They are --

MR. FINNICUM: No. They're individual sensitivity studies.

MR. CARROLL: Yes.

MR. FINNICUM: Okay.

We basically increased all of the motor operator valve failure rates and the common cause -- associated common cause failure rates by a factor of 10, and in this case -- I missed one. I'll go back to this. For the change in the MOV failure rates, the Core Damage Frequency increased to 8.5E to the minus 6.

Back to the second one, we did a second operator action study where we set the human error probability or the operator failure rate for actions performed outside the control room -- these were set to 1.0. In other words, he could perform no action outside the control room. That did not have a significant impact. It's basically up to 2E to the minus 6th.

The primary reason is we do not have many actions in the PRA that need to have action taken outside the control room. Most of the actions we credited were things that would be taken inside the control room.

MR. CARROLL: Okay. Now, all of this that you're describing is exclusive of shutdown risk.

MR. FINNICUM: This is exclusive of shutdown risk.

MR. CARROLL: This is another category. Okay.

MR. FINNICUM: Correct. These are the basic at-power Level I sensitivity analyses.

MR. CARROLL: Got you.

MR. FINNICUM: Another issue where we looked at the modeling issue -- in this case, we used the large LOCA as the safety injection tank injection requirements for a medium LOCA, did not have a significant impact at all.

MR. CARROLL: What's significant about that? What's the issue here?

MR. FINNICUM: The number of tanks required.

MR. CARROLL: Okay.

MR. FINNICUM: Secondly, for small LOCA and steam generator tube rupture, if we have these events which are small loss-of-coolant-type events and if we fail the injection system, the high-pressure injection system, if the operator reacts within sufficient time, we can cool the plant down and depressurize it to a pressure where we can align and use the shutdown cooling pumps for injection, and this was credited in the model.

This sensitivity analysis basically said assume that we cannot do that, that that is not a feasible operation. What would be the impact on the Core Damage Frequency? And the Core Damage Frequency increased to about 7 times 10 to the minus 6th. So, it's a factor of about 3 or about 4.

We did specifically do a sensitivity study for the reactor coolant pump seal LOCA, assuming that it could occur, as a sensitivity study, and the probabilities were low enough that they did not have a significant impact on the Core Damage Frequency. They were in the 10 to the minus 8th range.

We did another one -- this is an unusual one, and what we wanted to see is, if we assume we did no test or maintenance in-power and if we had no maintenance unavailability, what would be the impact, and we would expect to see some decrease in the Core Damage Frequency. We really did not see any significant decrease.

MR. CARROLL: What kinds of maintenance and testing were you --

MR. FINNICUM: We looked at all of the safety system tests and maintenance.

MR. CARROLL: That are in the standard tech specs?

MR. FINNICUM: The ones that were specifically modeled in the fault tree, the diesel generator test and maintenance unavailability, maintenance unavailability on like the safety injection pumps and the shutdown cooling pumps or the containment spray pumps. Anything where a pump would fail during a surveillance test and have to be maintained, we set those to zero. We did not see a decrease.

MR. CARROLL: Got you. I understand. Thank you.

MR. FINNICUM: Another issue we looked at is, for ATWS, if the moderator temperature coefficient is sufficiently positive when the ATWS occurs, the peak pressure will exceed the Level III values, and we will have what is assumed to be an unmitigatible LOCA.

For our plant, we calculated a value such that the probability of having an adverse MTC was about .01 or 1 percent of core life.

The sensitivity study we performed was -- assumed that, in fact, we had -- 10 percent of the time the MTC was adverse, that we would have an unmitigatible LOCA during that timeframe, and the Core Damage Frequency increased to 2.2E to the minus 6th.

MR. CARROLL: Didn't we learn last month, though, that the fuel design basis is to have no period of time when the MTC is positive?

MR. SEALE: Yes.

MR. FINNICUM: Yes. These original analyses were performed a number of years ago, and we maintained them as probabilistic.

MR. DAVIS: Mr. Chairman, while he's getting the next slide, I was provided some information on Palo Verde's Core Damage Frequency by Dean Houston from the IPE. Their result is 9E to the minus 5 for internal events.

MR. CARROLL: Which doesn't look too out of line with 8.1.

MR. DAVIS: No. It's about a factor of 50 higher than the internal events from System 80+ results of 1.7E to the minus --

MR. CARROLL: 8.1 is the original ground rules for 80.

MR. DAVIS: But the current number now is 1.7E to the minus 6th. I'm reading from --

MR. CARROLL: For Palo Verde?

MR. DAVIS: No, no. For System 80+.

MR. CARROLL: Well, I'm looking at the --

MR. FINNICUM: Let me go back to that slide.

MR. CARROLL: -- original ground rules, System 80, Pete.

MR. DAVIS: I'm talking about the current ground rules.

MR. CARROLL: Oh, okay.

MR. DAVIS: That's the current number, is the current ground rule.

MR. CARROLL: Yes, you're right.

MR. DAVIS: It's 1.7.

MR. SEALE: That's for System 80+.

MR. DAVIS: That's right.

MR. CARROLL: I just wanted to get calibrated on how the -- the starting point, the System 80 original ground rules, compared.

MR. DAVIS: Oh, okay.

MR. CARROLL: Okay.

[Slide.]

MR. FINNICUM: Again, on this slide, I repeated the base frequency, so we could refer to it. The LOOP frequency -- again, we increased that by a factor of 10, and we did not really see a large increase in overall Core Damage Frequency.

We set the loss of grid frequency -- that would be that basic input value, the 3.5E to the minus 2 value -- I jacked that up to .15 per year, and the overall Core Damage Frequency increased to 1.8E to the minus 6th.

Another sensitivity study I did is let's assume vesse_ rupture was not a credible event. What would the impact be? And it dropped our Core Damage Frequency to 1.5. Basically, we took one sequence out.

MR. SEALE: Fifty percent faster.

MR. FINNICUM: Yes.

MR. CARROLL: Well, I think, in all cases, though, you have to make the point that the 1.7E minus 6th is a very low absolute number.

MR. FINNICUM: Yes.

A couple of other ones we did were looking at what was the impact of the various common cause failure rates. We did one where we set all the common cause failure rates to zero, in other words assumed there was no such thing as common cause except for esels and for the batteries, and in this case, the Core Damage Frequency had a significant drop. It dropped from 1.7E to the minus 6th to 2.4E to the minus 7th. Basically, it shows that common cause failures are important to the risk for the plant.

MR. CARROLL: As modeled.

MR. FINNICUM: As modeled, yes, and as the data is used for calculating them.

Finally, just combining the two above, we said let's assume that, except for batteries and diesels, we have no common cause failure and assume vessel rupture is not credible. What would be the bottom line? And as expected, it's 1.4E to the minus 7th.

MR. CARROLL: What were you thinking when you decided to exclude diesels and batteries?

MR. FINNICUM: There is a large body of information on testing and operation of the diesels, and you have pretty good evidence of common cause failure there. So, there's no reason to even think about excluding that.

MR. CARROLL: Again, mentioning our meeting with -- our quadripartite meeting which included the French and Germans, they are, on the European pressurized water reactor, going to four diesels diverse.

MR. FINNICUM: Yes.

MR. DAVIS: Wait a minute. That sounds like an oxymoron. Four diesels -- you mean four different kinds of diesels?

MR. CARROLL: No. Two of one kind and two of another.

MR. DAVIS: Oh, okay. Two-by-two diverse.

MR. CARROLL: Two-by-two diverse.

MR. FINNICUM: By diesels diverse, two of one kind and two of another kind, do you mean they are two different types of diesels or just two different manufacturers?

MR. CARROLL: Different manufacturers.

MR. FINNICUM: Okay. Well, I do have slides on the particular issue of two diesels versus four diesels at the end of the slide package.

MR. CARROLL: Looking forward to it.

Let's see. We now move into shutdown?

MR. FINNICUM: Yes.

MR. CARROLL: Shall we eat lunch and come back for shutdown?

MR. FINNICUM: That's acceptable to me.

MR. CARROLL: All right. Why don't we reconvene at 1:05?

[Whereupon, at 12:06 p.m., the meeting recessed for lunch, to reconvene this same day, at 1:05 p.m.]

AFTERNOON SESSION

[1:05 p.m.]

MR. CARROLL: Let's reconvene. We're on shutdown risk.

[Slide.]

MR. FINNICUM: The next slide here presents the results of our shutdown risk evaluation and presents a comparison to the three other existing shutdown risk PRAs.

The bottom line -- we have a Core Damage Frequency for shutdown of 8E to the minus 7th, and it's pretty well balanced across the four areas we looked at. Loss of DHR contributes about a quarter of it, LOCA is 16 percent, loss of offsite power contribution is about a quarter, and fire is about a quarter. It is about an order of magnitude less than the comparable studies.

MR. CARROLL: Why do you think that's true?

MR. FINNICUM: Again, we have a large degree of separation, a lot of redundancy, and based on the mechanistic evaluation we had, we actually put in some features and procedures, and those will come out in the next slides.

MR. CARROLL: And what the others with respect to NSAC-84 and Seabrook?

MR. FINNICUM: Eric, could you address what the other elements are?

MR. SIEGMANN: I'm Eric Siegmann. I don't recall what the others were.

MR. SEALE: That's bad.

MR. LINDBLAD: You just eliminated 15 minutes of presentation time.

MR. CARROLL: I guess I would like to know what they are. Can we get a response to that?

MR. FINNICUM: Yes.

Eric, we'll need to get a response on that.

[Slide.]

MR. FINNICUM: Specifically, some of the design features that help contribute to reducing the shutdown risk is we will have two safety injection pumps operable. We do have the capability to inject to the RCS via the shutdown cooling system.

We have a safety depressurization system that we can use for feed-and-bleed cooling even in the shutdown mode. Our containment spray pumps can back up our shutdown cooling pumps. They can be used interchangeably. The IRWST does act as a sump for a LOCA.

Again, we have the alternate AC source to back up the diesels. We have a dedicated shutdown cooling system, as opposed to the old LPSI system that doubled as a shutdown cooling system, and we have put in place technical specifications for shutdown cooling or for shutdown modes.

MR. CARROLL: Do those include a requirement that two of the three diesel generators plus the alternate AC -- two of those three have to be available --

MR. FINNICUM: Correct.

MR. CARROLL: -- at all times during shutdown or when you're in mid-loop?

MR. FINNICUM: Eric, can you address that specifically?

MR. SIEGMANN: At all times in shutdown.

MR. FINNICUM: At all times in shutdown.

MR. SIEGMANN: Eric Siegmann. All times in shutdown, two of the three.

[Slide.]

MR. FINNICUM: As a direct result of the mechanistic evaluation that was performed for shutdown risk, we did add additional instrumentation.

We added two additional Delta P based narrow range RCS water level indicators, and we added two heated junction thermocouple based RCS water level and temperature indicators.

In other words, the same instrument performs both a level and a temperature function, but this instrumentation was added specifically based on the mechanistic evaluations.

MR. CARROLL: And the heated junction thermocouple instrument that you're talking about here is different than the one that's part of LOCA protection?

MR. FINNICUM: Fred Carpentino can address that specifically.

MR. CARPENTINO: There are two types of heated junction thermocouple strings in the upper head region.

One is the one we had for LOCA and other design basis events, and what we've added is two more strings that have higher resolution for level measurement in the mid-loop regime.

I have an overhead that shows the locations of the indications of the heated junction thermocouples if you'd like to see that.

MR. CARROLL: Not so much the locations but --

MR. CARPENTINO: There are two different types.

MR. CARROLL: Okay. And are these operable for LOCA conditions, for example? Do they add to what you already had?

MR. CARPENTINO: Yes, they do. They're pedigreed to the same degree as the original two strings.

MR. CARROLL: Okay. Thank you.

[Slide.]

MR. FINNICUM: Also as a direct result of the shutdown risk evaluations, we've made changes to more than 20 of our technical specifications to address our shutdown modes, and some examples of these changes -- one, we require two shutdown cooling divisions to be operable and at least one of the divisions is in operation in Mode 6.

We have two safety injection trains with one pump in each division is required to be operable in Modes 4, 5, and 6, and during reduced inventory operation, the containment is required to be closed, and there are a number of other changes, including the two of three that you mentioned earlier.

MR. CARROLL: I would have thought that would have been one of your more important examples.

MR. FINNICUM: It's one I overlooked on the slide,.

[Slide.]

MR. FINNICUM: Procedural guidance was also developed for our shutdown operations based on our evaluations, and these fall into the areas of reduced inventory operations, we have guidance for coping with the loss of DHR, how to detect and mitigate RCS drain down events, outage maintenance during shutdown.

MR. CARROLL: You mean management.

MR. FINNICUM: No, it's outage maintenance management.

MR. CARROLL: Okay.

MR. FINNICUM: Fire protection, the use of feed-and-bleed for RCS cooling during shutdown operations. These procedures are discussed in the report that's in Appendix 19.8A. They are also captured in Appendix B of our System 80+ emergency operating guidelines. They were extracted from the PRA area and put in the operations guidelines.

[Slide.]

MR. FINNICUM: For external events -- the tornado strike Core Damage Frequency contribution is 2 times 10 to the minus 7th. This number was calculated the same way that we calculate Core Damage Frequency for internal events, with full fault tree linking.

MR. DAVIS: Excuse me. Let me ask you a couple of things about that.

I noticed in the tornado CDF estimate, you assumed that you could not run the turbine back to pick up house loads. Why was that assumption made?

MR. CARROLL: Yes, I saw that, too.

MR. FINNICUM: I don't remember what the basis was on that.

MR. DAVIS: It wouldn't make a big change, but I was just wondering why you made that assumption.

MR. CARROLL: Probably sucked all the water out of the condenser.

MR. FINNICUM: Eric?

MR. SIEGMANN: Your turbine building is not -- the turbine building, the qualifications of the turbine building might be the reason.

MR. FINNICUM: Yes. I think what we're looking at is there was a potential for loss of condenser vacuum.

MR. MICHELSON: Yes, but how do you run back the turbine if you lose offsite power and you don't have on-site power yet? It takes a while to get it -- 10, 15, 20 seconds, whatever. How do you run back and keep control and bring the turbine up at the lower --

MR. DAVIS: They assumed they couldn't do it.

MR. MICHELSON: Yes, I assume that that's probably right.

MR. LINDBLAD: The turbine is running.

MR. MICHELSON: The turbine was running, but it's now tripped.

MR. CARROLL: No, no, no, no, no.

MR. MICHELSON: I thought the tornado --

MR. FINNICUM: On the tornado, we just assumed it could not be done.

MR. MICHELSON: This is the non-tornado case.

MR. WYLIE: If you assume the building destroys, it takes out your generator breaker and everything else.

MR. MICHELSON: Well, yes, if that happens, then you don't have it either, but I wasn't even going to speculate that. I just thought the tornado was the case.

MR. CARROLL: They're designed to go back and carry house load.

MR. MICHELSON: You mean they can go power-off, total blackout for 20 seconds and bring the turbine back up.

MR. CARROLL: There's no blackout, because the turbine is supplying the house loads.

MR. MICHELSON: But when you lose the offsite power, that thing is going to trip off.

MR. DAVIS: They run it back.

MR. MICHELSON: I don't think you can run it back. As soon as that thing starts winding down, those breakers are going to open.

MR. CARROLL: When the breakers open, it's going to go on the governor.

MR. WYLIE: Which breaker?

MR. CARROLL: The offsite breaker.

MR. WYLIE: You're getting a tornado strike --

MR. CARROLL: Yes.

MR. WYLIE: -- in the building.

MR. CARROLL: No, no, no, no, no. It has wiped out offsite power.

MR. WYLIE: Oh, okay.

MR. CARROLL: Then the unit just runs back and picks up house load.

MR. WYLIE: What's it hit, the switchyard?

MR. CARROLL: Probably.

MR. WYLIE: Your breaker is out in the switchyard.
MR. CARROLL: Yes.
MR. MICHELSON: I think it's gone.
MR. CARROLL: The breaker is open.
MR. WYLIE: And the unit is gone, too.
MR. CARROLL: Why?
MR. WYLIE: Because it shorts. If it hits the switchyard, it's going to short out the switchyard and your breaker and everything else.
MR. CARROLL: No, the breaker is going to trip.
MR. WYLIE: If a tornado hit a switchyard, there's nothing going to be intact.
MR. FINNICUM: We did not credit turbine run-back for a tornado.
MR. DAVIS: That was the original question, right?
MR. FINNICUM: Yes.
MR. DAVIS: And I think I know why now.
MR. FINNICUM: There are a number of issues.
MR. CARROLL: But a more general tornado case is that it takes out offsite transmission.
MR. WYLIE: If it is outside the switchyard.
MR. CARROLL: Then the breaker opens. It will back down and carry house load.
MR. WYLIE: If it rings the plant and it doesn't get your switchyard, then that's true.

