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COMBINED ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON GESSAR II

AND

RELIABILITY AND PROBABILISTIC ASSESSMENT

GESSAR II FDA REVIEW

LOS ANGELES, CALIFORNIA

OCTOBER 19, 1984

REPORTER'S TRANSCRIPT OF PROCEEDINGS

DAVID OKRENT, Chairman of the Subcommittees

JACK EBERSOLE, ACRS Member

C. MICHELSON, ACRS Member

LOS ANGELES, CALIFORNIA; FRIDAY, OCTOBER 19, 1984, 8:30 A.M.

MR. OKRENT: The meeting will now reconvene.

I think we agreed that we would next go into items nine A, B and D, and then item ten. And I don't plan to use more than two hours for those. So let's see how it goes. Whenever you are ready.

MR. QUIRK: My name is Joe Quirk from General Electric. We would like to begin this morning's meeting then with nine which is New Design Features/Philosophy Behind Design Changes. And that will include A, which is a summary of changes and basis. B, the ultimate plant protection system. And D, containment strength.

(Slide 1 shown.)

MR. QUIRK: I hope to put in perspective the presentation with this chart. It looks rather complicated, but I think it will be clear in a minute.

This is a scale of the year and we are going to talk about the new design features that have been added to GESSAR since it began and they are basically three periods of changes. The changes that resulted as a direct result of the NRC review at the PDA stage. There was a period of changes that while that design was being detailed and evolved from a conceptual design to a final design General Electric Company made some changes to improve and we will list those changes. There were then changes that resulted in this design as a result of the final design approval review and most recent change have been made since the severe accident approval review had been initiated by the staff. So there was a preliminary phase and a final phase.

MR. MICHELSON: The detailed design, was that for a particular plant in this country as well?

MR. QUIRK: Yes, it was.

MR. MICHELSON: Which one?

MR. QUIRK: For the Hartsville 16 design.

MR. MICHELSON: After Hartsville was postponed and then eventually canceled did you precede with the design or had it already been fully completed?

MR. QUIRK: It had been substantially completed. I'm saying about 90 percent type completed.

MR. MICHELSON: Have you completed it since then?

MR. QUIRK: I believe we haven't completed the last bit of it, but we have brought our documentation up to a level of readiness where it is all completed up to that point and the remainder would be completed.

MR. MICHELSON: So Hartsville essentially is the design we are dealing with here.

MR. QUIRK: That's right Hartsville and Phipps Bend were both sister plants if you will.

MR. MICHELSON: One more clarification and this is essentially the work that was done by your consultant firm in Los Angeles?

MR. QUIRK: That's right.

MR. MICHELSON: What was their name?

MR. QUIRK: C. F. Brown.

MR. MICHELSON: They no longer deal with this, do they?

MR. QUIRK: That's right, as a result of the cancellation.

MR. MICHELSON: Has all of their documentation been moved to San Jose or what happened to it?

MR. QUIRK: Yes. We have correlated all the design documentation and have it available for continuation if it were to come to that or for reuse if we were to sell another Nuclear Island design.

This chart summarizes changes that were made as a result of the preliminary design review. Now, this is not a complete list because there are many more changes. But what I tried to do was highlight what I felt were the more significant changes that were made so that you could get a feel for what kind of changes resulted at this first stage.

And the first one was the envelope citing parameters were pretty much increased I would say. Based on the idea that this is a standard plant design and it will be designed for most sites in the United States. That would exclude the Rockies and high seismic area such as

California. But the idea was to define a site envelope such that this design could be cited on roughly 85 to 90 percent of the sites in the U.S.

So as a result of that we increase the wind loading, the snow and ice loading and ground water to agree with -- to improve the design and make it so that it could be cited at a large number of sites.

MR. EBERSOLE: Explain the meaning of the elevation of ground water citing. What implications are there in the design about that?

MR. QUIRK: The concern here was that you may get flooding and tend to float the base mat. So they wanted to assure themselves that there wasn't a hazard such that that could occur so there was an interface criterion that deals with that.

MR. EBERSOLE: Thank you.

MR. MICHELSON: Do you require by design to keep the building dry, then, the auxiliary building?

MR. QUIRK: I'm trying to recall now. This was early 70's, but I believe the REG guide is one foot below ground and -- I can't recall that.

MR. EBERSOLE: Let me elaborate a little bit.

At Browns Ferry the flood came in and we had to deal with an eight foot above grade problem -- that's 1968 or 1969. And at that time we attempted to invoke this UPPS process and were thrown out. The one that you have got now. In the light of today's adoption of that system are you going to be able to cope with floods by permitting the system to flood out rather than build these weird walls and doors and things around these buildings?

MR. QUIRK: I would say it may come to that although our design is not to give up the rest of the plant due to flooding. It is to assume it happens and to prevent it and if it happens, to deal with it, such that the remainder of the plant not necessary to shut down is still available.

MR. EBERSOLE: The problem with that which I understand exists at Brown's Ferry today is the water level is pumped down and thus it does not reveal faulting in the inner four seals. In the concrete they experience down there when these bump down systems are shut down the ground water rises for the first time so there is an unreliable problem there in ascertaining if you can in fact -- and you don't ever get one.

MR. QUIRK: This has been looked at. I know the water proving requirements for the portions of the building below grade and everything was specified. So it has been looked at and was not overlooked and dealt with in the review.

MR. MICHELSON: I guess you appreciate though what

Jesse was saying is you don't know that the building waterproofed after another 20-30 years and there is no simple way to test it.

MR. QUIRK: I understand.

Another change that was added at this stage of the review was the leakage control system and basically this was to deal directly with the bypass leakage concern through line leakage, through valves and pipes. So the major penetrations that went into the containment -- we have an exterior external leakage control system that pressurizes the space between isolation valves and make the suction there higher than the pressure in containment so if there was any leakage it would be inward. This was a system that was added at this time.

We also upgraded RCIC system pretty much a paper exercise as we discussed at another time. To the engineering and safety feature status. That was a requirement of the staff at this time. We upgraded fuel building to withstand threat from tornado missiles and increased drywell pressure margin. Originally the drywell design presure was 25 psi. It was increased during this stage of the review to 30.

MR. EBERSOLE: Let me ask you: What is the equivalent to component cooling on your design? You will call it RB60 heavy that was non-safety single track system.

Has anything happened to it?

MR. QUIRK: It is a safety grade in the GESSAR design. The service water system necessary to accomplish safety functions has been upgraded as safety class.

MR. EBERSOLE: This is not the service order. This is closed loop system that has an interposed treated water loop?

MR. QUIRK: But it takes the heat loads from the equipment.

MR. EBERSOLE: It takes the heat out of the containment in the normal mode through RB60 heavy or do you use raw service water?

MR. QUIRK: Out of the containment?

MR. EBERSOLE: Yes. You used to use RB60W. You cooled the seals on the main coolant pumps with this auxiliary closed loop.

MR. QUIRK: You're right. We have two parts of the reactant building closed cooling water system that which is non-safety grade and supports normal operational loads. And that which is safety grade and is required in a postulated accident to remove the loads and keep the safety equipment functioning.

MR. EBERSOLE: Is the pump seal cooling safety grade now.

MR. QUIRK: No, it is not.

MR. CAMP: What is the drywell pressure capability for external pressures higher than the 30?

MR. QUIRK: That certainly is. For pressure conditions in the containment that loaded inward?

MR. CAMP: Yes.

MR. QUIRK: Yes, it is significantly higher than 30. Do we have a chart on that?

MR. HOLTZCLAW: It is going to be coming up again.

MR. EBERSOLE: As a practical matter since you say the main coolant pump seals are not cooled by safety grade equipment, that does not mean that they are dual track pumping systems. It's just not upgraded to seismic and all that I'm trying to address their reliability of the reactor cooling pump sealed cooling system.

MR. QUIRK: The purpose, of course, is to keep seals cooled and they would not degrade and leak excessively and the safety function there is really no safety function of the recirculation system except the pressure boundary function. And after postulated loss of coolant accident the first seconds of the accident we assume momentum of the water to continue.