MR. CARROLL: Has that ever happened, Charlie?

MR. WYLIE: Oh, yes, at Dresden.

MR. CARROLL: I know. I was just kidding.

MR. FINNICUM: For fire and flood, we made a scoping estimate calculation. We did a qualitative evaluation both in Chapter 19 and also in Chapter 9 looking at the separation, and the basic conclusion of the qualitative evaluation is, because of the high degree of separation, fires and floods should not pose a significant risk.

We did a scoping evaluation --

MR. MICHELSON: Now, let me understand why tornado is an order of magnitude higher than the fire. What's the tornado probability to begin with?

MR. FINNICUM: I cannot remember the specific --

MR. MICHELSON: What's the fire probability to begin with? We've got to start somewhere and kind of get a feel for where we lost a few factors.

MR. FINNICUM: The fire probability is higher than the tornado.

MR. MICHELSON: I would hope so.

MR. FINNICUM: It is. The big difference on the tornado strike is, one, we cannot recover offsite power for a period of 24 hours.

MR. MICHELSON: Okay.

MR. FINNICUM: Two, we don't have the alternate AC source. It was assumed that that would be gone. And three, it was also assumed that there was a potential for a vulnerability in the service water intake structure that might potentially take out both divisions.

MR. MICHELSON: What probability is that that's related to a tornado?

MR. FINNICUM: We used a value of .01.

MR. MICHELSON: It's a tornado-qualified building and all that good stuff. It's all designed for a tornado.

MR. FINNICUM: What we looked at is the vulnerability was a potential for trash getting thrown into the intake structure and clogging it up.

MR. MICHELSON: It's designed for that, isn't it?

MR. FINNICUM: It should be, and it's a caveat there, but in the analysis, we assumed a .01 probability that we would -- that we could fail that.

MR. MICHELSON: Okay. In spite of the design, it still failed.

MR. FINNICUM: Correct.

MR. MICHELSON: You could pick up a little that way.

MR. DAVIS: Which is true for everything.

MR. MICHELSON: I can go back in fire and start playing those games, too, and I'll get you way on up there.

It escapes me why those two are inverted.

MR. FINNICUM: As I said, the basic conclusion from the qualitative analysis is that, because of the high degree of separation, it should not be a risk.

We did a scoping analysis. Basically, we calculated from generic sources a fire initiation frequency and assumed that the fire took out one entire division and then quantified the loss of CCW models to estimate a Core Damage Frequency.

MR. DAVIS: I had a little trouble with the way you did that. In your analysis, it looked like you assumed that a fire occurred, then you had a factor that the fire went through some fire barriers in the same division, before you got the total division wiped out.

MR. FINNICUM: That was one specific type of fire. A fire in the diesel room would not cause a transient.

MR. DAVIS: Well, go ahead. I'll find what I'm talking about.

MR. FINNICUM: There was two types of fires, a fire that would occur in a room that could disable safety equipment and potentially cause a transient and one that would occur in a room that would disable safety equipment but probably would not initiate a trip, specifically the diesel generator room, and we assumed that we had a 10 to the minus 3rd probability of going through the barrier and getting to some vital equipment that would cause a transient.

MR. DAVIS: This occurs on page 19.7-31 and 32, and you may want to go back and take a look at that. It looked like you used this fire barrier for all fires, both type A and type B, which include switchgear, as well as diesel generator, battery room, and so forth, and this failure probability of the barrier is quite low, 1.2E to the minus 3, but go ahead. You may want to take another look at that.

MR. FINNICUM: Okay. We will take a look at it. That was our basic event where we took out one division.

We also looked at what would happen if we assumed there was a potential for a fire-induced seal LOCA, what the impact would be.

MR. CARROLL: What does that mean?

MR. FINNICUM: The fire tock out all seal cooling and seal injection on one division and we had a subsequent failure. We then would assume that, given those conditions, that we could have a seal LOCA.

Again, remember, the design -- CE's design contention is that the seals would not fail. In this case, we assumed that they could fail under these conditions. What would be the contribution?

MR. MICHELSON: The seal cooling system is divisionalized?

MR. FINNICUM: Yes. The cooling is provided by the CCW pumps. Each division of cooling water cools two RCP seals.

MR. MICHELSON: Two seals meaning two of the four pumps.

MR. FINNICUM: Two of the four pumps. So, if we lose CCW in one room, we will lose seal cooling to those two pumps. We still have seal injection from the charging system. The fire is assumed to take out one of the charging pumps. We also have the conservative assumption that the seal injection pump would be in the division that was affected by the fire and it would be gone.

MR. MICHELSON: Isn't the CCW required for the safety injection or CVCS or the equivalent of it?

MR. FINNICUM: It's required for the charging pumps --

MR. MICHELSON: Yes.

MR. FINNICUM: -- but not for the seal injection pump.

MR. MICHELSON: There are no coolers or anything related to it and no cooling of the motors or the pumps.

MR. FINNICUM: Correct.

MR. MICHELSON: Okay.

MR. FINNICUM: But we still have the other charging pump.

MR. MICHELSON: It's apparently all air-cooled.

MR. CARROLL: Yes.

MR. FINNICUM: The dedicated seal injection pump is an air-cooled pump.

MR. MICHELSON: And no coolers in the fluid system of it. Okay.

MR. FINNICUM: Correct.

MR. CARROLL: Except it's just one pump.

MR. FINNICUM: But we assumed that was failed anyway.

MR. MICHELSON: They assumed it was failed anyway. It was just trying to figure out what the arrangement was. I thought he would have had two, not just one.

MR. FINNICUM: We also did a similar analysis for the -- the total fire scoping analysis was about 6E to the minus 8th. We did a similar type analysis for the flooding, the basic cases where we assumed a flood would take out one entire division, and modeled the other one.

MR. MICHELSON: Excuse me. Now, on fire, since you're coming up with definitive numbers, I guess you didn't use the FIVE methodology, you used a true PRA, a true fire PRA.

MR. FINNICUM: No.

MR. MICHELSON: Well, how did you get numbers?

MR. FINNICUM: That was the methodology I was explaining. We calculated a fire initiation frequency based on generic data.

MR. MICHELSON: Yes. That doesn't tell you how you respond. That just tells you how often you get the fire.

MR. FINNICUM: We then assumed that we lost one entire division.

MR. MICHELSON: That's okay, but that may not be a good assumption, depending on where the fire is and so forth.

MR. FINNICUM: It's a reasonably conservative assumption, though.

MR. MICHELSON: It depends on where the fire is, whether it's in the control complex end of the business or back in the other end. Of course, you're also assuming no interactions, and if you do a fire PRA, you're supposed to pick up the interaction effects.

MR. FINNICUM: The qualitative analysis showed we did not have the interactions because of the divisional separation.

MR. MICHELSON: What does this number mean, then? You relate it, so far, quantitatively, only to the fire frequency.

MR. FINNICUM: Yes, it's a scoping estimate of where will we be. If we did a FIVE methodology, what we would do is we go into our individual fire rooms --

MR. MICHELSON: I think that we all understand the way it works, and you don't get numbers from it, but you do get a number here, and it's related to fire frequency.

MR. FINNICUM: In the full FIVE, FIVE goes into the quantitative valuation.

MR. MICHELSON: I won't get into that now.

MR. FINNICUM: Okay. There is accepted methodology.

MR. MICHELSON: You're just claiming you have an extremely low fire frequency and therefore you have a low risk from fire.

MR. SALTOS: Excuse me. This is Nick Saltos from NRC again.

I don't think that the frequency is low, and I don't think the result depends on the frequency of the fires that is assumed.

The assumption is that the integrity of the divisional separation will be maintained during a severe fire. That is a major assumption. The next step was to look how this integrity could be defeated.

MR. MICHELSON: But there are areas of the plant where you do not have divisional separation in the electrical area. Obviously, when you get to the control room itself, you really don't have true divisional separation, unless you want to talk about an inch being divisional separation. It's certainly not a concrete wall.

Now, you can buy some of these arguments out where you've got big heavy concrete walls, and I think they are valid. They become much less valid when you get into the --

MR. SALTOS: It does not assume a fire in the control room.

MR. MICHELSON: That's a zero probability.

MR. SALTOS: No.

MR. MICHELSON: If it's an assumption there are none, that means there is no probability of fire.

MR. CARROLL: I think they're saying there is no probability of a fire in the control room that would cause core damage.

MR. MICHELSON: Yes, but that's only an assumption.

MR. DAVIS: That's right.

MR. MICHELSON: They haven't done a PRA.

MR. SALTOS: For a fire in the control room, you would have to use the remote shutdown panel, and if you assume a set of probabilities, say 10 to the minus 2 or 10 to the minus 3 per year, then you have to see what is your probability of transfer to the remote shutdown panel, and then you have a plant trip and then you have a transient, and you look what are your safety functions and how you can access those safety functions so you can shut the plant down.

MR. MICHELSON: Which division is the back-up panel in?

MR. FINNICUM: It's in Division 1 or 2.

MR. CROM: This is Tom Crom. The remote shutdown panel is located in Division 1, but you can operate both divisions.

MR. MICHELSON: You mean they both come together at that point, electrically? You can't control them unless you bring wires and whatever in.

MR. CROM: It's all fiber optic into that particular area.

MR. MICHELSON: Pure fiber optic only.

MR. CROM: Yes.

MR. MICHELSON: That's interesting. How do you power this division, this backup panel? Only one division of power?

MR. CROM: I'm definitely not the person to answer that. We need to ask --

MR. MICHELSON: We'll get into it later on when we get to back-up control.

MR DAVIS: Before we leave the fire, I think it's important to qualify these results.

In Volume 20, in section 19.7.3, on page 31, the statement is made that "A quantitative assessment of the risk of internal fire cannot be made at this time, because detailed design information for cable routing and fire detection and fire suppression systems is not presently available," but they go ahead and attempt to produce this number, and they call it a conservative scoping estimate.

Now, the staff, in their FSER, says -- and if I'm characterizing it incorrectly, please correct me -- that they are not convinced it's conservative because of the things that were left out and the unknown information.

I happen to agree with that. I do not think you can demonstrate that this number is conservative. It may, in fact, go up, because some non-conservative assumptions were made, and one of them was that you don't have any contribution from control room fires. You also assumed that this divisional separation was 100-percent in place and worked always, there was no probability of failure for that, and I don't personally think that your fire initiating frequencies are conservative. I'm not sure where they came from, but they don't look conservative to me.

So, I think what it really says is that the COL has to redo this fire part when this information becomes available. So, I don't think it's worth worrying about the numbers at this point.

Now, have I said anything you disagree with?

MR. FINNICUM: No.

MR. DAVIS: Okay.

MR. MICHELSON: I think there's another thing you need to account for. If you're going to put the back-up control controlling both divisions from one location and the fire is near the back-up center, then you have to worry about spurious actuations in both divisions caused by heating up this room, because you probably aren't going to be able to condition the air in it anymore and so forth, and it's going to get hot.

MR. CARROLL: But in that case, the control room is still available.

MR. MICHELSON: Well, I don't know what the scenario is. I didn't put the fire in the back-up control. I said nearby somewhere such as to elevate the temperature. It's only one of the two divisions, and the fire will be in that division.

Now, you have to start chasing heating and ventilating and a lot of other things before you can fully understand what a fire might do. That's what a PRA, hopefully, does. It models the whole thing correctly. Only a good PRA would do that, of course, but if you model it correctly and you model environmental interactions correctly, then you'll pick the whole thing up out of a PRA and you've got believable numbers. Otherwise, I'm not sure what you believe out of all this subjective argument.

MR. SALTOS: This is Nick Saltos again from NRC.

We don't take these numbers to be accurate. I think what we got out of that is that the integrity of the divisional wall is very important, and the objective was to see what has to come through, what kind of requirements has to be served.

MR. MICHELSON: Remember, in the case of ABWR, we actually have two back-up control rooms, one for each division. Now, they're side by side and there is a divider wall between them, but you don't try to bring all the wiring and all the control and everything into one location in one division and claim that you're okay for controlling both divisions from it. It gets pretty argumentative after a while.

MR. CROM: Mr. Michelson, this is Tom Crom again. You know, you mentioned the possibility of fire near the area of the remote shutdown panel. In that particular fire, the remote shutdown panel is not energized. It's disconnected.

MR. MICHELSON: You have to look at the scenario to see whether there was any connection between that and the viability of what was coming from the main control room to begin with.

Now you've got to look at the routing of all the fiber optic cables going up to the main control room to find out whether you might have had to abandon the main control room because of the fire, not because it was chasing you out, because you lost control.

So, you've got to look to see where the fire is, and then you look to see if the back-up panel is sufficiently divorced from the fire to carry you on through, and that's the name of the game.

MR. CARROLL: Well, assuming that you can make the argument that the main control room is not affected by a fire in one or the other divisions, then I guess I think I agree with Tom that you're home free.

MR. CROM: The only time the remote shutdown panel will be energized is if there is a fire in the control room, and they are physically and electrically isolated from each other.

MR. MICHELSON: Well, if you go back and look at the history of fires, you'll find that has not been the history either. They have had to use back-up control panels for a couple of big fires that we have had, partly because a portion of the main control room capability was lost and they had to go to the back-up control panel to regain that capability, and that's exactly what they did. They didn't abandon the main control room, but they used both, in essence. I didn't realize this was a common, shared back-up control room.

MR. CARROLL: Okay. Moving on.

[Slide.]

MR. FINNICUM: This slide --

MR. CARROLL: Let's see. You passed over the seismic margin one, because you covered it yesterday.

MR. FINNICUM: Yes.

MR. CARROLL: Okay.

MR. FINNICUM: I don't even recognize it, because they're out of my package. I apologize.

This slide -- we had a comparison of the average of some IPE results. It basically shows what the average IPE Core Damage Frequency is as compared to the current System 80+ value and a relative breakdown of where the risk contributions are.

For System 80+, our big contributions are in the LOCA area, and this includes both the large and medium LOCAs, and in the transients area where, as you see, for a IPE, you see a large contribution from loss of site power, loss of coolant, and transients, and a lot of that shift is due to the design activities for System 80+.

MR. MICHELSON: What is a loss of site power?

MR. FINNICUM: It's, in actuality, a loss of offsite power. In the IPEs, it was called loss of site power.

MR. MICHELSON: Oh, I hadn't heard that one. I'm way behind, I guess. I didn't associate that with loss of offsite. I thought this was some new kind of a speculation about on-site.

MR. FINNICUM: The reason I kept that acronym is that was what was used in the report that we had.

MR. MICHELSON: I understand what it is now.

MR. FINNICUM: Okay.

[Slide.]

MR. FINNICUM: The Level II results that we looked at -- the primary area we were looking for from here was our containment performance, and there was no real clear definition of what constitutes containment performance. So, we looked at the various elements of 90-016 and came up with three different potential definitions and basically what we're looking at here.

What we're saying is, if we define containment failure as above-normal releases within the first 24 hours, then our containment reliability is .098 or the unreliability is .02.

If we look and say containment failure is defined as having a large release or a release greater than 25 rem at a half-mile, then our containment reliability is .097 or the unreliability is .027. That's fairly close.

And if we say that the containment failure is defined as any breach whatsoever of the containment, then the containment reliability is --

MR. CARROLL: At any time --

MR. FINNICUM: -- at any time --

MR. CARROLL: -- after the accident.

MR. FINNICUM: -- then it's .886, and this is, in general, .9, and it's consistent with the goal of .1 that was stated in 90-016.

MR. CARROLL: That last number is dominated by basemat melt-through.

MR. FINNICUM: Correct.

MR. BRINKMAN: This is Charlie Brinkman. Just for the record, since this is on the transcript, you misspoke some of those numbers, Dave, and I'd appreciate it if you'd go back over them again. You had some decimal places off. Simply for the transcript.

MR. FINNICUM: Okay.

The first item is, if containment failure is defined as failure with above-normal releases within the first 24 hours, containment reliability is .98, containment unreliability would be .02.

MR. CARROLL: Actually, the slides are part of the record, if I'm not mistaken. So, I don't know that you really need to do this. Do they get in if you buy a transcript?

MR. DAVIS: Besides that, I think he's corrected the only mistake he made. He said .098 before.

MR. CARROLL: All right.

[Slide.]

MR. FINNICUM: A little more detail on that. In looking at what we call an intact containment for 24 hours is 96.5 percent, this consists of a containment intact indefinitely. That's our 88.6 percent value. Late containment failure due to overpressure is .4 percent of containment failure probability.

MR. CARROLL: Now, included in that is preventing overpressure by opening the vents?

MR. FINNICUM: No. I did not credit opening those vents.

MR. CARROLL: Okay.

MR. FINNICUM: I also did not credit use of the reactor fan coolers.

MR. CARROLL: Okay.

MR. FINNICUM: Late containment failure due to basemat melt-through is 7.5 percent.