So the coast down feature of the pump for the first few seconds is defined as safety grade now if you are going to lose component cooling water it will take time to degrade those seals and cause failure of those and the resulting failure is increasingly small and can easily be made up by the RCIC and other systems.

MR. EBERSOLE: I was just thinking about avoiding leakage.

MR. QUIRK: We view that as an operational concern, because it would result in down time and replacement of seals. So we paid close attention to that.

What I tried to do here is give you a flavor that there were some significant changes made as a result of the preliminary review stage of the GESSAR.

(Slide 2 shown.)

MR. QUIRK: The next period was what I called the detailed design phase and that is that the General Electric Company then took their conceptual plant design and then detailed it and during that time some changes were made by General Electric Company and I've listed on this chart some of the types of changes that we made at that time.

First, we increased the allowable primary containment leak rate from three tenths of a percent per day to 1 percent per day. We did this to reflect the reduction and bypass leakage as a result of the leakage control system I just talked about earlier. And we also reduced the magnitude of the safety relief valve loads by 35 percent to reflect in plant test data.

Now what we mean here is that when we went through the PDA stage we established some very conservative dynamic pool swell loads and boundary loads and in the subsequent years as plant -- as these types of plants came on line we did some safety relief valve tests and we were able to demonstrate the conservativeness of the boundary loads established and thus we reduced the loads and left an adequate conservative margin but took out some of the excessive conservativism.

MR. EBERSOLE: Did you set a standard bypass to the condensor? There is a great variability to these things. Have you now a standard bypass?

MR. QUIRK: You mean like a third --

MR. EBERSOLE: Whatever.

MR. QUIRK: Yeah we have 30 percent in the GESSAR design and I think we have an option for higher. But the base design was 30 percent.

We also incorporated stainless steel clotting of the containment vessel in the wetted areas of the suppression pool. We did this to protect against corrosion and decrease the required maintenance and operatability improvement. We added a suppression pool cleanup system to improve the reliability of operations. We incorporated state of the art buckling methodology to improve stability analysis of the containment vessel.

MR. EBERSOLE: Is there a relationship between the power decrement when you trip the pumps and the bypass so that you stabilize on bypass if you trip the pumps?

MR. QUIRK: What pumps are we tripping?

MR. EBERSOLE: Main cooler. You get a power decrement of a substantial amount, right?

MR. QUIRK: That's right.

MR. EBERSOLE: Is that in any way related to the amount of bypass you pick?

MR. QUIRK: I believe the amount of bypass that we picked is based on a transient load rejection and that we then pick a bypass to the steam to the turbine and go on hot standby and if we can fix the failure.

Now, I think your question is if we trip the recirc pumps do we go on the bypass mode.

MR. EBERSOLE: Can we sustain that level of power on bypass?

MR. QUIRK: Anybody know the answer to that?

MR. KNECHT: I'm not sure of the exact answer, but just tripping the recirc pumps of course drops the power way down and we wouldn't really expect the excess -- the turbine will back off from the turbine control system to follow that transient.

MR. EBERSOLE: Where you lost the turbine first and tripped the main pumps and want to go to bypass.

MR. KNECHT: I don't think it is part of the design capacity to bypass the system. It is really a different question.

MR. QUIRK: Getting on these changes the control out drive return line back to the reactor vessel was deleted and this was to guard against nozzle cracking. And we redesigned the heat feedwater starter thermal sleeve to eliminate failure and leakage and facilitate in service inspection. These were the types of changes made during this design period by the General Electric Company.

MR. MICHELSON: In the past as I recall the CRD pumps were considered a source of water for vessel makeup. Having eliminated the vessel return line they are no longer a credible source, I guess.

MR. QUIRK: It still does provide makeup. Reroutes through the feedwater.

MR. MICHELSON: Through the drives themselves.

MR. QUIRK: Well, I believe the discharge lines connects to feedwater line and goes back to the vessel.

MR. MICHELSON: So they are still a credible source.

MR. EBERSOLE: I remember a problem because I think our idea of the RCIC containment cooling pump -- it was not seismically competent so the effects of a seismic event was to lose containment cooling and that rather promptly caused a high pressure in the drywell, which synthesized a loca trip signal. What did you do about that?

MR. QUIRK: Part of the reactor building closed. Cooling water system is safety grade and the part that is not is dropped off and isolated and the other part remains in operation and removes loads.

Now, the heat removal from the containment suppression pool is performed by the RHR system, the suppression pool cooling mode of the RHR system.

MR. EBERSOLE: I'm talking about the heat in the drywell. Where the reactor vessel is. There is a fast rising temperature when you lose containment cooling which was sustained by a single track system.

MR. QUIRK: I understand. What we did on that system is we -- it still in our opinion non-essential, non-safety grade but we have it so that it can be loaded on the diesel so that if you are given a loss of off-site power we can manually assume load a diesel.

MR. EBERSOLE: So you fix the power input to it but that's where you stopped?

MR. QUIRK: That's right.

MR. MICHELSON: What did you do about the rapid temperature rise in the drywell if you remove all atmospheric cooling.

MR. QUIRK: We have analyzed for the worst event like the loss of coolant in the drywell and have environmentally qualified the equipment necessary to perform the functions.

MR. MICHELSON: Approached temperatures of 550 degrees in there if you don't take any heat out because it is a very hot vessel and a lot of hot lines. So you are eventually approaching -- unless you have depressurized the reactor in the meantime you approach the temperature of the fluid which is about 550.

MR. QUIRK: We have gone through postulated event sequences for the full spectrum of large locas and small breaks and defined the worst environmental conditions.

MR. MICHELSON: Those are lower temperature conditions than simply bottling up a reactor at 550 degrees and letting the room go into thermal equilibrium with the reactor eventually.

MR. QUIRK: We have technical specs that would require specialization.

MR. EBERSOLE: Look at recent activities at the Hatchet Plant and notice how the operator never did invoke any kind of technique to lower pressure and temperature. So there was a near disaster.

MR. QUIRK: We are aware of that.

MR. EBERSOLE: I don't know what fell into the cracks but something did.

(Slide 3 shown.)

MR. QUIRK: These are a list of changes that we made during the final design stage of the review and, for example, as we talked a little bit about yesterday the addition of redundant remote shutdown station.

We had a remote shutdown station, and we argued extensively that it did not need to be redundant. But after going through a final design approval review and a lot of questions, the staff required us to make that change.

We also changed the piping material to avoid intragrandular stress corrosion cracking. We gave a presentation on that yesterday by Doctor Je ry Gordon.

We added requirements for anticipated transients without scram, and what I mean is we have committed to meet the rule that has been issued recently. And we have added redundant and the redundant adverse instrumentation and added redundant vent and drain valves to the scram discharge volume as a result of the abnormal operational occurrence at Brown's Ferry and we also talked a little bit about that yesterday.

MR. EBERSOLE: I think that redundant ought to be amplified. Redundant to close and it is now coincident to open.

Am I correct?

MR. QUIRK: Well --

MR. EBERSOLE: You put two thousand series on the drains and vents?

MR. QUIRK: I've a picture here.

MR. EBERSOLE: In other words --

MR. QUIRK: We will have to get back to you. We have an action --

(Slide 4 shown.)

MR. QUIRK: -- diagram of the BWR/6.

MR. EBERSOLE: There they are. They are this, both of them, so you are more anxious to close than ever?

MR. QUIRK: Oh, yes.

MR. EBERSOLE: That is additive to the thesis that you will be closed when you want to be closed?

MR. QUIRK: Yes. I hear you now.

We have talked about this before, Mr. Ebersole, and we have level indication switches here on both sides, redundant and adverse switches, so we can tell clearly where the level is at any time in this discharge volume. And the free volume --

You have heard it before.

MR. EBERSOLE: I've heard it before.

(Slide 5 shown.)

MR. QUIRK: The review stage that we are undergoing now is the severe accident approval stage and there has been a number of changes that have come about as a result of that.

We have made improvements to the BWR/6 design to meet all the reg and new reg and SRP requirements that came about as a result of Three Mile Island. And I have another chart on that in a minute to detail that.

We have upgraded the control room human factors design. Put in the SPDS and have met the latest staff guidelines on that. And we have gone through a rigorous cost benefit analysis to assess some 83 design changes that the staff has given to GE. And this is a list that's made up of just about anything one could conceive, including what is happening internationally, what happened in Europe, what were some of the things that happened in Japan, and we have addressed all of those.

And as a result of that assessment we have concluded that the natural evolution of BWR has included about a fourth of those changes that were on that list. And when we add the ultimate plant protection system it addresses maybe another fourth. So the design has been demonstrated to be acceptable.

MR. EBERSOLE: Along about the same time that we were trying to crank in the system at Brown's Ferry I had heard but never confirmed that Quad Cities put in a version of this system. Is that true, also for the flood?

MR. QUIRK: I don't know. I don't know.

MR. EBERSOLE: I wonder if you all could ascertain whether or not that is so.

MR. QUIRK: Okay. Limerick, for example, I believe has provisions here.

MR. EBERSOLE: Yes.

MR. QUIRK: To accommodate a lot of this.

MR. EBERSOLE: Years and years ago I understand Quad Cities was successful, but only under the rigors of a flood condition.

MR. MICHELSON: One of the interesting things about an UPPS type system is that you are proposing to use river water or whatever. Any particular reason why you didn't try to use as a first choice some nice clean water sources and only as a last choice the river?

MR. QUIRK: No. I would say that we looked at this as a last ditch investment protection system that should never be used.

MR. MICHELSON: The problem is of course the poorer the water source the greater the reluctance to ever fall back to it, and maybe one would be inclined to wait too long. It isn't any harder to put a fire engine up to a condensate tank than it is to put it to a fire pump. But Limerick elected to attach it to the fire pump.

Do you have any thoughts on the matter?

MR. QUIRK: My thoughts, Mr. Michelson, are that there are a compliment of diverse water sources on the BWR that are going to be adequate for any -- well, in my opinion -- would be adequate in any postulated event including loss of coolant accident and degraded transients, and that we have the existing capability to insure high quality water during these postulated events.

What we are talking about here is really an extra capability of the design that, given just -- just wipe out, you know, existing logic and systems and sources, that you have the capability given that to still accomplish the three safety functions.

MR. MICHELSON: Maybe I misunderstood then. I thought the purpose of this system was for instance in the case you have lost your AC power off-site and on-site that you have got a means now of getting water to the reactor. You have got plenty of water in storage but you can't get any of these sources into the reactor nor have provisions been made to back up a fire truck and pump it out of the tanks into the machine. The only provision is to pump it out of the river into the machine.

(Slide 6 shown.)

MR. QUIRK: I was going to get to this. I think we ought to do it now.

MR. EBERSOLE: Joe, could you give me a feeling for when you started the design of this system and how much manpower and consideration you have given to it up to now?

I have a feeling, partly because I did not get a full memo on it, that you haven't been working on it too long. Am I correct?

MR. QUIRK: That's correct.

MR. EBERSOLE: Like when did you start it?

MR. QUIRK: First of all, it is a conceptual design right now. That's all it is.

MR. EBERSOLE: When was it conceived of being worthy of consideration? Like the mid year?

MR. QUIRK: The first part of this year we actively started the debates with NGE.

MR. EBERSOLE: I sort of see it in its infancy.

MR. QUIRK: We are committing to provide it and there is a discussion that's going to come up later on -- insights on PRA's and things like that -- and I think I would like for some of this discussion for that.

MR. EBERSOLE: Sure.

MR. QUIRK: Let's go through the ultimate plant protection system, what it is aimed to do, and then I would like to address Mr. Michelson's concerns on blackout capability and how this matched up with that.

MR. MICHELSON: That is the only event for which it is presently designed; is that correct? I mean, the design objective was loss of all AC power.

Let me state it differently: What is this system for, what events have you specifically designed it for?

MR. QUIRK: Well, let me tell you what it can be. I don't know how to answer that because it can do a hell of a lot, as we observed yesterday, and I don't intend to know all the postulated events on which we could rely on this, but they are numerous.

MR. MICHELSON: I was only thinking in terms of what I've been reading, and I've only found one objective in your material and that was loss of all AC power.

The probabilities were coming up a little high -- risks, rather, were coming up high. And that's the only reason I can find for this system going in so far, so that's why it wasn't designed for fire and so forth.

So what I was asking was: What was your design objective beyond, let me say, beyond loss of all AC power? Any other objective presently for which it is being designed?

There is a lot of usage you might put it to, but you did not design it for adequate protection for fire. You can't claim it for flood because it wasn't designed for flood. So I gather it was only designed for one purpose and that's the loss of all AC power?

MR. QUIRK: That's correct.

The ultimate plant protection system provides core makeup water with pumping capability provided from outside the containment. Now, that is shown here. This is the makeup pump located outside the containment.

And in our conceptual design, Mr. Michelson, we have allowed that the diesel fire pumps can be connected to this pump and so besides the three AC divisions elsewhere in the plant, you can now have the diesels that run the fire pumps on the plant be the first source, primary source, for this pump.

But in the event those diesels are not available as well, we have auxiliary fire truck connection that could hook up to a fire truck and provide cooling and then it follows this route, goes through the auxiliary building -- this is all a new line here -- and goes in through an air operated valve that is normally locked, closed and not utilized.

We would, through air, open this valve and permit the water to go directly into the reactor, using the L -- existing LPCS injection path. So that is the flow path of this system into the reactor vessel.

Because this is an air operated valve it requires no electrical controls to operate. Not AC dependent or DC dependent.

In order to use it you must depressurize the reactor vessel, bottle air supply in the auxiliary building that would provide source to lift the safety relief valves and direct the depressurization effluent into the suppression pool. Pneumatic air.

It is also independent of any AC or DC power. So we have accomplished two of these functions, depressurize the vessel and provide makeup without any electric power.

The third important function of this system, then, besides depressurizing, besides cooling -- providing water into the core, is to remove heat. And we will allow the pool to boil and will vent then the containment and allow the heat to be removed through natural convection process.

MR. MICHELSON: In view of the slow response required, why did you even bother to use the air operated valve for the injection? Why not just a manually operated valve?

MR. QUIRK: I think these are some considerations that we might want to re-visit. Maybe even hydraulic. But -- part of our problem here is an air operated actuator and how do you seismically design that and who builds them? And if you want to upgrade, say, consider upgrading the system to seismic we run into these types of problems.

MR. EBERSOLE: I suspect you could use the manual wheel as an auxiliary, but that would be driven by the various -- in other words, it could be hydraulic and that eliminates the system. If you have got water to pump you have got water to open the valve.

MR. OKRENT: Could I ask: The material that's outside of the fuel building, in some other building or where is it housed conceptually?

MR. QUIRK: Let's see if I can answer that.

Here is our fuel building and we are showing that the air bottle supply is located in the fuel building.

This would be the reactor building, containment. It is outside the auxiliary building such as that connection and that pump.

MR. OKRENT: The fire truck connection and diesel driven fire pumps, what kind of a building are they in?

MR. QUIRK: We haven't established that at this time. I would think this would be external to the building so that you could just roll the truck up adjacent to it, connect with it.

Have we located -- have we placed location of that pump? We haven't.

MR. EBERSOLE: The equipment is scattered around which could be unified. There is not very much of it. And it could be encapsulated and protected and numerous other things. Made immune to fire virtually anyplace in the plant, and be sort of a last ditch central station to take it home, if you just apply yourself to it at this time now and look at it more nearly not as a tack-on but as an integral design feature.

Will those levels devices always be functional?

MR. QUIRK: Yes, they will. We have added these to make them so we don't need electrical power and DC power.

MR. EBERSOLE: I see the beginnings of a small package which could be re-looked, re-visited as a central shutdown feature.