Our containment isolation failures are 2.4 percent, and early containment failure constitutes 1.1 percent with -- the dominant portion of that is the ex-vessel steam explosion, which contributes .95 percent or, essentially, about 80 or 85 percent of the early containment failure probability.

The alpha mode failures is about .12 percent. The hydrogen burn or detonation constitutes about .03 percent, with a subtotal of 1.1 percent. DCH had a very small contribution and did not show up, and this is our total of 100 percent.

[Slide.]

MR. FINNICUM: Again, we performed a set of sensitivity analyses for the Level II analyses, and I present the -- the base case, I show the containment intact indefinitely value, then late containment failures aggregate, early containment failure aggregate, and isolation failure aggregate. These are in percent.

We looked at if we assume the hydrogen igniters are not available, the containment intact probability goes down slightly, late containment failure comes up slightly, early containment comes up, almost doubles because of the increased probability of hydrogen burn.

The second sensitivity study we looked at, assuming that the deflagration to detonation transition was more likely, that we were more likely to have a hydrogen detonation as opposed to a burn, the underlying assumption is, given a hydrogen detonation, the probably of containment failure is 1.

So, by increasing the deflagration detonation would increase containment failure probability, it really did not have a significant impact in the round-off error.

MR. CARROLL: Is that third number supposed to be 1.1 or .1?

MR. FINNICUM: Yes, it should be 1.1 instead of .11.

The next sensitivity study we looked at is assuming a lower heat transfer rate from the corium to the coolant. The impact of this is that you will ablate through the -- the concrete will be ablated sooner even in the presence of cooling. The intact containment probability went down, late containment failure probability went up, and again, the early containment failure was not affected, because this is primarily a late containment failure phenomena.

We looked at reducing the probability of containment spray recovery in the late case. What we looked at is, for the cases where sprays were unavailable initially, as part of this analysis we had credited recovery of containment spray late, after 24 hours.

We reduced the probability of doing that recovery and looked at the impact. The containment intact indefinitely went down, late containment failure again came up on overpressure. The early failure was not impacted, because the sprays were unavailable in the first 24 hours. Isolation toggle effects were based on how we structured the event tree.

A second item we looked at is no possibility of recovering any of the heat removal of the containment. The difference between this is we looked at recovery of the spray system, mechanically. Here we looked at -- there was three different factors -- recovery of offsite power late or use of the back-up spray system or recovery of the spray system itself, and we basically just turned that off and took a look at what would happen to containment. Again, this had a large increase on the late containment failure. This is all overpressure, steam overpressures.

We also looked at some of the things that impacted DCH. This is primarily thermally induced failure of the RCS piping during a high pressure sequence, and we've assumed either they always occurred or that it never occurred, and because of the very low impact of DCH on our containment failure probability, it really had no impact on the sensitivity studies.

[Slide.]

MR. FINNICUM: The next case we looked at is, for high-pressure sequences, we assumed that we could not depressurize with -- the RCS with the safety depressurization system, and again, we see no impact, primarily because DCH did not have a big contribution.

We also looked at assuming that the safety depressurization valves were 100-percent reliable and would depressurize all the sequences. Likewise, again, no significant impact, because it's primarily DCH-related.

We also looked at a sequence where we would increase the containment isolation failure rate for the general sequences to a 10 to the minus 2, and again, that did not affect the early and late primarily. It did have the impact on our isolation failure rate.

We also looked at -- terrible spelling here -- we also looked at depressurization of the RCS using the depressurization valves for medium and high pressure sequences.

The sequences we looked at with respect to RCS pressure are the 2,500-pound sequences, which are cycling relief valve, and the high-pressure sequences are those that are in the 1,000- to 2,000-pound range, medium pressure are in the 400- to 1,000-, 1,200-pound range, and there was originally assumed to be some potential for DCH in those sequences if we did not depressurize.

So, we looked at assuming that we could not depressurize further and see what the impact was. Again, no significant impact, because DCH was not a significant contributor.

The final one we looked at was failure of the operator to turn on the hydrogen igniters. We increased the failure rate by an order of magnitude.

There was a slight increase in the early containment failure probability due to an increase in the containment failure due to hydrogen burn.

[Slide.]

MR. FINNICUM: For the Level III PRA, as I mentioned earlier, the risk measure we looked at was the probability of exceeding a whole body dose of 25 rem at a half-mile, and the value we came up with is 5.3E to the minus 8th as the probability of exceeding 25 rem whole body dose at a half-mile. The goal was 1 times 10 to the minus 6th. So, we're well within the goal.

We also looked at -- we considered a small site, a 300-meter reactor boundary, and we got a 6.2E to the minus 8th probability exceedance for that.

MR. DAVIS: If you went past a half-mile, would that number go up or down?

MR. FINNICUM: I believe it goes down. I can't remember the exact curve.

MR. DAVIS: Well, the thing I'm concerned about is, if you have an energetic release, you can jump over the near-site people, and I don't know whether you have any contributing sequences that have an energetic release when the containment fails.

MR. CARROLL: A la Chernobyl.

MR. DAVIS: Well, it's a plume rise phenomena. You know, WASH-1400 showed that, that you can jump over the people within a mile or two miles, and then the plume comes down beyond that.

MR. FINNICUM: Stanley, do you have a comment?

MR. RITTERBUSCH: We understand that phenomenon. It turned out to be not dominant. At one point, we thought we were concluding that the people ought to run into the plant, but on an overall basis, that turned out not to be the case.

MR. MICHELSON: Well, if you fail the containment, you end up venting into the annulus, and the annulus is directed upward to the top of the containment and then out through some process. Doesn't this give you a higher level release and a somewhat energetic ejection out the top of this -- from the annulus?

MR. DAVIS: It's usually the thermal energy that causes the lofting.

MR. MICHELSON: Yes, but it's all hot gases, and it's all being vented up to the top of the containment, and I'm not sure what the routing is from there to the normal vent point.

MR. PALLA: This is Bob Palla with the staff.

I just wanted to mention that the bulk of the risk for this design is dominated by steam generator tube ruptures, and these would not have the kind of energies released that you're thinking of.

There is still a contribution from early failures from events like DCH, small probabilities, but I believe the bulk of the risk and frequency, as well, will not. I'd have to think about that further, but these should be dominated by the events such as steam generator tube rupture with lower energies released.

MR. DAVIS: Loss of feedwater is a big contributor, almost 30 percent to the core damage frequency, and steam generators is also a large one.

MR. FINNICUM: The tube ruptures -- when you look at the risk, the probability of exceeding 10 to the minus 6th --

MR. DAVIS: Is from tube ruptures because you bypass -- yes.

MR. FINNICUM: You bypass. You have no scrubbing.

MR. DAVIS: Right. Thank you.

[Slide.]

MR. FINNICUM: We also had looked at several different sensitivity studies.

One of the things we did is we looked at assuming all releases occur at the top of containment, where you'd get -- we looked at taking the releases all at the top of containment and seeing what would happen, and at the half-mile, the doses -- the probability of exceeding 10 to the minus -- or 25 rem went down, and that's probably due to the effect of the plume being distributed over a wider area.

We also looked at the releases occurring all at grade level, and the dose at a half-mile -- the probability of getting 25 rem at a half-mile increased slightly to 5.4E to the minus 8th, and this is the effect of the ground hugging plume there and the shadowing of the reactor building.

MR. MICHELSON: What is the release from the relief valves, then? What do you consider that? It's certainly not ground level, and it's certainly energetically directed upward, generally, but I suspect the plume is well beyond the top of the containment, even, because when those valves go off, they shoot a plume way up. That would be the steam generator tube rupture case.

MR. DAVIS: The core melt occurs after that.

MR. MICHELSON: That's true, yes.

MR. FINNICUM: Those valves are also in the valve house.

MR. MICHELSON: Yes, but they're directed upward and outward, I hope. You're relieving into the room?

MR. FINNICUM: Tom, can you address that?

MR. CROM: The main steam safety valve -- when they relieve in the valve house, they direct it outside or they relieve inside and then exit.

MR. CROM: The main steam valve house or the safety valve is directed outside the main steam valve house, yes.

MR. MICHELSON: You can blow the house apart if you aren't careful.

MR. CARROLL: It might be a little tough on people setting valves.

MR. MICHELSON: I don't think you'd want to work around them either. That all has to be outside and upward.

MR. SEALE: But as I understand it, the release height is an input into the meteorology part of the calculation, and what he's done here is just say it's all released on the ground, let's see what the meteorology does from there.

MR. FINNICUM: That's what I did in this sensitivity study.

MR. SEALE: You neglected all of that, for whatever reason.

MR. PALLA: Just a point of clarification not on the sensitivity but on the baseline calculations. I believe a number of 19.7, approximately 20 meters elevation was the release for steam generator tube ruptures.

MR. MICHELSON: That's the elevation. Isn't that about the elevation of the valve. That's not necessarily the elevation of the plume it creates. Those things are normally directed upward.

MR. FINNICUM: The calculation was based on the elevation of the valve.

MR. PALLA: The roof of the main steam line, main steam valve room.

MR. MICHELSON: Yes, that's the correct number for that.

MR. CARROLL: That's where it's released, but it jets up beyond that.

MR. FINNICUM: The third sensitivity study we looked at, we just basically increased the release fractions for iodine and cesium by one order of magnitude. What we say is the release -- or the probability of exceeding 25 rem went from 5.3E to the minus 8th to 6.4E to the minus 8th, which is not unexpected.

The next item we looked at is assuming that all of our containment bypass released were unscrubbed. The basic -- no change in the probability of exceeding 25 rem, in part because most of the releases were already unscrubbed.

Next we looked at increasing our containment isolation failure rate by one order of magnitude. In other words, we had more likely a direct release, and as expected, we saw a large jump in the probability of exceeding 25 rem.

We looked at doubling the basemat melt-through failure frequency to see what the impact was. The releases are not large releases on this. So, the probability of exceeding 25 rem did increase but not by a large amount.

We also looked at the concrete ablation failure, i.e. the basemat melt-through. Our calculations show that the time of failure, of a release would be at about 65 hours. We cut that in slightly less than half to see what the potential impact would be. Not a significant increase in the probability of exceeding 25 rem.

Likewise, we increased our ISLOCA frequency by two orders of magnitude, and that is a bypass release, and we went from 5E to the minus 8th to 1E to the minus 7th for probability of exceeding 25 rem.

MR. CARROLL: Now, you are going to move into the ACRS questions?

MR. FINNICUM: Correct.

MR. CARROLL: Can we do them in reverse order, since we're going to lose two members, and I think they're more interested in issue two than in issue one.

MR. FINNICUM: Okay.

MR. CARROLL: And then we'll pick up issue one.

MR. DAVIS: Is this the four diesels?

MR. CARROLL: Yes.

While you're getting ready, one question on the total PRA picture. Did Combustion consider going out and getting an independent review of their total PRA work, other than the review the staff provides?

MR. FINNICUM: We did not do that. During the first System 80+ -- using the original ground rules, the Level I analysis was reviewed, not in a great deal of detail, but was reviewed by Tenera Corporation.

MR. CARROLL: Okay.

[Slide.]

MR. FINNICUM: The issue here was asked about -- it's the two diesel generators plus the alternate AC source versus four diesel generators issue and what impact does that have on core damage?

MR. CARROLL: It's a two-by-two four diesel generator. So, you don't have to consider common mode failure.

MR. FINNICUM: We looked at this issue back in 1991. We did a sensitivity study. We looked primarily at unavailability of the diesel generator configuration. We did not include the support systems in the model. We were looking purely at the diesel generators, combustion turbine, and the busing availability, and we looked at two configurations.

For LOCA, we need to have two HPSI pumps. For transients, we would need to have only one HPSI pump.

Now, that has the most impact on the four-diesel configuration, and what we looked at is, for the configuration we had with two diesel generators with a gas turbine, that the unavailability for the transients was 1.4E to the minus 4th per demand, for LOCAs it was 2.7E to the minus 3 per demand, and if we looked at the unavailability of the diesel generators, for transient, the unavailability was about 1.3E to the minus 4th demand, which is a lower unavailability than for the two-diesel and the gas turbine.

For the LOCA, again, it was 1.9E to the minus 3, which is, again, a lower unavailability than we showed for the two-diesel with a gas turbine, but they were not significant differences, and to put in four diesel generators would have a significant effect on the design of the plant.

We would have had to have added two additional component cooling water systems and service water systems, and it would have had a significant impact on the footprint of the plant.

MR. CARROLL: Why?

MR. DAVIS: I thought these were air-cooled diesels.

MR. CARROLL: No, they're not.

MR. DAVIS: They aren't?

MR. FINNICUM: These are water-cooled diesels.

MR. DAVIS: But they could be air-cooled.

MR. FINNICUM: That I cannot address. Tom?

MR. CARROLL: They could be air-cooled.

MR. DAVIS: There are diesels this big that are air-cooled.

MR. CARROLL: Yes.

MR. WYLIE: What size diesels are we talking about here?

MR. FINNICUM: Tom?

MR. CROM: As I recall, the diesel size is 6.4 megawatts.

MR. WYLIE: This is per division?

MR. CROM: Per division.

MR. WYLIE: For two divisions.

MR. CROM: For two divisions. That's correct.

MR. WYLIE: So, what is it for four? Half of that?

MR. CROM: It should be essentially half, that's correct.

MR. FINNICUM: It would be slightly more than half.

MR. WYLIE: They could be easily air-cooled.

MR. CARROLL: Yes. Diablo is air-cooled, and they're bigger than that.

MR. DAVIS: In fact, it would be, I think, advantageous, because you would remove this dependency, and also, it would be cheaper, I think.

MR. FINNICUM: That was our 1991 study, and we looked at some potential costs, but the main shot was, looking at water-cooled diesels, that the impact would have been major on the design under that consideration.

MR. WYLIE: The size of the diesel that you say is $25 million there, is that a half-size or a full-size?

MR. FINNICUM: Eric?

MR. SIEGMANN: That's full-size.

MR. WYLIE: That's full-size. All right. Are we comparing apples and apples or apples and oranges? I'm not sure about your numbers.

MR. CARROLL: Now, what happens if you do go diverse, two of one kind and two of another, and let's just, for the sake of argument, say that common cause failure has been eliminated.

MR. FINNICUM: If you say, for the sake of argument, that common cause failure is eliminated by selecting different manufacturers, you will see the benefit of this increase. How much I don't know, but it will increase, because this addresses the common cause with four diesels.

In the past, we have not considered different manufacturers as being diverse. Part of that is based on the assumption --

MR. CARROLL: The staff has told us they would consider that diverse.

MR. FINNICUM: In years past, in other arenas -- for example, the reactor trip switchgear -- we were told that different manufacturers of switchgear did not constitute diverse.

MR. LINDBLAD: Of course, the gas turbine, while it's not safety-grade, is a diverse power supply.

MR. CARROLL: Okay.

Now, this assumes -- or does this assume that, if I have one diesel out of service for more than 72 hours, I'm going to have to shut the plant down, because I don't meet GDC-4, I'm vulnerable to a common cause failure, unless the staff has, since we last talked to them, decided to give you credit for --

MR. CROM: This is Tom Crom. The answer is the tech specs still requires the plant to be shut down if a diesel is out for more than 72 hours.

MR. CARROLL: Okay.

MR. SEALE: Even if there are four of them?

MR. WYLIE: No, no, no. He's talking about two.

MR. ARCHITZEL: This is Ralph Architzel from the staff. We're responding to your letter on that point at the moment right now.

MR. CARROLL: Do you know what the response is going to say?

MR. ARCHITZEL: Basically, I believe the applicant hasn't requested any relief from the standard tech spec, and we're not going to pursue it independently.

MR. CARROLL: Okay.

Does the applicant intend to --

MR. ARCHITZEL: That's prejudging. It's with the technical staff. It's got management review, but that's where we're heading right now.

MR. CARROLL: Okay.

Do you plan to ask for such relief?

MR. RITTERBUSCH: This is Stan Ritterbusch. Yes, we are going to have a discussion. I can't predict where it's going to end up, but as a result of the comments and interest expressed here, we are going to go back and ask that question as we go through the final stages of the technical specification confirmatory effort.

MR. CARROLL: Okay. I would say to you that the $43 million, or whatever it is, isn't very much money. If I were the guy buying this power plant for Podunk Light and Power, I could use up $43 million in 60 years in down time costs in a big hurry.

MR. MICHELSON: Is it $57 million for the two diesel plus -- is that plus gas turbine, the $57 million?

MR. FINNICUM: I believe so.

Eric, could you answer that?

MR. SIEGMANN: Yes.

MR. MICHELSON: It is the combined cost and all the insulation of them and so forth.

MR. SEALE: According to that, then, I assume, when I buy four diesels, I get four seismically-qualified diesels, but when I buy two diesels plus a gas turbine, I get two seismically-qualified diesels plus a non-qualified gas turbine.

MR. CARROLL: Well, I'd say it another way. I'd say plus a gas turbine that has a HCLPF of .36 g.

MR. SEALE: Okay.

MR. CARROLL: I was wondering about that number. What happens if the gas turbine is running when the earthquake happens?

MR. FINNICUM: If it is running when it happens, I'd have to think about that a minute.

MR. DAVIS: You mean in a test mode?

MR. CARROLL: For whatever reason.

MR. DAVIS: Or is the plant down?

MR. CARROLL: No. Maybe you're running it for peaking.

MR. FINNICUM: We would not be running it for peaking.

MR. CARROLL: Why not?

MR. FINNICUM: Because it's not hooked to the switchyard.

MR. CARROLL: Yes, but I can displace loads in the plant.

MR. FINNICUM: Oh, okay. I understand.

MR. SIEGMANN: This is Eric Siegmann with Combustion Engineering.

I'd like to point out -- we discussed four trains with two diverse with another utility. I'd like to point out that, when you go to two different vendors on diesels or on any part, whether it be HPSI pumps or what, you're doubling your spare parts, you're doubling your procedures, and your training, and my personal opinion is you're going to be doubling your maintenance errors, okay?

It is not clear to me that manufacturing defects and common cause failures associated with the manufacturing or installation of a single vendor is going to be overcome by the increase in maintenance errors associated with having two different vendors on the site.

Your pre-existing maintenance errors tend to dominate your failure rates, and I suggest you had better look carefully before you suggest that you double your inventory of component on a site, as the British have done at a cost of 50 percent more for the BNFL plant, and it's a questionable decision.

MR. CARROLL: One of your problems, though, is although, practically, you know, I believe a lot of what you say, these PRA rascals keep hitting me with this stupid beta factor that I can't overcome. They say just trust me, this is the right number, and the only way to defeat them is to say I don't have to use your dumb number, I'm going to have diverse equipment.

MR. CROM: This is Tom Crom from Duke Engineering, and I guess I'm trying to put my two cents in, too, on what, really, common mode failure is, because what I've seen, at least at Duke plants, common mode failure does not really occur just because of different manufacturers.

Unless you would go all the way to having different fuel sources, different governors -- I mean it would have to be strictly more than just different manufacturers to be totally diverse.

MR. CARROLL: Oh, I agree, but I don't think it's that hard, if you really wanted to, to make it that way.

MR. CROM: We'll include even probably different maintenance crews into that.

MR. CARROLL: I have yet to confirm it, but I now understan[illegible] the major airlines flying 767s across the drink are using different maintenance crews on the two engines. They must know something.

MR. DAVIS: I had a question on the previous slide. Why are the unavailabilities different for transients and LOCAs?

MR. FINNICUM: For LOCAs, we need to -- for a large LOCA, we need to have two HPSI pumps, and that impacts the number of buses and support systems within that arena. It has the same impact on the four diesels.

In this case, we'd need to have -- two of the four diesels would have been operating for a LOCA. Here we'd need one of the two diesels, but more buses would have to be available.

MR. DAVIS: Okay. For the transient, you just need one of four diesels?

MR. FINNICUM: Yes.

MR. DAVIS: That failure rate looks awful optimistic if you consider common cause failures.

Go ahead.

MR. CARROLL: Well, what I really want to see is some realistic credit given for the AAC. I think you and the staff have got to work out something so you get some realistic credit for it.

MR. DAVIS: Aren't you seeing it here in this slide? Showing the fourth diesel doesn't improve things over the two diesels with the combustion turbine.

MR. CARROLL: If you believe it, but the staff is saying, until Combustion comes to them and says I want credit beyond 72 hours, they are going to have to shut down --

MR. DAVIS: Okay.

MR. CARROLL: -- and I think you need credit for the gas turbine, because I think it's going to do its job. You may want to put some restrictions on how many diesel generators you can have out of service when the tornado warnings are up or something like that.

MR. ARCHITZEL: This is Ralph Architzel again from the staff. We did credit the gas turbine in the tech specs for the reduced inventory mode. So, it is in there at that point.

MR. CARROLL: That's correct.

MR. ARCHITZEL: And there is also a difficulty with CE's design in that they only have the two divisions, as opposed to GE that had the three. So, there is some more difficulty with CE's design.

[Slide.]

MR. FINNICUM: The benefits -- I've really covered the benefits and the detractions.

MR. CARROLL: The first bullet under the first bullet, did we learn that you were saying that two additional full-size diesels far exceed the cost of one gas turbine?

MR. WYLIE: That's what they're saying.

MR. FINNICUM: That's apparently what the costs are.

MR. CARROLL: What's the cost of four half-size diesels versus two full-size plus a gas turbine? That's the comparison I would make.

MR. SEALE: You could buy the same number of megawatts in a diesel.

MR. FINNICUM: I cannot answer that question.

MR. CARROLL: Okay. So, the costs on the preceding slide are for the case I just talked about, not for the case I'm interested in. Okay.

[Slide.]

MR. FINNICUM: In 1992, an additional sensitivity study was done. This one was performed by Duke Engineering Services for the British Nuclear Fuels, and it looked at -- essentially at risk or Core Damage Frequency with a smaller model.

They looked at loss of offsite power, small LOCA, large LOCA, and general plant trip as initiators within a smaller model, and their conclusion is that the two diesel with the alternate AC source had a lower CDF than four diesels without the AAC. Let me show you the number.

[Slide.]

MR. FINNICUM: The number here -- they show that, for their model, that the Core Damage Frequency was 1.7E to the minus 6th per year with two diesels and an alternate AC source. For the four diesels without the alternate AC source, it went up to 2.4E to the minus 4th, which probabilistically speaking is essentially the same number.

The diesel generators again were dominated by the common cause failure both of the diesels and of the support system. These models did include support systems. For the base case, the two diesels with the AAC, it looked at our current support system design, in general. For the four diesels, they looked at having two more with component cooling water systems. It did not look at air-cooled diesels.

The real conclusion is, in looking at the numbers, I would say that there is not a big difference, there is just essentially no difference between the two, and that there is some cost differential in going to the four diesels.

MR. CROM: This is Tom Crom again. I want to touch a little bit on that cost number.

One, I want to point out, one, that the combustion turbine is twice as large as the diesel generator. We're talking about a 15-megawatt combustion turbine.

MR. CARROLL: That's correct.

MR. CROM: The other thing, those numbers -- you know, you're talking about, essentially, you could have half-size diesels, but if you look at the cost, where that number is really driven is more in building costs and concrete. You don't get that mu^h reduction going to a smaller diesel. So, even if you have a half-size diesel, the number is not going to be that much different.

MR. CARROLL: Plus more switchgear.

MR. CROM: That's correct.

MR. SEALE: Unless you're able to go to air-cooled, which cuts down on the peripherals, I would think.

MR. CROM: That would come down even on the two diesel generator case, but still, those cost numbers are going to be based on concrete and your building costs more than it is on your diesel generator and your support systems.

MR. CARROLL: Your two diesels are just getting up in the size range where I'm not sure it is practical to have air-cooled.

MR. SEALE: But four --

MR. CARROLL: Definitely you can air-cool.

MR. WYLIE: That's right.

MR. CARROLL: Okay. We know what the issue is.

MR. WYLIE: Just one other thing, though. Will these numbers change if this plant were sitting at Turkey Point and you had Andrew hit it?

MR. DAVIS: Which numbers?

MR. WYLIE: Well, I don't know. I'm talking about the overall numbers here.

MR. CARROLL: You mean the risk numbers.

MR. WYLIE: The risk numbers, yes.

MR. DAVIS: Because of the combustion turbine not being hardened?

MR. WYLIE: Well, the had combustion turbines down at Turkey Point that were wiped out. They had all their lines wiped out for a week. They were down to two diesels, and they labored on that. They lost one of those during that time and they relied on one. I think that a plant sitting in that particular location, to get hit by a hurricane, is very vulnerable.

[Slide.]

MR. FINNICUM: To pick up the last set of slides, the last issue, it was the issue on the failure rate used for failure of the MOVs to close, especially close in the EFWS system. I understand this was Mr. Catton's question, and he's concerned that the failure rate we used, the base failure rate, was too optimistic.

MR. CARROLL: Do we know where Carl is?

MR. SEALE: He got pulled out by Med for some reason.

MR. FINNICUM: The failure rate we used was 4E to the minus 3. It was based on generic data from the key assumption ground rules, and it was basically for valves tested on a quarterly basis.

We understand Mr. Catton's contention is that, for failure of the MOVs to close -- and I understand, it's probably in the blowdown situation -- they really should be much higher, 8E to the minus 2.

MR. CARROLL: At least for some valves.

MR. FINNICUM: At least for some valves.

[Slide.]

MR. FINNICUM: What I basically did is -- we wanted to take a look at a sensitivity study of what would be the impact on Core Damage Frequency. We looked at really two cases.

The first case, we increased the failure rate for all MOVs for fail to close failure mode from 4.0E to the minus 3 to 8E to the minus 2 to take a look at the impact, and our Core Damage Frequency went from 1E to the minus 6 per year up to 3E to the minus 6 per year, a factor of about 2.

For the second case, we increased the failure rate for all EFW valves for all modes, both failed open and fail to close, up to 4.0E to the minus 3 to 8E to the minus 2.

The base Core Damage Frequency went from 1.7E to the minus 2 up to 4E to the minus 6 -- 1.7E to the minus 6 per year to 4E to the minus 6 per year, which is an increase of about 2 1/2. So, it would have some impact on Core Damage Frequency but not a large impact.

[Slide.]

MR. FINNICUM: Some of these other things that I've prepared a slide is based on the operating conditions. I now have a fuller understanding of what his concern was, and it was the blowdown loads, and at this point in time, this really does not address that issue.

[Slide.]

MR. FINNICUM: Now, what we did is we had looked-- we used a generic Core Damage Frequency or generic valve failure rate. We had looked at other available industry sources in selecting that, and the value we selected was within range of what we see in current PRAs and other sources. We did not have access to the 8E to the minus 2 number, and that's why we used the 4E to the minus 3.

MR. DAVIS: I think that number is based on some German data that --

MR. SEALE: Yes, that's right.

MR. DAVIS: -- Dr. Catton had obtained.

MR. FINNICUM: I could not find that in the sources, and we looked at U.S. -- we had some access to

Swedish data, and their generic data -- and this is for valve fails to change position, and it's in the same realm as what U.S. and generic --

MR. MICHELSON: That's the numbers you're using for valves that have to operate under blowdown kind of conditions?

MR. CARROLL: Yes.

MR. FINNICUM: Yes. We used the 4E to the minus 3.

MR. MICHELSON: You have no database for that except the few tests that have been run.

MR. FINNICUM: Yes.

MR. MICHELSON: Is this reflecting those few tests?

MR. FINNICUM: Not that I know of. We used generic data, and what I've shown is --

MR. MICHELSON: My question was did you use generic data for blowdown conditions? The answer has to be no, because there isn't any.

MR. CARROLL: No. They used the EPRI data.

MR. MICHELSON: They used the generic, no loaded --

MR. CARROLL: 4E to the minus 3.

MR. MICHELSON: -- and that's in the right ballpark for that, but it's clearly not right for valves under duress.

MR. DAVIS: You'd better go back to the other slide. They raised it considerably to see what the effect was.

MR. MICHELSON: You mean they raised it and it showed no effect?

MR. DAVIS: A minor effect.

MR. CARROLL: Look at the Case 1/Case 2 slide.

MR. MICHELSON: Which parts of the PRA? You know, like auxiliary feedwater would be one case.

MR. FINNICUM: We did two things.

MR. MICHELSON: Your plan is probably in fair shape from this viewpoint, because you don't have many energetic lines to begin with outside of containment, and furthermore, they have a limited impact when they do fail. You may not have a problem with it, but that doesn't make your numbers right. It just means that --

MR. FINNICUM: I can't say that. I used the generic number consistent with the data sources I had available, and if we changed it, I can see what the impact is.

MR. MICHELSON: I'm not even sure anymore how good these generic numbers are.

When people start -- and I'm sure Duke will confirm all of this. When they started looking at their valves closely, they realized that there were many, many valves that were way out of adjustment, and as a consequence, the failure rate was -- in the generic data - was not reflective of what was really there.

When they went back and fixed it up, I'm sure, if anything, the generic numbers are now conservative, after the readjustment and so forth had taken place, but that's the uncertainty in these cases of isolating breaks. We don't have enough data to know what the impact is, but the generic numbers probably aren't too bad now, once they got all the valves tuned up, but they weren't too good before. They were way off before.

MR. CARROLL: How about the fail to open case for the depressurization valves? How bad would that hurt you?

MR. FINNICUM: For the depressurization valves, I believe we already addressed that in --

MR. CARROLL: The MOVs are open already.

MR. MICHELSON: It depends on which ones you're talking about.

MR. FINNICUM: For the safety depressurization valves in the RCS?

MR. CARROLL: Yes.

MR. FINNICUM: Those are -- inside containment, both the block valves and the isolation valves have to open. The failure rate we used there is based on an 18-month test interval. So, it is higher than the 4E to the minus 3. It's somewhere in the 10 to the minus 2 range.

MR. MICHELSON: With a testing interval of whose valves? Testing interval means nothing.

MR. CARROLL: He's put in a cheat factor for the fact that the valves are only tested every 18 months.

MR. MICHELSON: That's okay.

MR. CARROLL: That's for aging effects.

MR. MICHELSON: That doesn't help the orders of magnitude that your reliability numbers may be off to begin with. It just makes a small correction in a big error.

The valves that have to open under full differential pressure have been just about as bad as those that have to open under blowdown or close under blowdown conditions, and again, they found that, in many cases, they just would not have hacked it if they'd had to, but they fixed it and now they should. So, the problem went away only because it was -- I think because the valves have been properly adjusted now.

MR. CARROLL: Well, I believe there are some valves out there that are just mis-designed that are going to have to be replaced. They don't have a big enough operator on them.

MR. FINNICUM: There are some very glaring cases of that. In the early days, San Onofre I found out that their -- one of their injection valves -- the motors couldn't open the valve at all, and they had to go back in and replace them with a bigger motor.

MR. MICHELSON: I think the generic database -- maybe we're beginning to get to the point where maybe it's not so bad now for these cases, but it certainly was way off in the past, and we just didn't realize it.

MR. CARROLL: We're in the category of things that Combustion perceived ACRS had some questions about, Carl. We've dealt with the MOV issue, and while you were out, we discussed the two EDGs with AAC.

MR. MICHELSON: I heard some of that.

MR. CARROLL: Have we got any other issues?

MR. FINNICUM: That concludes the slide show I have.

MR. CARROLL: I know. I know. This jogs me to say is there anything else we want Combustion to do some analysis?

MR. MICHELSON: Other than the things we've asked for during the meeting?

MR. CARROLL: Yes.

MR. MICHELSON: Namely the pressurization analysis, for instance, on the auxiliary feedwater. They were going to supply that.

One thing I have kind of an uneasy feeling about -

- I think they've got to go back and sharpen pencils a whole lot on their flood analysis to make it believable. I'm not sure that there's really a problem, per se. It's just not yet believable, because they haven't done it in sufficient detail. I think, once they do and catch all the little things, you know, things like the leaking seals where you're going out into an area exposed to a site flood, things like that, once they get those things taken care of, I think they're okay. They've promised to fix all those that we talked about.

MR. FINNICUM: Mr. Michelson, we do have in the PRA, in 19.15 now -- one of the insights is that, on the final design, the COL will go back in and factor in the site-specific and final design information, and it specifically calls out look at the external events.

So, I agree. These were scoping estimates only. We wanted to look at a number, and that's why I don't call them the Core Damage Frequency.

MR. DAVIS: I think your main purpose was to see if you had any obvious vulnerability that you'd overlooked, and for that purpose, I think it was probably adequate, and the question is now is it really worth doing anymore till you have the plant, and I guess I don't think it is.

MR. CARROLL: Okay. I guess what we have left are some remaining questions from yesterday. Let's see. Who had questions?

MR. COE: I think one question that was left over from December was the question that appears on page 69 of your handout. This was Mr. Wylie's question. Maybe we've already addressed it.

MR. SEALE: Over-voltage protection?

MR. COE: Yes, over-voltage protection, and I don't think we have discussed that yet. I know we've looked at it, you've looked at it, but we haven't discussed it in committee yet.

MR. MICHELSON: What page is it on?

MR. COE: Page 69 of the handout that I provided to you, not the staff handout.

MR. WYLIE: It looked all right. They provided volts per hertz relay, and it looks fine to me.

MR. CARROLL: Is Duke Engineering doing this stuff?

MR. WYLIE: I do not know.

MR. MICHELSON: Was that done on your earlier plants?

MR. WYLIE: Yes.

MR. MICHELSON: Not done for everybody.

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

MR. CARROLL: I just wondered whether Duke

Engineering was doing this electrical stuff.