MR. OKRENT: One reason why I asked about the location, if it is out of doors then it is subject to cold weather, and I can recall a fire at the new McCormick building in Chicago, which was itself constructed with reinforced concrete and so forth but burned to the ground for two reasons. There were flammables inside and it was a cold winter day and the firemen couldn't get the water going.

MR. EBERSOLE: From the standpoint of investment versus function, I think you have got a big bargain here that deserves a a little more consideration than just scattering around the corners.

MR. QUIRK: We are listening very closely to some of your observations.

MR. EBERSOLE: It is a bargain.

MR. QUIRK: In response to Mr. Michelson's question regarding blackout, I hope all this fits on here.

(Slide 7 shown.)

MR. QUIRK: This is kind of a summary. Let me drop off the last column and we will come back and pick it up.

By the way, this is not in your handout. Sorry.

What I've down here is identify a time frame and very arbitrarily I assume zero to four hours than greater than four hours and the condition is complete loss of off-site AC power. On-site AC power.

And I have listed the systems, essential safety functions and instrumentation.

Of the essential safety functions we are going to talk about core makeup, depressurization and heat removal. For the first, say, zero to four hours we haven't placed a steam driven reactor core isolation cooling system safety grade. It has makeup from the condensate tank or from the suppression pool.

It is a high pressure system so it can make up the vessel under high pressure conditions and all the way down to about 100, 125 p.s.i. And it really has no containment heat removal capability. The heat is stored inside containment during this period and the instrumentation for this system is DC power operated.

MR. MICHELSON: During that period, now, there is no requirement to depressurize, is there?

MR. QUIRK: That's right.

MR. MICHELSON: So your entire system is still at temperature and pressure?

MR. QUIRK: Right.

MR. KNECHT: That's not quite right. We do rely on the operator during that four hour period or maybe over that entire period to slowly depressurize the vessel down to maybe two hundred pounds or so, not for the operation of RCIC but to take the heat load off the drywell. So we want to bring the pressure down and release that temperature.

MR. MICHELSON: Removing the energy over into the suppression pool, of course?

MR. KNECHT: That's correct.

MR. MICHELSON: Is there some kind of standing instruction that says if you are in a power blackout to start depressurizing?

MR. KNECHT: There will be those operating instructions that would be established.

MR. MICHELSON: But that is your intent, though, to upon experiencing a power blackout of 30 minutes duration or greater --

MR. KNECHT: There is a time period.

MR. MICHELSON: That time period four hours or is that time period 30 minutes?

MR. KNECHT: More like 30 minutes.

MR. MICHELSON: By the time four hours comes by will you be down to two hundred pounds?

MR. KNECHT: Yes.

MR. EBERSOLE: Do you anticipate you can go back now to the ironclad requirement and relax those somewhat?

It occurs to me if this system was in place at Brown's Ferry fire they would have put it out in five minutes because they knew they had a blackout. But they didn't. So it burned on for six hours.

MR. QUIRK: Can we talk about that on the next slide, because I do not see the benefit of using this system for fire protection, that the whole process is to direct water in on the core, under loss of all electrical power, AC power, and we are going to look at events in the design in a minute. And to come up with a piping scheme that may direct it out to other areas of the plant may detract from the simplicity of the system.

MR. EBERSOLE: What do you mean?

MR. QUIRK: When I hear you say fire protection --

MR. EBERSOLE: I'm talking about --

MR. QUIRK: Given a fire can this make up to the vessel?

MR. EBERSOLE: Irrespective of whether we are successful or not, I'm going to keep the core cool anyway.

MR. QUIRK: It can indeed do that.

MR. EBERSOLE: I'm not talking about fire protection.

MR. QUIRK: I'm sorry. And most of my comments yesterday were --

MR. EBERSOLE: Yes, I gathered as much.

MR. QUIRK: Let's see. I didn't want to talk about this four hour period because I think there is some misunderstanding.

Initially in FSAR and GESSAR says two hour capability, standard review plans say two hour capability, documented four hour capability. Now, during our review it became apparent to me if I demonstrate that this system has a capability much greater than four hours, for example, ten hours, that a lot of these questions on PRA become less important, such as initiating event frequencies. You don't have to argue over data base so much.

So I agreed to provide a report to the commission that documented what kind of capability our existing plant had. And that report said ten hours and it also listed some modifications that could be made that would facilitate the operator to conduct this action from the control room.

Now, this was all before any debate on ultimate plant protection system ever came about. So this ultimate plant protection system was not on the scene at all, so I was trying to show an existing capability around ten hours.

As we evolved through our review process on severe accidents we started talking about ultimate plant protection system. And we have come full circle such that the General Electric Company on the GESSAR design has committed to provide the system and we have kept only the conceptual design here.

As a result of that commitment, I've gone back and looked at the capability of the RCIC system and I don't think it should have a capability, arbitrary capability at ten hours because all we have got to do is survive the near term, and we have the existing capability for that, and we have indefinite blackout capability.

So I would hope to back off on any changes suggested of the RCIC system to get it to a ten hour capability because I've offered something better and I don't think I need to do both.

MR. EBERSOLE: This is one example of which I was trying to suggest there might be many where you could back off.

MR. MICHELSON: Just to expand my understanding a little bit, in the unlikely event RCIC experiences a single failure what is the basic plan then? During the power blackout RCIC fails to start, what is the basic plan at that point?

MR. EBERSOLE: Well --

MR. MICHELSON: I want his answer. I don't want yours.

The reliability of RCIC is not really that great. It seems like it is a credible single failure during the power blackout, but is there a plan for that?

MR. QUIRK: I would like Don Knecht to answer that.

MR. KNECHT: Obviously on a station blackout if the RCIC failed to start we would be left with no injection with the exception of UPPS. The -- since we haven't located it and it is still conceptual, the time it takes to get the system in operation is still uncertain. But we believe that within 30 minutes the system should be operational.

And the operator would know that RCIC has failed to start and at least have tried it once to restart it, which takes just a matter of a few minutes. So within about 30 minutes we think the system could be on line.

Now, that is sufficient time to avoid core damage in the boildown process in absence of breaks or something else that's concurrent with the blackout.

We believe it is a backup, although the timing is pretty critical. We appreciate that.

MR. MICHELSON: That's why I was wondering if it was sufficiently credible, it was worthwhile to have an emergency plan on what do you do if you have a blackout and RCIC won't run.

MR. KNECHT: The operator is to start going to the station or whatever.

MR. MICHELSON: It would be based on enough analysis to verify that it would keep you out of trouble.

I think it is a credible single failure because RCIC does not have that sterling a start record.

MR. EBERSOLE: I think we have got to converge on an issue and settle it.

Power blackout, to me, I thought, was in fact diverse and would not be treated like Eastern Airlines treated its ore pumps, you wouldn't maintain it or otherwise subject it to common load failure, that it could be excluded from the power blackout configuration. But then I found unfortunately at River Bend it was getting cooling water from one of other two diesels, which I'm sure you will fix.

Now, if you fix it and if you provide diversity and maintenance and fuel and all the good things that make it diverse, is it not fair to say it is part of the blackout complex?

MR. QUIRK: Absolutely.

MR. EBERSOLE: So you would use it first?

MR. QUIRK: Yes. In fact, you know, the -- there is a concern -- General Electric has been in disfavor at times with their utility customers because we are licensing a plant here that's designed for the future and I've just gone through all the changes that we have made to this through the review process and the staff sometimes doesn't differentiate between shoulds and coulds and puts backfit on operating plants and it is not viewed very happily by the utility that General Electric really was the instigator.

The words may not be right, but we have been in the doghouse before.

Now, the question here is backfit and the point just made. We have made steps in the BWR/6 design to have the third HPCS diesel generator a different type of manufacture than the other two, separated from the other two, and therefore, we believe diverse and independent and it should be the solution for existing BWR/6 plants for blackout. And we firmly believe that and we believe it for the circumstances.