MR. CROM: Yes, we are.

MR. CARROLL: That makes me suspicious given your heritage.

[Laughter.]

MR. SEALE: No comment.

MR. MICHELSON: For the type of breakers that you now have in mind for this plant, are you going to go to solid state controlled over voltage and under voltage and so forth, or is it going to be the old relay type?

MR. CROM: I am not sure I can answer that right off hand.

MR. MICHELSON: It could be either.

MR. CROM: It could be either right now. It is not specified.

MR. MICHELSON: It did not seem to be specified. The solid state devices have got some unique characteristics.

MR. CARROLL: Duke found out about some of them once not too long ago.

MR. MICHELSON: Yes. And they are programmable too, and you can program them wrong. You get 100 breakers in the plant all programmed wrong, and you get the right event, and they all cascade in the wrong way.

MR. WYLIE: I guess I can ask this other question too about the basis of the test, the ITAAC Test Number 9 in the table. You provided me an answer to that.

MR. CROM: Yes, that's correct.

MR. CARROLL: That is page 70.

MR. COE: It has not been formally submitted yet.

MR. CARROLL: Okay. What's next.

MR. COE: The next one was --

MR. SEALE: You had a question for Bill Shack.

MR. COE: That's correct. That was on page 47 and 48 of your hand-out. That was the question regarding copper content that Dr. Catton referred to Dr. Shack.

MR. CARROLL: Dr. Shack does not have to answer it until next month. Are you prepared to?

MR. SHACK: I am prepared to, yes. In this one, you know, your mother was right: cleanliness is next to godliness; a cleaner steel is a better steel. Cleaning up things like phosphorous and sulfur are going to give you a higher upper shelf energy to begin with. You are just going to have more toughness.

Reducing the cooper, again, there is obviously some limit to where the cooper clusters become so far apart that their embrittlement effect is minimal, but it seems to me a sort of an ALARA process toward the cooper content is a good thing.

Just how much benefit you can go by going very low

I do not think is clear. There is no reason not to go low.

MR. CARROLL: If you can do it.

MR. SHACK: If you can do, do it.

MR. DAVIS: Bill, I am not a metallurgist, but when Ivan came back from Europe he wrote a trip report. One of the inclusions of that was that some of the Europeans are convinced that copper doesn't have anything to do with embrittlement. Did you see that, by the way, his trip report? This came as a real surprise to me, and I do not know what the basis was, but I guess it does not have anything to do with the question.

MR. SHACK: Yes. That sounds like an awfully sweeping statement.

MR DAVIS: Well, I am paraphrasing.

MR. SHACK: I would like to know what context they think that in.

MR. DAVIS: You may want to ask him about that when you see him, because I found it surprising too.

MR. CARROLL: Yes. That was the report on the trip he made.

MR. DAVIS: Way back.

MR. CARROLL: That we finally got just before last meeting.

MR. DAVIS: Right.

MR. CARROLL: Do you remember me commenting why it took him so long to get it?

MR. SEALE: September or something like that. August or September.

MR. SHACK: I have seen arguments like that in terms of low temperature embrittlement. You know, when you are talking about support structures, it is not so clear that copper has an effect on embrittlement at those low temperatures.

To say that copper has no effect on embrittlement at reactor operating temperatures, I would like to see the context that statement was made in.

MR. CARROLL: Okay. So we will tentatively -- or say this one is tentatively scratched off the list subject to Bill --

MR. SHACK: Clarifying with Ivan just what that was about.

MR. CARROLL: All right.

MR. COE: The next one was on page 54, the hand-out. I do not know whether we covered this. This is the post accident radiation monitor basis for the temperature qualification requirement.

MR. CARROLL: Yes, we did.

MR. COE: Okay. That one is finished.

MR. CARROLL: Who have you got down?

MR. COE: Well, talk to Catton.

MR. CARROLL: I think I asked the question, actually. Maybe Catton did, or maybe both of us did. I think we have dealt with it.

MR. COE: Okay. The next one was page 41 of the hand-out. This was the discrepancy in the CESSAR regarding use of cobalt-based materials. That was your question, Mr. Chairman.

MR. CARROLL: Sounds like they resolved it. That they've fixed the problem that I identified. Now the Staff has got to fix the problem, the rad protection section. It looks like they understand that.

MR. COE: Okay. The next one was on page 46 of the hand-out, and that was --

MR. CARROLL: Just on the subject of cobalt, what do you know about that, Bill? How close are we to having some decent replacements for Stellite?

MR. SHACK: When I read the EPRI reports, it looks pretty good.

MR. CARROLL: That is what I was going to say.

MR. SHACK: But if I was buying a plant --

MR. CARROLL: You would still want the option of being able to use Stellite.

MR. MICHELSON: EPRI doesn't make valves.

MR. SHACK: Yes. It is a big investment here. I would like to see a little more field stuff. The research reports look very good for some of these replacements. They have, I guess, some field experience is now being accumulated, but if I was investing my billion dollars, I still would be specifying Stellite at this moment.

MR. SEALE: Could I ask in that regard from the ABB people, this represents a considerable or serious attempt to reduce the amount of cobalt relative to what was in the System 80. Is that correct?

MR. RITTERBUSCH: Yes.

MR. SEALE: I know at Palo Verde they've indicated some problems with cobalt.

MR. CROM: Yes. This is Tom Crom. Yes, most of the current plants have anywhere from .1 to .2 in weight percent cobalt and a lot of the materials where we are talking about .05.

A little bit of a discussion on Stellite replacement, it is not only a factor of the material itself, but the design of the valve. There is currently replacing in plants all globe valves and check valves that have Stellite in them, and they are doing that regularly during maintenance.

The real concern yet is on gate valves where there is a high torque and thrust and the galling. The problem is that the test results that were done were based on the thrust settings that may be calculated, but typically when you do the tests on the thrusts it is a lot higher.

Now, that is a little bit dependent on the design of the valve. You can design the valve appropriately that does not require that high thrust, then you may be able to get away with the replacements.

MR. MICHELSON: We just don't have much experience. The only experience we have is with stellite, and we'd have to have a whole new test program, because that would have a definite effect on the thrust requirements.

MR. CROM: I understand.

MR. CARROLL: GE got rid of the stellitted rollers on their control blades.

MR. MICHELSON: But they didn't get it on their valves.

MR. CARROLL: Well, except the control blades are in the core, which is not --

MR. MICHELSON: That is not nice either.

MR. CARROLL: And I guess combustion has some stellite in the control rod drive mechanism.

MR. SHACK: It is interesting that they have committed to lower the cobalt in the stainless steel more than GE has in the ABWR.

MR. CARROLL: Is that correct?

MR. SHACK: Yes. GE is sticking with the 0.1 percent rather than the 0.05 percent. Obviously, somebody has different perceptions of what is commercially achievable.

MR. CARROLL: Do you want to change your mind?

MR. RITTERBUSCH: We checked before we wrote down 0.05.

MR. CARROLL: Good.

MR. COE: The next one was on page 46, is question number 7 regarding the capability for System 80+ to handle frequency degradation without tripping the reactor. This is Mr. Carroll's question.

MR. CARROLL: It was my question and I got the answer and then I heard Lindbald say he had a question about this, so we haven't put that one to rest. I don't know what his question is.

MR. COE: Then the next one is on page 53, question number 12. This had to do with the extent to which tech specs allow alternate AC to be used as a backup for the diesel generators.

MR. MICHELSON: Is that question 11 or 12?

MR. COE: It was question -- excuse me, 11. Page 53.

MR. CARROLL: We know a bit more from what we heard earlier this afternoon. I think you can scratch that guy off.

MR. DAVIS: This provision has been accepted by the Staff; is that correct?

MR. CARROLL: What is that?

MR. DAVIS: That the AC can be used to replace one diesel generator?

MR. CARROLL: The tech specs will require the two-of-three concept during shutdown.

MR. DAVIS: Oh, okay. I'm sorry.

MR. CARROLL: But the issue of what credit you get for the AAC during mode 1 is still up in the air. That was what we commented on in our letter last month and what we hear Combustion and the Staff are going to be discussing in the near future.

MR. SIEGMANN: I would like to correct something there. This is Eric Siegmann.

Our tech specs require two of the three during shutdown. I am not too sure what the NRC requires. But they might require two of the three during mid loop or reduced inventory.

MR. CARROLL: The tech specs you have voluntarily submitted require two of three at all times in mode 6?

MR. SIEGMANN: No, in shutdown. That's modes 4, 5 and 6, basically.

MR. CARROLL: Four, 5 and 6. Okay.

MR. ARCHITZEL: Ralph Architzel from Staff.

I believe we are going to require, in mid-loop, two power sources on site. And we are allowing the use of the combustion turbine. They could have just proposed their two safety diesels as opposed to the combustion turbine. So we allowed them to use the combustion turbine.

I don't believe that we require two in all shutdown modes because -- well, in mode 6 when you're fueled up, you'll require more than one.

MR. CARROLL: But if they're willing to commit to that, you're not going to disagree with it?

MR. ARCHITZEL: To two at all times? We didn't disagree, no. I'm not aware on that particular point.

MR. CARROLL: I saw it someplace in here. Okay.

MR. COE: The next question was one for the Staff. Question number 14, this was responded to on the Staff's handout about six pages on the back.

MR. MICHELSON: Are we done with CE's handout yet?

MR. COE: No, you have a couple questions remaining.

MR. MICHELSON: I have a couple others I want clarification on.

MR. COE: Right.

It's about six pages from the back of the Staff's handout. The pages are not numbered. But about six from the back, it's question number 14. This is Mr. Carroll's question regarding the use of the term "vital areas" commonly for both security and RP purposes.

And the Staff has responded with a copy of a letter the ACRS wrote in February.

MR. CARROLL: All right, a copy of the response we got.

MR. COE: A copy of the response, yes. The response to that letter.

MR. CARROLL: I'm not giving up. But scratch it off the list.

MR. COE: Okay.

And the next question is responded to immediately after that by the staff, question number 15, also Mr. Carroll's question regarding the -- an apparent discrepancy in the FSER on the requirements for the OSC being in the ITAAC.

MR. CARROLL: I thought they answered that during the meeting, that that was an oversight and they were going to fix it.

MR. COE: And they have indicated that's a confirmatory item.

And the last question then on Staff responses is the following question after that. Number 16, regarding the exception that was taken on the topical report that was referenced in this -- in the draft FSER, also Mr. Carroll's question .

MR. CARROLL: What was the answer?

MR. COE: See attached page.

MR. CARROLL: We didn't look at that. Why don't you move on while we're looking.

MR. COE: The next one is on the handout that I have distributed on page 49 through 51, Mr. Michelson's question -- oh, excuse me. We have already discussed that one.

MR. MICHELSON: Well, yes. But as long as you've got it open now, though, it wasn't -- it is not entirely clear to me how we finally ended up as to how it is going to be resolved.

MR. COE: I have two notes. One is that we will treat a portion of it at the subcommittee meeting in May, the Auxiliary and Secondary Systems --

MR. MICHELSON: It's now going to be June.

MR. COE: Which is now going to be June.

MR. MICHELSON: June 8th, I guess it is, something like that.

MR. COE: Also a portion of it will be treated at the April subcommittee meeting of this subcommittee.

MR. MICHELSON: I didn't understand that. Is that going to be because we pick up chapter whatever it is where it appears?

MR. COE: Yes.

MR. MICHELSON: Much of the argument in here is going to have to be getting all these cats together that have different stories and get the story to come together.

MR. COE: That's correct. I was anticipating that in Chapter 9 of the CESSAR that we would be discussing, the fire protection issues.

MR. MICHELSON: Okay. So that will be in April, then?

MR. COE: In April.

MR. MICHELSON: But we will won't hear from the University of Maryland and so forth until later, June. I don't know why we couldn't get that meeting earlier except that nobody could come.

MR. DUDLEY: This is Noel Dudley. The report from Maryland will be issued in the next couple of weeks.

MR. MICHELSON: Okay. But, of course, what we wanted to do is discuss it with the department head and so forth to understand it and then act accordingly on System 80. That doesn't fit too well, a subcommittee that doesn't meet until June.

MR. COE: Not very well, no.

MR. MICHELSON: When are you going to write a final letter?

MR. COE: It will be out the door in June.

MR. MICHELSON: Well, that doesn't fit at all, then. We had it in May, but I understood there was somebody who couldn't come. Why didn't we move it further up? If the report is going to be in, in two weeks, why isn't it going to be in April, the meeting?

MR. DUDLEY: I could work on that.

MR. MICHELSON: Why don't you get with Ivan again and find out because really that needs to come before we talk about the diesels, the fire protection, with CE.

MR. COE: I think we will have to look into that.

MR. MICHELSON: I didn't think well enough. But clearly June is just too late to fit much of anything.

MR. CARROLL: Backing up one, I am happy with the staff response.

MR. COE: On question 16.

MR. MICHELSON: That took care of 9, then.

MR. COE: All right. Well, then, there was one final question. Mr. Michelson had asked Number 10 on page 52 regarding the design capability of the doors between the turbine building and the nuclear annex.

MR. MICHELSON: There the claim is that they will be designed for the tornado case, which is a differential pressure of 2.4. Those are going to be pretty good doors. People have started looking at doors very closely and have found that that is a tough requirement.

MR. CARROLL: How do you open it?

MR. MICHELSON: How do you do other things? If it is a big double swinging door, it gets tough to latch. But at any rate, the only question I had was: Has somebody done the calculations of the postulated brakes that might occur in the neighborhood of those doors to see if the 2.4 bounds those calculations or not?

MR. STAMM: The answer is we have done enough to know that we can make a commitment to make it work at 2.4.

MR. MICHELSON: A lot of big vent areas available and so forth?

MR. STAMM: Yes, in the turbine building typically you have a lot of spaces for venting.

MR. MICHELSON: No jet impingements on the door or anything like that?

MR. STAMM: No, if you wanted to, I could show you a sketch.

MR. MICHELSON: Not really. I will just take your word for it. You looked at all the hazards of a pipe break in the vicinity. The doors are protected against that?

MR. STAMM: Yes, the doors are actually --

MR. MICHELSON: They are pressurized and it is less than the tornado design requirements?

MR. STAMM: That's right. The doors actually are on the annex side of an enclosed stairwell. The piping is run on the floor below, about 20 or 30 feet below the elevation of the doors.

MR. MICHELSON: So it already shielded well from the direct impingement effects?

MR. STAMM: Correct.

MR. MICHELSON: That should take care of the answer. 2.4 is the key. That is a good door.

MR. CARROLL: Those are our follow-up questions.

MR. COE: I have just one reminder for CE. We are still waiting for additional information on the two follow-up questions we discussed the last meeting -- questions 3 and 8 from the December meeting. There were some commitments made to add some language to the answers.

Also from this meeting, we had earlier discussed their commitment to try to get additional information.

MR. RITTENBUSCH: We agree.

MR. COE: Okay.

MR. MICHELSON: One clarification on this door business since we have now identified the turbine door. Are all doors to the nuclear island qualified then, all doors to the outside, for tornado depressurization effect? They would all be qualified for the 2.4; is that correct?

MR. OSWALD: Todd Oswald. Yes, all exterior doors are.

MR. MICHELSON: Okay. Thank you.

MR. CARROLL: Okay. Those are the questions.

All right. I guess we have one other item on the agenda that Stan tells me he can deal with in exactly 60 seconds. The challenge is on ITAAC.

[Laughter.]

MR. CARROLL: Now I have already told him that we have been Tony James'd to death on ABWR, so we really didn't need any more of that.

[Slide.]

MR. RITTENBUSCH: You may know that I have been with Combustion Engineering for 24 years now. I hope that qualifies me to summarize in 60 seconds what it took us three or four years to produce.

I am not going to read to you the details of our approach to CDM and what it is. What I would like to say is that we have participated with the NRC staff, meetings with General Electric and the industry ever since the beginning of the ITAAC and the CDM saga several years ago. We have adopted all of their lessons learned.

This means that the process that we used for developing our CDM is nearly identical to that used by General Electric. Our designs, however, are different. Therefore obviously, we have a different ITAAC, and in a few cases, a few different approaches, which I will state in a second.

[Slide.]

MR. RITTENBUSCH: I am now going to go to the next slide. You are probably all familiar with the basic segments of certified design material. I am simply going to point to Section 3, which is the non-system specific design descriptions. This is one area where we had a difference from ABWR. ABWR proposed four non-system specific ITAACs. We have two.

MR. CARROLL: Sometimes referred to as DACs.

MR. RITTENBUSCH: That is correct. You said it first.

MR. CARROLL: Okay.

MR. RITTENBUSCH: That finishes my presentation.

MR. MICHELSON: Tell us what the two are.

MR. RITTENBUSCH: Piping and shielding.

MR. MICHELSON: You don't need any DAC, then, for your instrumentation and control arrangements?

MR. RITTENBUSCH: Well, that is covered in our sections on ITAAC.