MR. EBERSOLE: In order to realize that of course it has to be treated entirely differently. It has got to have a separate flexible network.

MR. QUIRK: Yes. And I can tell you that we have specified through interface documents in essence what you just said and these are guidelines for the customer. And to my knowledge most of them follow them. And in this case I was surprised, as you know, because I was corrected on the record.

MR. EBERSOLE: Well, I think it is unfortunate but true that the vendors and utilities don't view the line very well and it is obligatory on GE to come along behind and scrape up their messes.

MR. QUIRK: I don't believe that to be the case. I don't think it is obligatory.

MR. EBERSOLE: Is the third diesel a diverse package not to be included in the total station blackout, provided of course it is so properly designed? Keep the electrical systems completely intact and separated, no common trays, no whatever?

MR. RUBIN: It is diverse, of course, separate. We did include the common mode failure in our modeling, the two similar diesels. Less likely we thought, not incredible, was common load failure of all three.

MR. EBERSOLE: Do they use common fuel systems?

MR. RUBIN: Separate fuel systems.

MR. MICHELSON: But not separate in the sense that all are coming from a common source of fuel oil out in the yard?

MR. EBERSOLE: Is that so?

MR. MICHELSON: You don't order separate oil from separate companies, for instance, to be delivered to the yard?

MR. RUBIN: We --

MR. MICHELSON: If there is water in the fuel oil there is water in all three?

MR. RUBIN: We had inquired about that and I don't think we have an answer to that.

MR. MICHELSON: Combustion air is common to all three, wind storm has come through and there is a lot of leaves, or whatever, it is common combustion? So even through they are diverse they are still diesel engines -- certainly not non-existent.

MR. QUIRK: Although the air is of course common. I don't know how to -- but the intake points are different.

MR. MICHELSON: Admittedly. I'm stretching it, but it is a point.

MR. RUBIN: Certainly a better situation.

MR. EBERSOLE: You don't send the same man the same day to adjust the governors on all three, would you?

MR. RUBIN: Sure.

MR. MICHELSON: Do you design the heating ventilating air conditioning as three separate systems?

Let me say that differently.

Is the diesel engine completely separate in terms of environmental control?

MR. KNECHT: Yes.

MR. MICHELSON: That's part of your scope, isn't it, and design?

MR. KNECHT: Yes. The HPCS diesel room and the other divisional diesel rooms are all part of the GESSAR scope and they are in separate buildings and have separate -- totally independent.

MR. EBERSOLE: Why isn't the third diesel radiator cooled? If you do it then you are independent of service water. Is there any reason it shouldn't be radiator cooled like the standard industrial big engine?

MR. KNECHT: I don't have an answer. It has its own dedicated service water. GESSAR design is different.

MR. EBERSOLE: I hear you, but I would rather know that it has got its own --

MR. KNECHT: It did have its own loop.

MR. MICHELSON: You can afford the loss of off-site power, on-site power and RCIC and still survive with the UPPS system, so that's all you really need to say.

MR. KNECHT: It is just an extra level of protection.

MR. MICHELSON: That's right, but you do need to ultimately plan on loss of RCIC and fall back to UPPS and you say you are taking care of that?

MR. KNECHT: Yes.

MR. EBERSOLE: Isn't the third diesel relatively small? Not a monster. It is in the size range why --

MR. KNECHT: You are talking --

MR. EBERSOLE: Yes, it is in the size range where radiator cooling is practical.

MR. KNECHT: I believe it is something like four or five KW.

MR. OKRENT: We have one hour left for our carryover of yesterday's agenda and you do want to leave time for items 10 and 9-B, so let's try to keep the remaining discussion on UPPS as pointed as we can.

MR. QUIRK: This is the last slide.

(Slide 8 shown.)

MR. QUIRK: It is why we did it. I refer to this as a chimney chart but what it really is is these are the postulated degraded events in the plant and this the assessed frequency of core damage per reactor here.

And the chimney that is shown in white is the probability of these events without UPPS. And the crosshatch chimneys are the changes in those probabilities because of UPPS, as a result of UPPS.

And the bottom line is that we reduced the overall core melt probability by about a factor of ten. We are going to talk later about insights and PRA's.

We could not justify the UPPS system on a cost benefit basis and that, you know -- that speaks highly, I think, for the BWR/6 Mark III design, that we have shown it has a low core melt probability.

Now, Mr. Okrent and other members of the subcommittee are skeptical of numbers and I think that's good, and I think GE engineers are skeptical of numbers.

And when you consider that and uncertainties and other things, we stepped away from that chart and said, well, if you could add this system to the extent that we have talked about today, and get these results, it just seems like an enhancement of investment protection and something that is desirable. And we have committed to do that and that's basically why we have done it.

I would like to address the subject of hydrogen control. Insights and PRA show us that you shouldn't spend a nickel on this design with its feature for hydrogen control. We didn't.

We went in and offered not to do anything in that area. And in fact one subcommittee member stated yesterday that if you have a design that may have superior -- I don't think that was the word -- improved capability for prevention, does it get some break on mitigation. I don't think I heard the answer to that question yesterday, but we offered UPPS in that mind that we are doing what we believe is responsible as engineers in reducing the overall core melt probability.

We have demonstrated that even if you assume a hydrogen event occurs, we have done containment structural analysis to show that failure location in the containment will not drain the pool or fail the drywell barrier such that the outside consequences, given pool scrubbing, are acceptable.

Why should we put in igniters systems which use AC power and the dominant sequence is lost of AC power. It just doesn't appear to be a cost-beneficial thing to do. So we offered to put in UPPS and hope for consideration of no igniters. And I would be very interested in hearing what the subcommittee members think of that proposition.

MR. OKRENT: Well, if I can bring to mind one or more recent committee letters relating to the question of severe accident policy and containment and so forth, they have emphasized, in the committee's opinion, the importance of defense in-depth and that one should not weaken, let's say, one aspect of that defense because of a seeming or perhaps real gain in another area.

The committee has said it believed that prevention and mitigation are important. As you know, it has been urging on the staff and the commission that containment performance criteria be developed. It has, on various occasions, indicated that it thought it important that there be a very good containment capability given a core melt for future plants -- well, for existing plants too, but certainly for future plants.

So I think the committee in various ways has been consistent in recommending this approach of defense of depth, including defense against severe core accidents, with containment.

I realize that, to the person who thinks he has achieved a very small medium estimate of core damage frequency, he may -- the question can arise: Why do I have to go through a wide range of scenarios and try to have my containment cope with the bulk of these, even though I'm calculating them to be so low in probability, et cetera?

I guess, speaking for myself here, I really think it is a very considerable need, from the utility's point of view, to try to get the chance of an accident down. From the economic point of view, both long down times are very expensive. And if you can get it down there and -- they believe it and if the investment community believes it, you may have a better chance of getting money to build new plants, I would say.

But from the point of view of public protection, with the very considerable uncertainties that remain, and the things that we are unable to treat in the subjectivity of much of the PRA, and so forth, I think you will find the reception of the public to a plant that is counting on prevention and on containment would be almost the wrong direction to go. That's my own feeling.

In other words, if they were given some kind of a choice, unless you could demonstrate unequivocably, you just can't have an accident. Well, I've seen my --

Let me finish just a bit.

I used to think that if you have a hundred kilowatt reactor in which you could lose the water and the reactor would remain cool just from the air, and which had a limited amount of excess reactivity in the reactor, so that it wouldn't get into trouble that way, the reactor like this would be almost like falling off a log to license or relicense.

I've seen years spent in reviewing safety questions for such a reactor. So there is a public interest in what might be released even from reactors as small as that, and there certainly will be from large reactors. And there is no way by experience that you are going to be able to demonstrate the claimed low frequency of core melt. I mean, it is strictly a calculated number.

MR. QUIRK: Mr. Okrent, the function of containment of course is to protect the health and safety of the public for very serious degraded core melt events that we are postulating. And that function can be accomplished by barriers or by filtration and release.

And I think you are referring implicitly in your comments that a containment should be -- should fully mitigate the postulated hydrogen events with intact barriers. Maybe I'm misreading your comments.

MR. OKRENT: What the committee has said is that it thinks there should be prevention and mitigation. It has suggested that the staff develop containment performance criteria, given the core melt accident.

These did not -- the committee never suggested that there be a criterion of no release, given no matter what core melt accident occurred. You wouldn't find that in anything the committee has written, but it has suggested that there be some containment performance criterion which would at least provide guidance as to what degree of protection from the containment, given the range of core melt scenarios, might at least represent a threshold of acceptabilty, if that's what the criterion said, or so forth.

There inevitably are some physically possible accidents for which you find it impractical to provide containment, so one knows there is not going to be a perfect containment capability, at least not in the current approach to reactors.

MR. EBERSOLE: Awhile ago you mentioned that the initial system is subject to the same failure that created the need for it, and I recall yesterday one of my colleagues said I was in the patching game. I was identifying a missing element of an integral design.

That's a different piece of work. But not the patching game.

But after hearing what this thing will do in the presence of a meltdown, I began to look on the containment, that is not the drywell portion but the exterior portion, as something in place that causes me as many problems as it improves on.

You are saying that if you take the effluent through the suppression pool even in the case of a core melt you are in pretty good shape. So I began to say what is the external shell for, because to close it is a problem, in the context the heat rejection, and to simply collect the effluent from the suppression pool and possibly control it or effluent processing is worthy of consideration.

You said in essence yesterday I don't need containment even for the worst case.

MR. QUIRK: I did and I mean it and I believe it. But that doesn't take me to the conclusion that I ought to do away with it.

I think the containment is the suspenders and the prevention systems are the belts, if you will, and that concept is good. I'm not proposing we delete containment.

What I am proposing is that in the very far interspectrum low probability events that would threaten that barrier, I'm saying let it be threatened and follow through the consequences and if the consequences are acceptable then say so what?

Now, I don't ever expect to get to that situation or that series of events so I feel that's an acceptable answer, but if my judgment is wrong and we came there anyway, with the research that we have done with pool scrubbing, I believe the pool is the filter and the resultant release is acceptable, and I feel comfortable with that position.

MR. EBERSOLE: The containment has always been a two way sword. For a long, long time people didn't realize they had to let the heat out.

MR. QUIRK: I was wondering that if -- let me ask a rhetorical question.

If plants are required to fully mitigate postulated hydrogen events with intact barriers would those plants have to have a capability to provide indefinite blackout? I just hope that would be food for thought. I don't really want an answer.

Mr. Michelson, could I get your opinion on this question as to the igniters.

MR. MICHELSON: Well, I think I would have to say I agree with David Okrent on this, that I'm concerned for instance with the UPPS system as you proposed it, it is not -- it is not single failure proof obviously. For instance, you must get the containment open. We haven't delved into the problems of how do you get the purge valves open, the line valves. We are dealing with how do you get these open, where do you purge, to what pressures do you start at? There are lots of details that I think until they are worked out it is hard to get a good comfort on the UPPS systems. And I can see a number of places where it might fail so I still think you have to fall back to defense in-depth, which means diversity and multiplicity of ways of handling these events within reason.

Containment and igniters are not necessarily unreasonable steps. At least they certainly are possible to do, practical to do, and not overly expensive to do. At least in the igniters part.

MR. EBERSOLE: Joe, let me ask you what the power -- consumption power of the igniters and if you say more than ten kilowatts I'm going to be surprised.

MR. QUIRK: Anybody know?

MR. KNECHT: No, I don't know.

MR. EBERSOLE: That's almost a domestic power plant.

MR. KNECHT: Did you say ten kilowatts?

MR. EBERSOLE: Yes.

MR. KNECHT: No, I don't believe they are that high. It is just a few kilowatts with well plugs and --

MR. EBERSOLE: We are talking about less than one thousand dollars in a generating unit so it is chicken feed.

MR. KNECHT: Non-qualified non- --

MR. EBERSOLE: Whatever. You can buy them mounted on springs for running in trucks.

MR. QUIRK: Mr. Ebersole, what do you mean when you say APU's?

MR. EBERSOLE: That's a standard expression for any kind of small power unit or vehicles or whatever, aircraft, that's what runs the aircrafts when it's on the ground, services.

MR. MICHELSON: I think you also have to recognize that the UPPS system is not designed for some of these events where in the core could get into serious trouble such as a seismic event, wherein you, for one reason or another, lost both on-site and off-site power. UPPS system is not designed to put out the kind of heat or also not available -- I would like to see some things -- enough things that one of them might possibly work.

MR. EBERSOLE: But certainly when you get around to the UPPS system and look at how simple it is.

MR. MICHELSON: I think it can be upgraded without undo difficulty and made to provide firm answers. If it were then I might reconsider whether you need igniters, but right now it is just another system.

MR. QUIRK: Now you are talking.

MR. MICHELSON: Now, it's just another non-qualified system which helps with diversity and helps with multiplicity but it doesn't necessarily help for a common cause event that gets them all. You have not for instance --

MR. QUIRK: With the present approach and philosophy there is no incentive for us to do because it wasn't in our mind -- if there can be an incentive, if we do say harden this and separate it and using Mr. Ebersole's words bunker it and seismically qualify it, would that obviate the needs for igniters, for example.

MR. MICHELSON: That's the time when you reevaluate now whether you really also need the igniters.

MR. QUIRK: We are at a conceptual stage right now and this is the time to answer those.

MR. EBERSOLE: You seem to have a horror of igniters yet I can't see a big investment in them. They are really just a bunch of spark plugs. What is the horror of igniters; is it just the thesis that you might have to use them?

MR. QUIRK: I think it is the worst band-aid we have ever invented. If you have hydrogen, I shudder to think you are going to turn on igniters because timing is important. If you are late, what happens then?

MR. EBERSOLE: On the other hand there is probably ignition points all over the place anyway.

MR. QUIRK: I suspect that's right. I apologize for extending this time.

MR. OKRENT: It is perfectly fair for you to ask questions. We asked quite a wide range of questions, so why not you have an opportunity.

MR. QUIRK: I couldn't help but notice with amusement I thought tables were reversed here. One member was suggesting deleting containment. And this side was saying no, we would be on opposite sides of the table.

MR. OKRENT: Well, it is. I believe I correctly have stated what has been a continuing ACSR position and in fact the NRC used to talk a great deal about defense in-depth when the severe accident issue arose to the top of the pile after Three Mile Island. It was there before, but I would say the NRC fluttered around as to just where it stood on the severe accidents.

My guess is -- now I'm speculating -- in view of the claims by the large dry containment people that they have containments that are really quite effective, given a core melt, and the claims by GESSAR that it has a containment that's quite effective -- I include the pool in this -- given a core melt, that the NRC may in the not too distant future say, "Yes, and we're also for defense in-depth including severe accidents."

MR. QUIRK: I agree with your statement.

MR. EBERSOLE: I got to go back to your ace in the hole, your oasis at the bridge and don't make him weep. There is not that much in it to make good.

CHAIRMAN OKRENT: Well, look, we had better go on. We have 40 minutes left and we want to cover containment and --

MR. QUIRK: Let's do that. One slide on containment strength. These are the changes that we've made as a result of severe accident review.

(Slide 9 shown.)

MR. QUIRK: The analysis from our previous unmodified design was that all areas of the containment exceed 45 psi service level seal limits except the nuckle region in the head design. It was less than that and this was a problem we had with the staff. They wanted us to increase the strength requirements so that all boundary components of the containment were at least 45 psi we have committed to make that change by modifying the curvature characteristics of the head design and now we meet 45 psi service level seal for the containment boundary.

MR. MICHELSON: Does the containment boundary include the ventilation valves or outboard isolation valve?

MR. QUIRK: Yes, sir.

MR. MICHELSON: Downstream of that you go to steel pipe for awhile then, duct work.

MR. QUIRK: What do you mean outboard? Outside containment?

MR. MICHELSON: Yes.