MR. MICHELSON: That means there are no additional things to be confirmed in future dates?

MR. CARROLL: Oh, yes, both for that and for --

MR. MICHELSON: Well, that is what a DAC is about.

MR. CARROLL: Yes, except they put them into Section 2 kind of stuff.

MR. MICHELSON: Well, what you are saying is that two of the four DACs you put in a different section in a different way and you don't call them DACs anymore?

MR. RITTENBUSCH: I will be a little more explicit. On the other two DACs we have done sufficient additional design work such that we thought it was more appropriate to put them into the context of ITAAC rather than DAC.

MR. MICHELSON: Sure.

MR. RITTENBUSCH: I believe the staff concurred.

MR. MICHELSON: Okay. I understand now. I was just wondered how you were doing it.

MR. ARCHITZEL: This is Ralph Architzel from the staff. I don't think staff quite characterized it that way. We are probably pretty much in disagreement that they don't have four DACs right now. We are going to be working that out. That is one of the confirmatory items.

MR. MICHELSON: Is it a question of which section they go into?

MR. ARCHITZEL: It is not the difference in what they are going to do, but a difference in how we characterize them. We believe there are four DACs. We are going to get together with them.

MR. MICHELSON: I thought it a little skimpy for a final design.

MR. FRANOVICH: Yes. It's true. When you take the example of the control room design, that is certainly not a final design. That's still a DAC and so fundamentally we don't agree that it is analogous to a system ITAAC.

MR. RITTERBUSCH: I want to apologize to Staff. I didn't mean to say we agreed on DAC. I knew we did not agree on the terminology on what we call these. What I wanted to say is that the Staff has looked at the manner of incorporating our control room and I&C into the CDM and I believe we do have agreement on that.

MR. FRANOVICH: Yes, we do have agreement on the process, on the design process, no doubt.

MR. MICHELSON: Let me ask the subcommittee chairman then how are we going to treat -- are we going to look at the same four areas as we did or are we not going to look at I&C at all?

MR. CARROLL: We did. We looked at I&C in December in the context of Chapter 7.

MR. MICHELSON: Yes.

MR. CARROLL: And the certified design mechanism.

MR. MICHELSON: I didn't realize we were covering all the certified design material at the same time.

MR. CARROLL: Yes.

MR. MICHELSON: Okay.

MR. CARROLL: Similarly, we looked at the human factors area in terms of --

MR. MICHELSON: That was our review of the CDM as well, okay.

MR. CARROLL: Because that is really the best place to look for an overview of it.

MR. MICHELSON: For an overview, that's true.

MR. CARROLL: Okay. I would point out that I went through the Staff FSAR on 14.3 and it goes on for quite a number of pages and it does describe the Staff's process by which they set up multidisciplinary task forces, if you will,1 to look at whether this certified design material was really what ought to be in there and I was fairly impressed with what was described there.

Did I say that well?

MR. ARCHITZEL: One point I would like to make, to make sure that the ACRS is familiar -- we are currently doing an independent or quality verification, if you will, of the ITAAC, the CDM. I am not talking about the trip that went up there with the multidiscipline team. We also had a multidiscipline team doing a quality verification of the ITAAC and we have got back right now a stack of comments on the ITAAC that we are going to work out first internally and then the remainder of those comments we'll be interacting with CE to resolve problems, so it's not totally finished yet but that process is ongoing right now.

MR. FRANOVICH: I would like to add to that, that is a confirmatory item in the FSAR and it is a QA check between Tier 1 and the SSAR material.

MR. CARROLL: I did note on page 14-3 something Earl always is asking about. There is no design information presented in the CDM or Section 14.3 that is not also contained in the various sections of the CESSAR-DC.

Staff did not base its safety evaluation for the design on the information in the CDM and therefore this section of the report contains no safety evaluation of the design.

MR. MICHELSON: Therefore we don't need to look at it.

MR. CARROLL: But the point is they are saying in very clear terms that what they are basing their safety determination on is the SSER and not the CDM.

MR. MICHELSON: Yes, we don't need to review the CDM at all to determine safety of the plant.

MR. CARROLL: Okay. Your one minute is up, Ritterbusch.

MR. RITTERBUSCH: Okay, thank you.

MR. CARROLL: Is there any more that should come before this august body? We're adjourned -- I just want to mention to Bill that you understand what you are on the hook for next month?

MR. SHACK: Yes.

MR. RITTERBUSCH: May I ask a question?

MR. CARROLL: Yes.

MR. RITTERBUSCH: Can you give us an assessment of the schedule? I heard you mention something about meetings in June. It is my understanding that --

MR. CARROLL: I said the report will be out the door in June.

MR. RITTERBUSCH: Okay. Well, I will state my understanding and if you agree, I would appreciate it. It's my understanding that we will have two days of meetings on April 5th and 6th.

MR. CARROLL: Correct.

MR. RITTERBUSCH: And at that time we will assess the need for an additional half-day, approximately, clean-up meeting at some point in late April or early May.

MR. CARROLL: That is probably --

MR. DAVIS: Early May probably.

MR. CARROLL: Yes, probably early May, yes.

MR. RITTERBUSCH: Thank you.

MR. CARROLL: We do have the question of whether we need to have a presentation to the full committee in April. We are still discussing that and hopefully we'll get some resolution to that during our full committee meeting over the next two days.

MR. RITTERBUSCH: We will support that.

MR. CARROLL: I don't know that you need to.

MR. RITTERBUSCH: If asked.

MR. CARROLL: Okay. I am arguing that the subcommittee is virtually the full committee and --

MR. MICHELSON: It is not the full committee though. The members of the full committee have to make the same decision we have to make and they have to hear enough of a presentation so they feel comfortable.

MR. CARROLL: The members of the full committee that have not participated in these meetings are Ernest, who won't be here in June --

MR. MICHELSON: That's a plus, I guess, in a way.

MR. CARROLL: And Hal, who has participated in what I think the areas of his interest are.

MR. MICHELSON: If you feels that he has participated, then that's fine --

MR. CARROLL: Well, that's what I want to debate, and Bill, who has beer participating, and Ivan is a member, so --

MR. MICHELSON: So you probably are okay.

MR. CARROLL: Well, I don't know. Doug raises the question of whether we have to --

MR. MICHELSON: But the public needs to -- you know, they can come to these meetings, too, but somehow they generally come just to the full committee's final discussion.

MR. SEALE: Check with the lawyers.

MR. CARROLL: So we are going to have to check that one out.

MR. MICHELSON: You could make it an hour.

MR. CARROLL: We could have an hour of overview.

MR. MICHELSON: No one says how long it has to be, only that I think it's probably good politics to do it.

MR. CARROLL: I think it is, too.

MR. MICHELSON: If somebody says they have got to hear six hours, then I guess you have got to arrange for six hours.

MR. CARROLL: Then I'm going to argue with them.

Okay, we stand adjourned.

[Whereupon, at 3:04 p.m., the meeting was adjourned.]

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 ABB CE Plant Design

DOCKET NUMBER:

PLACE OF PROCEEDING: Bethesda, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Marilynn Estep

Official Reporter
Ann Riley & Associates, Ltd.

ABB Combustion Engineering

System 80+™ Standard Plant
CESSAR - DC Section 3.9.6
Testing of Pumps and Valves

Thomas D. Crom
Duke Engineering & Services, Inc.

ACRS ABB-CE Standard Plant Designs Subcommittee
March 8 & 9, 1994

System 80+™ Standard Plant

Testing of Safety-Related Pumps and Valves

- Section 3.9.6 General
- Table 3.9-15 Inservice testing plan for pumps and valves
- Pump and valve testing provisions
- COL Applicant responsibilities

System 80+^{TM} Standard Plant

Section 3.9.6 General

- Definition of "Safety-related" for IST applicability
  - Safety-related pumps and valves include those necessary to ensure:
    - The integrity of the reactor coolant pressure boundary.
    - The capability to achieve safe shutdown of the reactor and keep it in a safe shutdown condition.
    - The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures in excess of 10 CFR 100 Guidelines.

System 80+™ Standard Plant

Table 3.9-15 Inservice Testing Safety-Related Pumps and Valves

- IST Plan (Table 3.9-15) developed to identify safety-related components tested in accordance with:
  - ASME/ANSI OMa-1988 of Addenda to ASME/ANSI OM-1987;
    - Part 1 - Relief Valves
    - Part 6 - Pumps
    - Part 10 - Valves (other than relief valves)
  - 10 CFR 5O Appendix J Testing

Table 3.9-15 Inservice Testing Safety-Releated Pumps and Valves (continued)

- Plan format for pumps
  - Pump name
  - ASME Safety Class
  - Test parameter
    - DP - Differential pressure
    - SPs - Static suction pressure
    - SPo - Operating suction pressure
    - Q - Flow
    - V - Speed
  - Test frequency
  - Test configuration
  - CESSAR-DC figure number

- Plan format for valves
  - Valve number
  - Valve name
  - Valve type
  - Valve actuator
  - ASME Safety Class
  - ASME Code Category
  - Valve function (cont. isol., TIV, PIV)
  - Required testing (stroke, leak test, etc.)
  - Test frequency
  - Test configuration
  - CESSAR-DC figure number

System 80+™ Standard Plant

Testing of Safety-Related Pumps

- System 80+ Design includes provisions for:
  - Full flow pump testing during plant operations (quarterly)
  - Capability to measure NPSH throughout pump operating range
  - Redundancy and separation of systems and components to allow complete pump testing with minimal impact to plant operability.

System 80+^{TM} Standard Plant

Testing of Safety-Related Valves

- System 80+ Design includes provisions for:
  - Capability to perform required testing
  - Redundancy and separation of systems and components to allow maximum valve testing with minimal impact to plant operation

System 80+^{TM} Standard Plant

COL Applicant Responsibilities - General

- COL Applicant provids details of IST:
  - Test procedures
  - Test schedules
  - Test frequencies
  - Baseline preservice test program

System 80+^{TM} Standard Plant

COL Applicant Responsibilites - Pumps

- COL applicant establishes baseline pump design qualification testing encompassing design conditions which demonstrate acceptable pump performance
- COL applicant ensures pump specified is not susceptible to inadequate mininum flow and inadequate thrust bearing capacity
- COL applicant develops pump disassembly program for all safety-related pumps; program based on:
  - Historical pump performance
  - Pump components' performance
  - Non-intrusive test results

COL Applicant Responsibilities - MOVs and POVs

- COL applicant establishes baseline valve design qualification testing encompassing design conditions which demonstrate acceptable valve performance (i.e., torque, thrust, force); conditions vary based on valve type (MOV, POV) but generally are:
  - Fluid flow
  - Differental pressure (including pipe break)
  - System pressure
  - Fluid temperature
  - Ambient temperature
  - Minimum voltage/pneumatic or hydraulic pressure
  - Minimum and maximum stroke time requirements

COL Applicant Responsibilities - MOVs and POVs (continued)

- COL Applicant ensures that MOV specified for each application is not susceptible to pressure locking and thermal binding
- COL Applicant will periodically test MOVs (per Generic Letter 89-10 paragraphs D and J) and POVs to demonstrate continuing capability for design basic conditions

System 80+™ Standard Plant

COL Aplicant Responsiblities - Check Valves

- COL Applicant establishes baseline valve design qualification testing encompassing design conditions which demonstrate acceptable valve performance (i.e., stroke and sealing) for check valves; check valves are tested for design conditions of:
  - Required operating cycles to be experienced by the valve - numbers of operating cycles and duration
  - Severe transient loadings (pipe break/waterhammer)
  - Sealing and leakage requirements
  - Operating medium temperature and gradients
  - Vibratory loading

COL Aplicant Responsiblities - Check Valves (con't)

- COL Applicant ensures that check valve application is proper (size, type, Location, orientation) as recommended by manufacturer.
- COL Applicant ensures capability of nonintrusive testing, measurable flow through check valve

System 80+™ Standard Plant

COL Responsibilites - Valve Disassembly Programs

- COL Applicant develops valve disassembly program for all safety-related valves; program based on:
  - Historical valve performance
  - Valve constituent components' performance
  - Non-intrusive test results

ABB Combustion Engineering

System 80+™ Standard Plant
Flood Protection

Thomas D. Crom
Duke Engineering & Services, Inc.

ACRS ABB-CE Standard Plant Designs Subcommittee
March 8 & 9, 1994

System 80+™ Standard Plant

External Flood Protection

Site Parameters:

- Grade elevation - 90+9 (reference)
- Maximum groundwater level - two feet below grade (elevation 88+9)
- Probable maximum flood (PFM) level for site - one foot below grade (elevation 89+9) (PMF defined in ANSI/ANS-2.8)

System 80+^{TM} Standard Plant

External Flood Protection

Design Features:

- Concrete construction joints sealed with waterstops
- External penetrations below grade sealed
- Doors/accesses at least one foot above grade (elevaticn 91+9)
- Seepage will end up in sumps in basement through floor drains

System 80+™ Standard Plant

Internal Flood Protection

Features:

- Station Service Water is located outside the Nuclear Annex
- Component Cooling Water and Emergency Feedwater Systems are fully separated by division
- Divisional wall is primary means of flood control in the Nuclear Annex
- No doors are provided up to EL. 70+0 in the divisional wall and diesel generator rooms
- Reactor Building Subsphere separated into quadrants

System 80+^{TM} Standard Plant

Internal Flood Protection

Features (continued):

- Flood barriers provide separation between electrical equipment and mechanical systems at the lowest elevation within the Nuclear Annex
- Emergency Feedwater pump is located in separate compartment within each quadrant with each compartment protected by flood barriers
- Flood doors are provided with open and close sensors and are alarmed in the control room
- At higher elevations electrical equipment is elevated above the floor so that flooding events will not affect components

System 80+^{TM} Standard Plant

Internal Flood Protection

Features (continued):

- Floor drainage systems are separated by division and by quadrant in subsphere
- Safety Class 3 check valves are provided to prevent backflow of water to areas containing safety related equipment within a division
- Each subsphere quadrant and each diesel generator room is provided with redundant Safety Class 3 sump pumps and associated instrumentation, which are powered from the diesel generators
- No water lines are routed above or through the control room or computer room

System 80+™ Standard Plant

Internal Flood Protection

Features (continued):

- Water lines to HVAC air conditioning units around the control room are contained in rooms with curbs
- Component Cooling Water Heat Exchanger Structure is divisionally separated such that a flood in one division cannot flood the other division
- Door leading from the Turbine Building to the Nuclear Annex is located above the maximum Turbine Building flood elevation

System 80+™ Standard Plant

Internal Flood Protection

Analysis:

- An analysis was performed which demonstrated that the following system volumes can only flood one division of the Nuclear Annex assuming no operator action to terminate the flood

  - One Component Cooling Water System division including external piping and surge tank
  - Incontainment Refueling Water Storage Tank
  - One Emergency Feedwater System division including Emergency Feedwater Storage Tank
  - Fire Protection System including two external water supply tanks
  - Chemical Volume Control System including external Holdup Tank, Boric Acid Tank and Reactor Makeup Tank

System 80+™ Standard Plant

Internal Flood Protection

Analysis (continued):

- The COL applicant shall perform the flooding anaysis associated with high and moderate energy line pipe rupture analysis outside of containment
- The divisional and interdivisional flood barriers ensure that a high or moderate energy line break outside of containment can be mitigated assuming loss of offsite power and single failure (normal operating systems such as the Component Cooling Water System are excluded from the single failure criteria)

System 80+™ Standard Plant

High Energy Lines

High Energy Systems within Containment:

- Main Steam System
- Main Feedwater System
- Steam Generator Blowdown System
- Steam Generator Wetlayup and Recirculation System
- Reactor Coolant System
- Safety Depressurization System
- Chemical Volume and Control System
- Safety Injection System
- Emergency Feedwater System

System 80+^{TM} Standard Plant

High Energy Lines

High Energy Systems outside Containment:

- Main Steam System
- Main Feedwater System
- Steam Generator Blowdown System
- Emergency Feedwater System (steam line to turbine driven pump)
- Chemical Volume Control and System (two inch line)

System 80+™ Standard Plant

High Energy Lines

Location of high energy lines outside Containment:

- Main Steam, Main Feedwater and Steam Generator Blowdown Systems penetrate containment annulus area through guard pipes and are located in main steam valve houses, yard, and turbine building
- Emergency Feedwater System steam line to turbine driven pump located in main steam valve houses, turbine driven pump rooms, and vented pipe chase between turbine driven pump rooms and main steam valve house
- Chemical Volume and Control System located in pipe chase from containment penetration to Chemical Volume Control System area

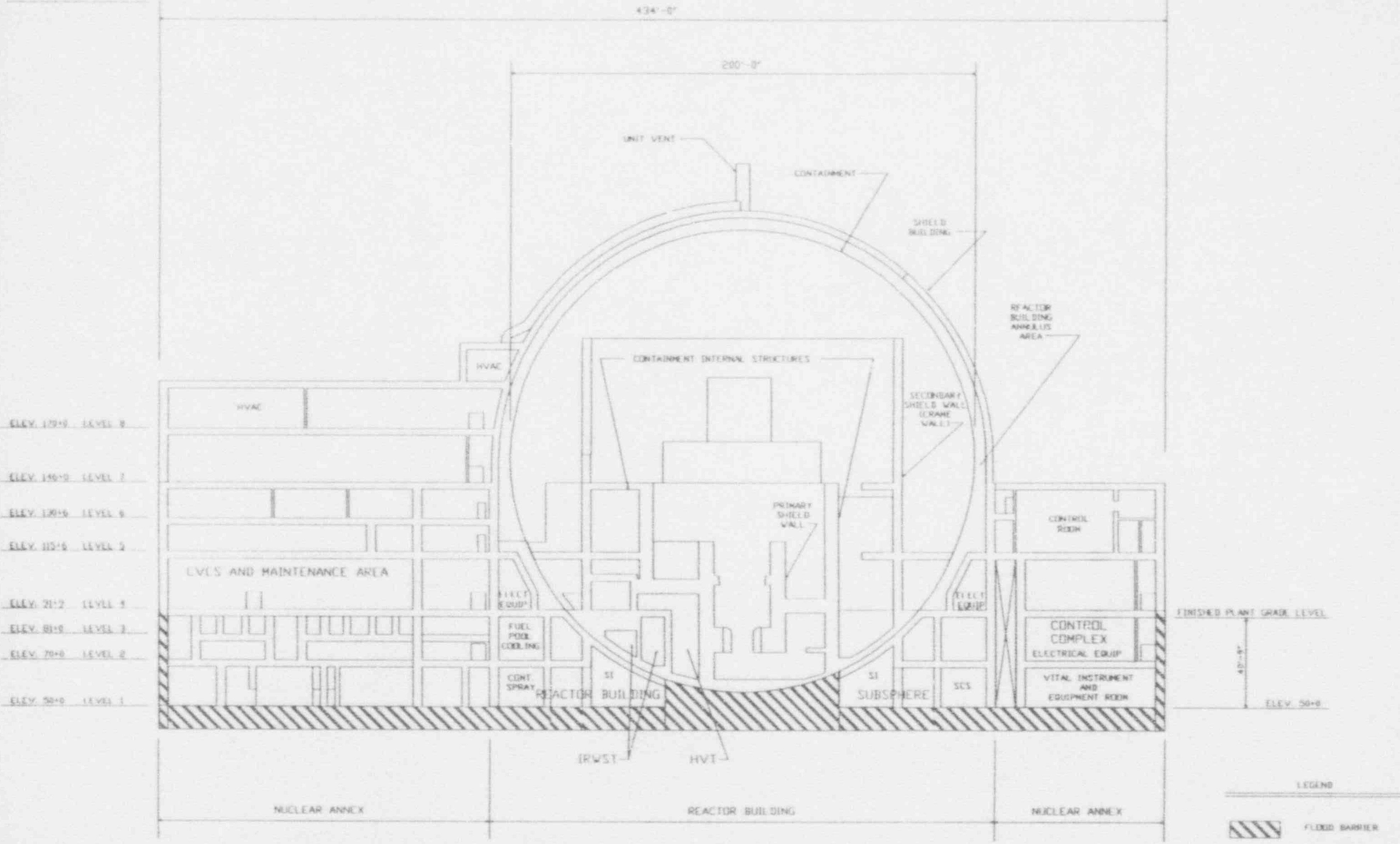
System 80+™ Standard Plant

Flood Protection

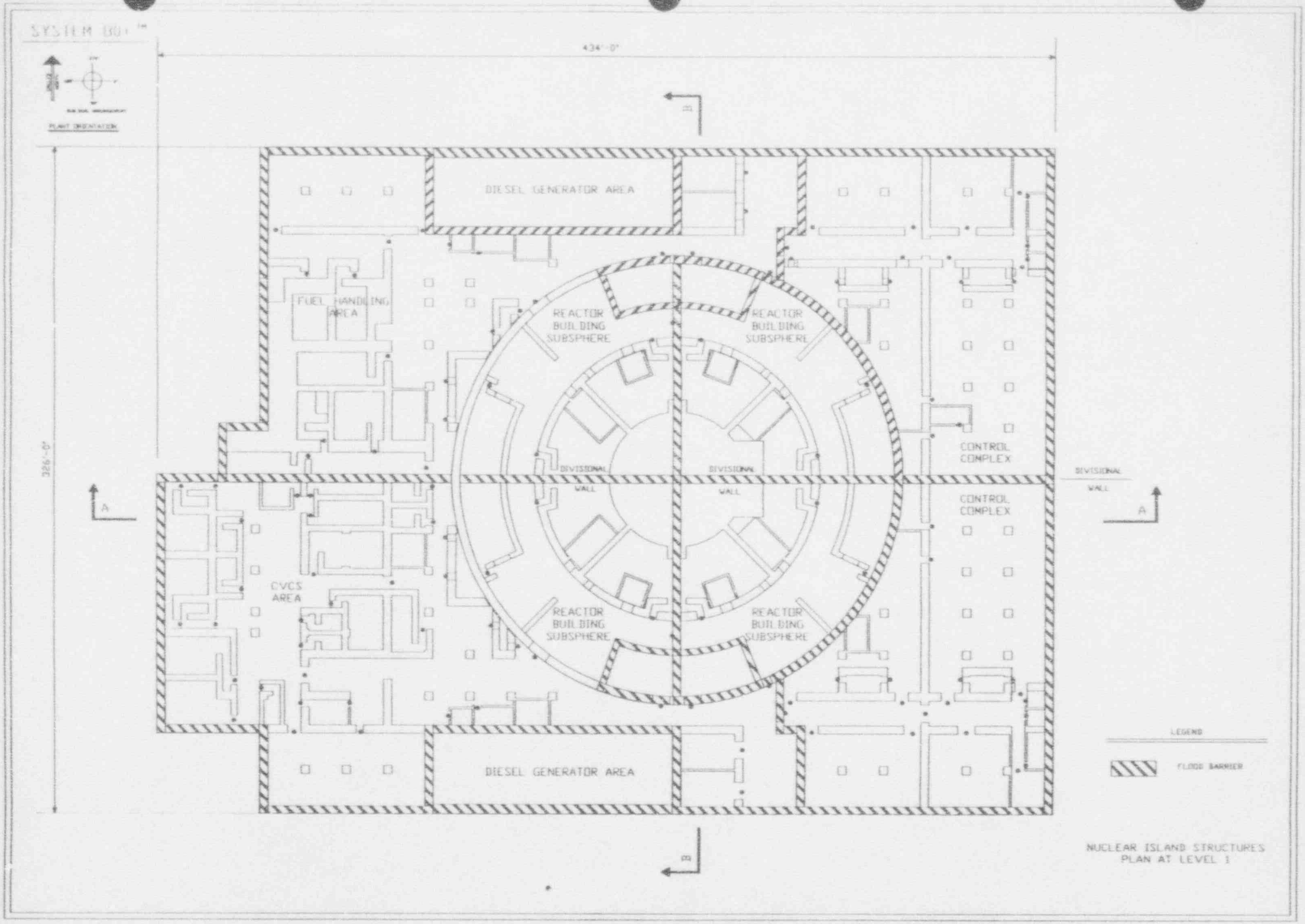
ITAAC Scope:

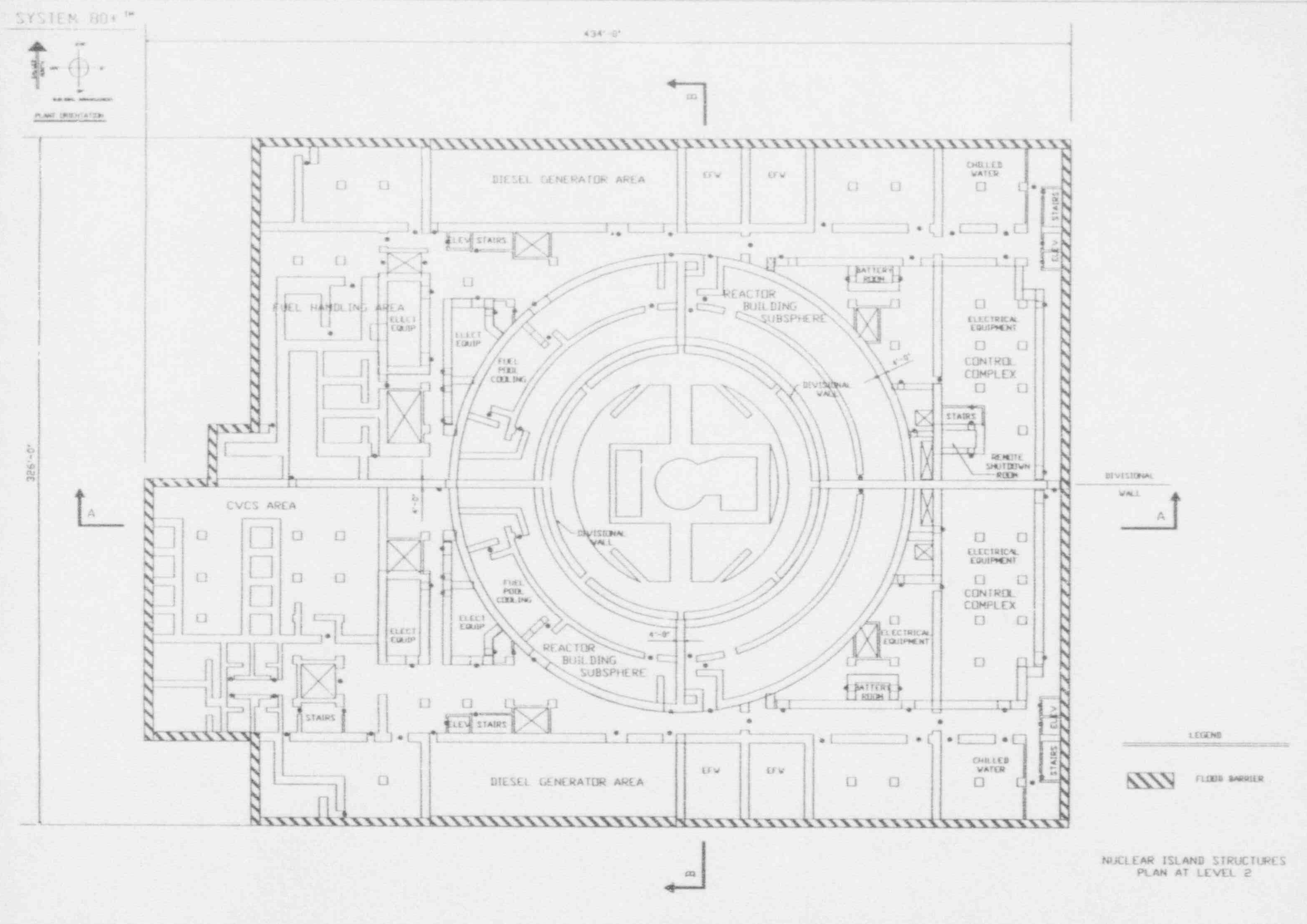
- Flood barriers
- Structural load from flooding
- Sensors on flood doors
- Divisional and quadrant separation of floor drains
- Station Service Water located outside the Nuclear Annex
- Divisional separation of systems
- Safety Class 3 check valves to prevent backflow
- Reactor Building Subsphere and Diesel Generator rooms provided with redundant Safety Class 3 sump pumps powered from the diesel generators

SYSTEM 80+™



NUCLEAR ISLAND STRUCTURES
SECTION A-A


SYSTEM 80+™
PLANT ORIENTATION
434'-0"
326'-0"
DIESEL GENERATOR AREA
FUEL HANDLING AREA
REACTOR BUILDING SUBSPHERE
REACTOR BUILDING SUBSPHERE
REACTOR BUILDING SUBSPHERE
REACTOR BUILDING SUBSPHERE
DIVISIONAL WALL
DIVISIONAL WALL
DIVISIONAL WALL
CONTROL COMPLEX
CONTROL COMPLEX
CVCS AREA
DIESEL GENERATOR AREA
A
A
B
B
LEGEND
FLOOD BARRIER
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 2

ABB Combustion Engineering System 80+TM Standard Plant

Chapter 19 - Probabilistic Risk Assessment

David J Finnicum

ACRS ABB-CE Standard Plant Designs Subcommittee

March 8 & 9, 1994

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System 80+ Standard Plant
Probabilistic Risk Assessment

- Objectives
- Approach
- Methodology
- Results
- ACRS Issues

ABB

System 80+ Standard Plant Probabilistic Risk Assessment

- Objectives:
  - Comply with Severe Accident Policy Statement providing a Level III PRA for the System 80+ Design
  - Demonstrate compliance with EPRI ALWR Mean Core Damage Frequency Goal of 1.0E-5 events/year
  - Demonstrate compliance with large release goal of 1.0E-6 events/year
  - Demonstrate containment performance/reliability
  - Support evaluation of design changes and demonstration that System 80+ provides an increased level of safety.

ABB

System 80+ Standard Plant Probabilistic Risk Assessment

- *Approach:*
  - Establish baseline PRA for System 80+ (i.e., System 80)
  - Use PRA as evaluation tool for assessment of design changes
  - Prepare Level III PRA for System 80+
  - Include Evaluation of External Events

ABB

System 80+ Standard Plant
PRA Methodology

- Level I
  - Small Event Tree/Large Fault Tree Approach
  - Front Line System Fault Trees Include:
    - System Component Failures
    - Common Cause Faults
    - Maintenance Unavailability
    - Operator Actions
    - Full Support System Models

ABB

System 80+ Standard Plant PRA Methodology (Cont.)

- External Events
  - Qualitative Screening of External Events
  - Quantitative Analysis of Tornado Strikes
  - Quantitative Scoping Analyses
    - Internal Fire
    - Internal Flood
  - Seismic Margins Assessment for Earthquake

ABB

System 80+ Standard Plant PRA Methodology (Cont.)

- PRA Based Seismic Margins Assessment
  - Modifiy Level 1 Fault Tree Models to Include Seismic Failure of Structures and Components
  - Construct Seismic Event Trees
  - Solve Seismic Core Damage Sequences Using Fault Tree Linking to Obtain Cutsets for Each Seismic Core Damage Sequence
  - Calculate High Confidence of Low Probability of Failure (HCLPF) Values For Components and Structures
    - HCLPF values calculated using EPRI CDFM Approach
    - HCLPF calculations used a Review Level Earthquake of 0.6g with a modified NUREG/CR-0098 Spectral Shape
    - System 80+ specific response spectra curves for 0.3g design basis earthquake reviewed against RLE spectra

ABB

System 80+ Standard Plant PRA Methodology (Cont.)

- Shutdown Risk
  - Outage Divided Into 4 Plant Operating States (POS)
    - Mode 4, 5 (Normal Inventory), Mode 6F (IRWST Full, Refueling Cavity Empty
    - Mode 5R (Reduced Inventory, Including Mid-Loop)
    - Mode 6E (IRWST Empty, Refueling Cavity Full, Upper Internals Removed)
    - Mode 6I (Refueling Cavity Full, Upper Internals in Place)
  - For Each POS, Event Trees Developed for 4 Event Types
    - Loss of DHR
    - Small LOCA
    - Fire
    - Loss of Offsite Power

ABB

System 80+ Standard Plant PRA Methodology (Cont.)

- Shutdown Risk (Cont.)
  - Initiating Event Frequencies Taken From BNL 1991 Study
  - Fault Trees Were Developed For Each Branch Point
    - Trees Modified From Level 1 Fault trees
  - Human Error Probabilities Developed for Two Response Times
    - 40 Minute Response for Reduced Inventory
    - Two Hour Response For All Other Events

ABB

System 80+ Standard Plant PRA Methodology (Cont.)

- Level II
  - Define and Quantify Plant Damage States
  - Develop Containment Event Tree and Supporting Logic Models
  - Quantify CET
  - Define Release Classes
  - Perform Sensitivity Analyses for Selected Parameters

ABB

System 80+ Standard Plant PRA Methodology (Cont.)