MR. QUIRK: The containment barrier would change at the out board valves.

MR. MICHELSON: So you are going to 45 pounds up to that point?

MR. QUIRK: Yes, sir.

MR. MICHELSON: Thank you.

MR. CAMP: Are you going to talk about drywell strength any more?

MR. HOLTZCLAW: I wanted to catch Dr. Camp's question here.

He had asked earlier what the drywell strength was for pressure exterior to the drywell and the walls in a portion of the roof slab exceeded 200 psig capability, and there is an area of the roof slab that's the weak link of the drywell but it has a capability of 96 psig.

We are going to be talking about this in a further meeting when we talk about the overall issue of the containment structure analysis, because of the fact that we did make this head design. We have gone back and reanalyzed the whole containment to reestablish the relationship of the various pressure capabilities of various containment locations.

MR. CAMP: The weak point is the top of the drywell.

MR. HOLTZCLAW: It's a portion of the roof slab.

MR. CAMP: For both external and internal loads that's the weak point?

MR. HOLTZCLAW: I just got the numbers here for external loads but I believe it is the weak point for both of those.

MR. CAMP: Is the effect of whether or not the pool dumps out important?

MR. HOLTZCLAW: From the standpoint of the capability it is not as important. It is important in looking at the progression of accident scenarios where you would postulate a failure of the drywell due to say a global detonation. It would fail that roof slab then you would have the potential for pouring cold water or the water in the upper pool on to core debris.

MR. CAMP: Thank you.

(Slide 10 shown.)

MR. QUIRK: And the final discussion item is the control room human factors design, which I think is a strong point in our GESSAR design.

When we came out with the GESSAR design it included the solid state control room implement system that

GE provides, and since then we have improved it by upgrading the control room to facilitate emergency response and we have -- reporting with the staff describing these features and the capability and they were just closing the review on that. And that emergency response system includes the safety parameter display capability.

MR. EBERSOLE: There has been a recent LER come in that highlights a complex problem using solid state equipment as intermixed with electromagnetic, the time coordination of these things, you know, solid state is instantaneous. Old devices have several time lags, and there has been some curious evolutions in the field where they are no longer time coordinated to do things they were thought to be.

So I thought that was a new thing on the scene that needs to be looked at. Solid states has produced some interesting evolutions in the actual experience.

MR. QUIRK: That concludes items 9-A, B and D and we would be ready to go into item ten at this time.

I would like to introduce Dave Foreman.

MR. FOREMAN: This discussion is non-proprietary, yes.

I would direct your attention to a letter that was sent from Mr. Tsutsumi of Tokyo Electric Power Company to Doctor Faulkner dated September 26, 1984. That's the basis for most of the presentation material I have here.

MR. EBERSOLE: Go over those names again. I would like to track that down.

MR. FOREMAN: That's spelled T-s-u-t-s-u-m-i.

MR. EBERSOLE: From whom to whom?

MR. FOREMAN: Dr. Howard Faulkner of NRC.

MR. MICHELSON: Tokyo Electric Power Company.

MR. FOREMAN: Tokyo Electric Power Company, yes.

So the material I have is fairly recent. I think it is important for you to understand that we are still in a design phase and so the information that you will see is subject to change. And, in fact, if you had asked this question a few months ago the answer -- drawings you see would have been different, and if you ask the question a few months from now, they probably will be different again.

(Slide 11 shown.)

MR. FOREMAN: The title was gross differences but I don't know how extensive you want to get into differences so I've used their letter as a starting point. And I have used some judgment in trying to determine what kind of differences you wanted to look at. I think it is important to understand the project itself so that you can understand the configuration that it has.

It started out with an advanced engineering team to perform a feasibility study through the conceptual design of a new type of BWR. It involved five different BWR makers, including one from Japan, one from the United States, one from Sweden and one from Italy. The periods of study were from July of 1978 to June of 1979.

(Slide 12 shown.)

MR. FOREMAN: This was then followed by design phases, phase two, and we are now into phase three, with the objective being design and study from an advanced BWR, based on the results of the feasibility study having two parts, basic design and optomized design, and a test and development phase. The basic design was phase two. Optomized phase, design phase three which we are in now. And three makers: GE, Hitachi and Toshiba. A joint study of six electric power companies and three makers.

MR. EBERSOLE: Give me a shred of information about why there apparently is no interest on the part of German, French and British components in this business?

MR. FOREMAN: I personally can't answer that question. It might have something to do with what Tokyo Electric Power Company is willing to fund.

MR. EBERSOLE: I see.

(Slide 13 shown.)

MR. FOREMAN: The objectives of ABWR development are standardization of plant design, first of all, improvement of the operatability and load following capability, improvement of citing efficiency and financibility, establishment of 1300MWE class plant, the improvement of plant capacity factor more than 80 percent, reduction of exposure to less than two hundred man/rem per reactor year, improvement of reliability and safety.

So as you go through the design some of these things will begin to conflict with one another. Some of them are harder than others. The establishment of a 1300MWE class plant, for example, is a hard requirement that you can't easily measure. Some of the others will have to be traded off.

(Slide 14 shown.)

MR. FOREMAN: I am trying to judge what you mean by gross differences. I think we can focus on six technical features of the ABWR which certainly are different.

The large plant output, 1300MWE class plant, reactor recirc system with the internal pumps, fine motion control rod drive mechanism, improved core and fuel -- although for improved core and fuel you might look on that as an evolution that you had -- emergency core cooling systems, reinforced concrete containment vessel.

MR. EBERSOLE: Anything about the refueling cycle time? Is it still about the same as it was?

MR. FOREMAN: Yes.

(Slide 15 shown.)

MR. MICHELSON: What do you mean by the way it was? What is it presently postulated for the GESSAR?

MR. EBERSOLE: About a year, isn't it?

MR. MICHELSON: I'm not sure.

MR. FOREMAN: Would you say 18 months is our standard?

MR. HOLTZCLAW: Yes.

MR. FOREMAN: In fact, if you had to look at where are we moving with longer fuel cycles you would have to look at the United States.

MR. MICHELSON: 18 months between each refueling or only the first one? Each one? Thank you.

MR. FOREMAN: I'll present some information. Most of it came from the drawings that you see came from that letter, which you can look up.

In the area of reinforced concrete containment vessel, I've been in discussion with the team that is in Japan this week and I've made a judgment that I'll change that drawing.

(Slide 16 shown.)

MR. FOREMAN: The internal pumps system is already used in Europe and so the -- there were number of good reasons for wanting to go to an internal pump system in the ABWR design. They are outlined on this viewgraph.

(Slide 17 shown.)

MR. FOREMAN: One of the primary goals for design in Japan is reduced exposure, so by having no recirc valves or external pumps you reduce the in-service inspection requirement and the exposure from the recirc pipe and reduce operator exposures.

And I'll have another viewgraph that shows the vessel later, but by having no recirculation pipe you cut down on the chance for pipe break and therefore improve the safety of the plant.

Without that recirc pipe valve and external pump you can have a more compact containment, which means it is going to cost less dollars to build and to finance. And that has a way of steamrolling itself.

MR. MICHELSON: On the other side of the coin, haven't you introduced now a potential leak at the bottom of the vessel in the form of shaft seal?

MR. FOREMAN: That's true.

MR. MICHELSON: How did you view that problem?

MR. FOREMAN: Well, the plants are already operating in Europe with good results, and so you are looking at a different kind of problem. It is a seal problem versus a pipe crack problem.

MR. MICHELSON: Recirc line break of course also assured a two-third core coverage?

MR. FOREMAN: You will see in a later viewgraph that is always not also -- we are assuring a core recovery, but in this case you have to postulate the failure of a seal.

MR. MICHELSON: That's right. Which might be of the order of one to 300 gallons a minute?

MR. FOREMAN: That's right, and that has to be taken into account when we look at pumping capacity and that sort of thing in the design.

MR. MICHELSON: This is not an incredible break? Big pipe breaks, even cracks in -- significantly when they have leaked?