- Level III
  - Risk Measure Selected - Dose at 0.5 Miles
  - Use MACCS to Determine Dose at Distance
    - Meteorological Data for Bounding Site Provided by EPRI
    - Demographic/Population Data not used
    - Assumed No Evacuation
    - Calculated Complementary Cumulative Distribution Function (CCDF) for Whole Body Dose at 0.5 Miles and at 300 Meters from Reactor
  - Sensitivity Analyses for Selected Issues

ABB

System 80+ Standard Plant PRA Results

ABB

System 80+ Standard Plant
Core Damage Frequency Contributions

Initiating Event	System 80 CDF (Original Groundrules)	System 80+ CDF (Original Groundrules)	Major Design Contributor
Large LOCA	1.8E-06	5.0E-08	IRWST, 4T ECCS
Medium LOCA	3.6E-06	9.1E-08	IRWST, 4T ECCS
Small LOCA	9.4E-06	4.4E-08	4T ECCS, EFWS
Secondary Side Break	9.0E-07	2.0E-10	4T ECCS, EFWS
SGTR	1.1E-05	8.0E-08	4T ECCS, EFWS
Transients	1.2E-05	3.3E-08	4T EFWS, F&B
Loss of Offsite Power (Including SBO)	3.8E-05	1.0E-07	2 DG + AAC, EFWS, 6 BAT.
ATWS	4.8E-06	1.7E-07	4T EFWS
Interfacing System LOCA	4.5E-09	5.2E-10	High Pres. Pipe
Vessel Rupture	1.0E-07	1.0E-07	
Total	8.1E-05	6.7E-07	

ABB

System 80+ Standard Plant
Core Damage Frequency Contributions

Initiating Event	System 80+ CDF (Original Groundrules)	System 30+ CDF (Current Groundrules)	Changed Methods & Assumptions
Large LOCA	5.0E-08	1.1E-07	Include Check Valve CCF, Change HRA Calc. Methods, MOV Failure Rates
Medium LOCA	9.1E-08	3.1E-07	
Small LOCA	4.4E-08	2.1E-07	
Secondary Side Break	2.0E-10	2.1E-09	
SGTR	8.0E-08	3.0E-07	
Transients	3.3E-08	5.7E-07	
Loss of Offsite Power (Including SBO)	1.0E-07	2.8E-07	
ATWS	1.7E-07	4.9E-08	
Interfacing System LOCA	5.2E-10	5.2E-10	
Vessel Rupture	1.0E-07	1.0E-07	
Total	6.7E-07	1.7E-06	

ABB

System 80+ Standard Plant
Level I Model Sensitivity Analyses

Sensitivity Case	Core Damage Frequency-
Base	1.7E-06
Increase all operator error rates by factor of 10	9.4E-06
Set HEP for all operator actions outside control room to 1.0	1.9E-06
Increase all MOV failure rates and CCF rates by factor of 10	8.5E-06
Use large LOCA SIT model for Medium LOCA	1.7E-06
Aggressive secondary cooldown not feasible for Small LOCA or SGTR	6.7E-06
RCP Seal LOCA for station blackout	1.7E-06
Set test and maintenance unavailability to 0.0	1.7E-06
Increase probability of adverse MTC for ATWS to 0.1	2.2E-06

ABB

System 80+ Standard Plant
Level I Model Sensitivity Analyses (Cont.)

Sensitivity Case	Core Damage Frequency
Base	1.7E-06
Increase LOOP frequency by factor of 10	2.0E-06
Loss of Grid frequency set to 0.15 per year	1.8E-06
Vessel Rupture not credible	1.5E-06
Set all CCF ratesto 0.0 except for diesels and batteries	2.4E-07
Set all CCF to 0.0 except for diesels and batteries and set vessel rupture to 0.0	1.4E-07

ABB

System 80+ Standard Plant
Comparison of Shutdown PRAs

Event	System 80+	NSAC-84	NUREG/CR-5015	Seabrook
Total CDF	8.4E-07	1.8E-05	5.2E-05	4.5E-05
Loss of DHR	23%	71%	82%	61%
LOCA	16%	10%	8%	18%
LOOP	25%	0.7%	10%	6%
Fire	36%			4%
Other		18%		11%

ABB

System 80+ Standard Plant
Shutdown Risk Evaluation

- Design Features Contributing to A Reduced Shutdown Risk

  - Two Safety Injection Pumps Operable
  - Capability to Inject to the RCS Via the SCS
  - Safety Depressurization System
  - Containment Spray Pump Doubles as a Shutdown Cooling Pump
  - IRWST Acts as a Sump in a LOCA
  - Alternate AC Source
  - Dedicated SCS Train Independent of LPSI
  - Technical Specifications for Shutdown Modes

ABB

System 80+ Standard Plant Shutdown Risk Evaluation (Cont.)

- Instrumentation Added as Result of Shutdown Risk Evaluation
  - Delta P Based Narrow Range RCS Water Level (2)
  - HJTC Based RCS Water Level and Temperature (2)

ABB

System 80+ Standard Plant Shutdown Risk Evaluation (Cont.)

- More Than Twenty Technical Specifications Modified to Address Shutdown Modes

- Examples of Changes:

  - Two SCS Divisions to Be Operable And At Least One Division in Operation In Mode 6
  - Two SIS Trains, With one Pump in Each Division Required to Be Operable in Modes 4, 5, and 6
  - Containment to be Closed during Reduced Inventory Operation

ABB

System 80+ Standard Plant Shutdown Risk EValuation (Cont.)

- Procedural Guidance Developed For Shutdown Operations
  - Reduced Inventory Operations
  - Coping With Loss of DHR
  - Detecting and Mitigating RCS Drain Down Events
  - Outage Maintenance
  - Fire Protection
  - RCS Cooling Using Feed and Bleed
- Shutdown Operations Procedural Guidance Included in Appendix B of System 80+ EOGs

ABB

System 80+ Standard Plant External Events Analysis Results

	Core Damage Frequency	
Tornado strike - Total		2.5E-07
Internal Fires (Basic)	6.1E-08	
Fire-Induced RCP Seal LOCA	5.2E-10	
Fire Inside Containment	1.34E-09	
Fire - TOTAL (scoping estimate)		6.3E-08
Internal Flood (Basic)	1.3E-08	
Flood-Induced RCP Seal LOCA	1.2E-10	
Flood - TOTAL (scoping estimate)		1.3E-08

ABB

System 80+ Standard Plant
Seismic Margins Assessment Results

- Plant HCLPF is 0.73 g (Goal > 0.50g)

  - Dominant Contributor is Seismically Induced Sliding/Overturning of Containment Shell

  - Second Dominant Seismic Sequence is LOCA in Excess of ECCS Capacitywith HCLPF of 0.86 g. This Event Includes Seismically Induced Failure of RCP Supports

ABB

System 80+ Standard Plant
Comparison of PRA Results With IPEs

	IPE Average	System 80+
Core Damage Frequency (Internal Events + Flood)	7.8E-5/yr	1.7E-6/yr
LOSP	26%	1.7%
LOCA	24%	43.1%
ATWS	3%	2.9%
Flood	10%	0.8%
ISLOCA	1%	0.03%
SGTR	5%	17.7%
Other Transients	31%	33.7%

ABB

System 80+ Standard Plant Containment Performance

- If Containment Failure Defined as Failure With Above Normal Releases Within First 24 Hours:
  - Containment Reliability = 0.98

- If Containment Failure Defined as Having A Release Greater Than 25 Rem at 1/2 Mile From Reactor:
  - Containment Reliability = 0.973

- If Containment Failure Defined as Any Containment Failure:
  - Containment Reliability = .886

ABB

System 80+ Standard Plant
Level II PRA Results

	Conditional Probability	
Intact Containment for 24 hours	96.5%	
Containment Intact Indefinitely		88.6%
Late Cntm. Failure, Overpressure		0.4%
Late Cntm. Failure, Basemat Meltthrough		7.5%
Subtotal		96.5%
Containment Isolation Failure	2.4%	
Early Containment Failure	1.1%	
Steam Explosion		0.95%
Alpha Mode		0.12%
H2 Burn/Explosion		0.03%
Subtotal		1.1%
TOTAL	100%	

System 80+Standard Plant
Level II Model Sensitivity Analyses

	Cnt Intact Indefinately	Late Cnt. Failure	Early Cnt Failure	Isolation Failure
BASE	88.6	7.9	1.1	2.4
H2 Ignitors Not Available	87.5	8.0	2.1	2.4
Deflagration to Detonation Transition Likely	88.6	7.9	.11	2.4
Low Heat Transfer from Corium to Coolant	87.8	8.8	1.1	2.3
Reduced Probability of Cont. Spray Recovery	88.4	8.2	1.1	2.3
Containment Heat Removal Not Recovered	74.8	21.8	1.1	2.4
Thermally Induced Failure of RCS Piping Always Occcur	88.6	7.9	1.1	2.4
Thermally Induced Failure of RCS Piping Never Occurs	88.6	7.9	1.1	2.4

ABB

System 80+ Standard Plant
Level II Model Sensitivity Analyses (Cont.)

	Cnt Intact Indefinately	Late Cnt. Failure	Early Cnt Failure	Isolation Failure
BASE	88.6	7.9	1.1	2.4
RCS N Not Depressurized by SDS for Sequences with Cycling Relief Valves	88.6	7.9	1.1	2.4
RCS Always Depressurized by SDS for Sequences with Cycling Relief Valves	88.6	7.9	.11	2.4
Increase Containment Isolation Failure Rate to 1.0E-02	88.1	7.8	1.1	3.0
RCS not Depresxsurized by SDS for Medium and High Pressure Sequences	88.6	7.9	1.1	2.4
Increase Probability that Operator Fails to Turn on H2 Ignitors from 3E-02 to 3E-01	88.3	7.9	1.4	2.4

ABB

System 80+ Standard Plant Level III PRA Results

- Frequency of exceeding Whole Body (WB) Dose of 25 Rem at:
  - 1/2 mile from reactor = 5.3E-08/year
  - at 300 meters from reactor = 6.2E-08

ABB

System 80+ Standard Plant
Level III Model Sensitivity Analyses

	Probability of Exceeding 25 Rem at 1/2 Mile
BASE	5.3E-08
Releases occur at Top of Containment	5.0E-08
Releases Occur at Grade Level	5.4E-08
Increase Iodine and Cesium Release Fractions by One Order of Magnitude	6.4E-08
Containment Bypass Releases Unscrubbed	5.3E-08
Increase Containment Isolation Failure Rate by One Order of Magnitude	4.4E-07
Double Basemat Melt-Through Failure Frequency	5.5E-08
Concrete Ablation Failure Occurs at 30 Hours Instead of 65 Hours	5.4E-08
Increase ISLOCA Frequency by Two Orders of Magnitude	1.0E-07

ABB

System 80+ Standard Plant
Previously Identified ACRS Issues

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System 80+ Standard Plant
MOV Fail to Close Issue

- ISSUE: Failure Rate Used For Failure of MOV to Close, Especially For EFW System MOVs, May be too Low.

  - ABB-CE Used Failure Rate of 4.0E-03/Demand for Failure of MOVs to Operate
    - Based on generic data
    - For valves tested on quarterly basis

  - Contention is That Failure Rate for Failure of MOVs to Close Should be 8.0E-02/Demand
    - Failure rate higher due to accident conditions

ABB

System 80+ Standard Plant
MOV Fail to Close Sensitivity Studies

- Case 1: Increase failure rate for ALL MOVs, "Fails to Close" Failure Mode only, from 4.0E-03 to 8.0E-02
  - Base Core Damage Frequency = 1.67E-06/yr
  - Resulting Core Damage Frequency = 3.01E-06/yr
  - CDF Increases by Factor of 1.8

- Case 2: Increase failure rate for ALL EFW MOVs, BOTH "Fails to Close" and "Fails to Open" Failure Modes, from 4.0E-03 to 8.0E-02
  - Base Core Damage Frequency = 1.67E-06/yr
  - Resulting Core Damage Frequency = 4.14E-06/yr
  - CDF Increases by Factor of 2.5

ABB

System 80+ Standard Plant
MOVs Designed For Environment

- MOVs Are Designed To Operate Per Their Specific Location In The Plant
- MOVs Are Purchased And Qualified And Previously Tested For Accident Exposure And Physical Location
- Each Motor Operator Is Designed For Unique Accident Exposure & Physical Location (i.e., inside or outside containment)

ABB

System 80+ Standard Plant
Publically Available MOV Failure Rates

Source	Fail to Open	Fail To Close	Fail to Operate
EPRI ALWR KAG	-	-	4.0E-03/D
NUREG/CR-4639	6.1E-03/D	4.4E-03/D	2.8E-03/D
NUREG/CR-4550	-	-	3.0E-03/D
PVNGS IPE	3.2E-03/D	3.2E-03/D	-
MP2 IPE	2.1E-03/D	2.1E-03/D	-
SONGS IPE	3.0E-03/D	3.0E-03/D	-
IP2 IPE	-	-	1.6E-03/D
ANO2 IPE	-	-	5.8E-03/D
NREP	2.7E-03/D	2.7E-03/D	-
WASH1400	-	-	2.7E-03/D
PSL IPE	6.2E-03/D	2.4E-03/D	-

ABB

System 80+ Standard Plant
MOV Failure Rates - Swedish NPRs

MOV Fails to Change Position

PIPE DIMENSION	BWRs	PWRs
DN < or = 100 mm	7.9E-03/D	5.3E-03/D
100 mm < DN < 200 mm	6.3E-03/D	1.7E-03/D
DN > 200 mm	7.2E-03/D	3.3E-03/D

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System 80+ Standard Plant
2 DGs with AAC vs. 4 DGs Issue

Why does SYSTEM 80+ DESIGN have 2 DGs/Gas Turbine rather than 4 DGs when 4 DGs results in a lower CDF?

ABB

System 80+ Standard Plant
1991 EDS Sensitivity Study

CASE	UNAVAILABILITY 2 DGs with GAS TURBINE	UNAVAILABILITY 4 DGs	BENEFIT FACTOR
Transient	1.42E-04/D	1.29E-04/D	1.1
LOCA	2.70E-03/D	1.93E-03/D	1.4
COST $	$57M	$100M	- $43M

- STUDY DID NOT INCLUDE SUPPORT SYSTEMS, ONLY EDS & BUSES
- STUDY COMPARES RELIABILITY ONLY - DOES NOT TAKE INTO ACCOUNT RISK IMPACT OF SEQUENCES
- COST OF 1 DG CONSERVATIVELY ESTIMATED AT $25M
- COST OF GAS TURBINE ESTIMATED AT $6.5M (from United Engineers & Contractors)

ABB

System 80+ Standard Plant
Advantages Of 2 DG + AAC System

- Benefit Does Not Justify Cost
  - Cost Of 2 DGs Far Exceeds Cost Of 1 Gas Turbine For Slightly Better CDF
  - Benefit Of 4 DGs Is Most Realized For Large LOCAs, Which Only Make Up A Small Percentage Of Total CDF

- Additional Drawbacks With 4 DGs:
  - Larger Plant "Footprint" - Other Costs Imbedded In This
  - More Equipment Means More Complicated Operating Procedures And Operator Actions Which Negatively Impact Safety

ABB

System 80+ Standard Plant
1992 EDS Sensitivity Study

- New sensitivity study was done by DE&S to extend earlier results into an examination of risk impact

- Four separate initiators were examined
  - Loss-of-Offsite Power
  - Small LOCA
  - Large LOCA
  - General Plant Trip

- RESULTS: 2 EDGs with AAC had lower CDF than 4 EDGs without AAC

ABB

System 80+ Standard Plant
1992 EDS Sensitivity Study (Cont'd)

- RESULTS

	CDF
2 EDSs with an AAC Source	1.71E-06/year
4 EDGs without an AAC Source	2.40E-06/year

- EDG Failures dominated by CCFs (based on industry data)

- Diverse AAC more than offsets benefits of 4 Redundant EDGs

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ABB Combustion Engineering

System 80+™ Standard Plant

Section 14.3 - Certified Design Material

S. E. Ritterbusch

ACRS ABB-CE Standard Plant Designs Subcommittee
March 8-9, 1994

ABB

System 80+™ Standard Plant
Certified Design Material

- The Certified Design Material (CDM) is that information that is necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed, and the acceptance criteria met, a facility referencing the certified design will be constructed and will operate in conformity with the design certification, the provisions of the Atomic Energy Act, and the Commission's rules and regulations.

- CESSAR-DC, Section 14.3 summarizes the criteria used by ABB-CE to develop the CDM.

- FSER Section 14.3 provides NRC staff approval of CDM with respect to "necessary and sufficient".

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System 80+™ Standard Plant
Certified Design Material

- **Tier 1 Certified Design Material**
  - Section 1: Introduction and General Provisions
  - Section 2: System Design Descriptions, Figures, and ITAAC
  - Section 3: Non-System Specific Design Descriptions and ITAAC
  - Section 4: Interface Requirements
  - Section 5: Site Parameters

- Development methods and selection criteria for each CDM section in CESSAR-DC Section 14.3

- Selected summaries of CESSAR-DC material incorporated into the CDM

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- Requirement for ITAAC in 10CFR52.47 & 52.97
- SRM on 90-377, "Rqmts for Design Certification Under 10CFR52" - 2/15/91
  - Graded approach for application based on safety significance
  - ITAAC confirm design, and are not basis for safety decision
- Multiple iterations & senior management meetings 1991-1993
- Industry reviews

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