MR. FOREMAN: Yes.

There is an added advantage in that when you don't have the pumps to operate, you don't have the jet pump to pump through, you reduce the power for the system. Therefore you -- that improves for financing capability also.

The smaller the inertia of the internal pump you can go to a thyristor inverter and improve your control response. And in all of these things there are positive and negative aspects to the change.

MR. EBERSOLE: Was there special attention given to the seals because of this problem that C[illegible] mentioned, such as guaranteeing limited flow by maybe having static seals which effectively act in a diverse way with the stops, whatever?

MR. FOREMAN: I can't answer that question. I'm sure the team is reviewing that, but I can't answer that question.

(Slide 18 shown.)

MR. FOREMAN: I've thrown in a picture which shows the difference between the internal pump system and jet pump system.

(Slide 19 shown.)

MR. FOREMAN: The next difference which I would like to address is the fine motion control rod drive. The fine motion control rod drive includes the diversity of drive mechanisms, both electric power drive and backup scram, and then it has a hydraulic drive for the scram. It is felt that this improves the scram reliability because of the diversity.

The fine motion drive we feel improves the fuel integrity because you now are no longer moving in four inch increments with each movement of the control rod.

The electricity power gang mode drive improves the plant capacity factor. This is so that you aren't moving control rods in a different fashion, and you can -- that way you can move them in gangs and still not come up against --

There is no wear seal, so therefore it is inspection free. Once again this is a very important feature for the Japanese. It also avoids reactor coolant inflow at scram and reduces exposure during the annual inspection.

MR. EBERSOLE: What if anything happened to the rod drop problem?

MR. FOREMAN: I'm not familiar with the rod drop problem.

MR. EBERSOLE: You know the reason that we have this awkward startup, one rod at a time, that takes a day to implement and prevent gang withdrawal now is, as I understand.

MR. FOREMAN: We have gang withdrawal on the BWR/6.

MR. EBERSOLE: You don't hold with stepout one at a time?

MR. FOREMAN: No.

MR. EBERSOLE: Brown's Ferry comes out one at a time, doesn't it?

MR. QUIRK: No.

MR. EBERSOLE: You know the rod drop problem?

MR. FOREMAN: Yes. And analyzed for that and that's one of the design criteria that we established right up front.

MR. EBERSOLE: Does this thing have to have motion --

velocity --

MR. FOREMAN: This does not have a velocity limitor.

MR. EBERSOLE: And it comes out gang-wise?

MR. FOREMAN: And it comes out gang-wise.

MR. EBERSOLE: You must have gotten rid of the rod drop problem.

MR. FOREMAN: We designed it in just like we do on BWR/6, 5, 4.

MR. MICHELSON: Why do you know longer needed velocity limitor?

MR. FOREMAN: Because we have the electric fine motion drive and also we have designed for the case of a rod drop.

MR. MICHELSON: It is just a lot faster rod drop without a velocity limitor. That's all.

MR. FOREMAN: Yes.

MR. MICHELSON: You are designing now for a much faster rod drop?

MR. FOREMAN: It is within the limitations that are imposed by the safety requirements.

MR. EBERSOLE: You are going to discuss the picture of the new drive? I see it is in the handout.

MR. FOREMAN: To the extent that I can discuss it, I will. I'm certainly not an expert on fine motion contro rod drive.

(Slide 20 shown.)

MR. EBERSOLE: That's the place where you know when you push it up and when you pull it down it becomes decoupled and comes out too fast and isn't worth too much. There have been many suggestions that there should be a circuit up there that tells you when you decouple.

Has it got one?

MR. FOREMAN: No. You see, what we have got is we have got -- we have got the rod up against the electric drive and so that's going to be pushing it along, and following.

MR. EBERSOLE: It is the coming out that's the problem?

MR. FOREMAN: Right. We have --

MR. EBERSOLE: Drive comes out but the rod stays up? It is still a cruciform rod, isn't it?

MR. FOREMAN: Yes.

MR. EBERSOLE: It hangs up on the core and sometime later it falls out at high speed and reactive transient results?

MR. FOREMAN: We take all of that into account in the design, just like we do for the present plans.

MR. MICHELSON: In your present plans you have velocity limitors?

MR. FOREMAN: That's correct.

MR. MICHELSON: How were you able to eliminate it here? What did you do to compensate?

MR. FOREMAN: We were able to design it out. I can't answer specifically how we are able to do that, but we were able to look at the temperatures, peak temperatures that you get to with that kind of an accident and to design it away. I can't tell you specifically today how that happened.

MR. EBERSOLE: Does this have to have a water cool seal at the vessel interface?

MR. FOREMAN: No.

MR. MICHELSON: This doesn't have to have a scram discharge volume?

MR. FOREMAN: No, it does not have a scram discharge volume. That's one of the things that's been eliminated.

MR. MICHELSON: Is this similar to the design that the people in Sweden use?

MR. FOREMAN: Yes.

One of the differences between this and the design that Sweden uses is the number of accumlators that we have for each rod. I guess I should say the number of rods assigned to each accumulator. We are choosing right now at this time to assign two rods to each accumulator. The

Swedes choose to assign more than that.

MR. EBERSOLE: This is a hydraulic motor?

MR. FOREMAN: This is an electric motor.

MR. EBERSOLE: What is that scram inlet that I see?

MR. FOREMAN: There is a hydraulic scram and the electric motor then -- once the rod scrams hydraulically -- that's why we have accumulators -- we have a pump of charges up the accumulators, uses that hydraulic scram as a primary scram mechanism. The back up scram is the electric motor, and it follows along say two minutes later so that in the case where you weren't able to scram by way of hydraulics you were able to scram two minutes later with that electric drive.

MR. EBERSOLE: As the thing is driven up electrically, does it face again the closed volume control rod drive accumulator?

MR. FOREMAN: There is no accumulator at the other end. It goes into the vessel.

MR. EBERSOLE: Okay.

MR. FOREMAN: So you have been able to eliminate the scram discharge volume with this and that's one reason you are able to reduce the exposure to the operator.

MR. EBERSOLE: Where does the discharge go in the normal hydraulic scram, discharge water go?

MR. FOREMAN: Into the vessel.

MR. EBERSOLE: You mean you have an --

MR. FOREMAN: That's correct.

MR. EBERSOLE: You no longer have an requirement for continued floor for seal cooling at the vessel rod interface?

MR. FOREMAN: That's correct.

MR. EBERSOLE: Did you improve the seal design or -- 550 at that point and it has got to live. Right now you have got to have water running through it all the time.

MR. FOREMAN: I can't answer the question.

MR. EBERSOLE: This is fine structure maybe we shouldn't be into.

(Slide 21 shown.)

MR. FOREMAN: You have to understand also I feel a little bit naked here because I'm without our joint customers and I'll answer the question if I can, but --

GE is -- with the joint partners isn't doing all the design on this, or all of the calculations and that sort of thing. A lot them are be done by Hitachi, Toshiba.

One area that probably would happen no matter whether we were working on ABWR or trying to come up with better fuel, because in addition to building reactors and nuclear islands and that sort of thing, we do sell fuel.

But one of the things that we are designing is a reactivity difference between the upper and lower fuel. We are going with the control sale core concept by having low reactivity fuel assembly around the control rod for operation, which you already have seen on domestic plants.

Increased thermal margin caused by adoption of fine motion control rod drives. Core flow control with wide flow window -- caused by the high enriched fuel design. And I've got another slide of differences.

MR. MICHELSON: I guess you felt that you have the problem solved on how to control the orientation of rods and so forth during fabrication so you don't get them upside down. You have non-uniform loading within the individual rods.

Do you think that would be a non-problem or have you analyzed what happens if I get a rod in backwards and put it into the core? Hotter rod?

MR. FOREMAN: I'm certain that's all done.

MR. MICHELSON: I just want to make sure I understood. Thank you.

(Slide 22 shown.)

MR. FOREMAN: Here is a side by side comparison of a typical BWR and ABWR. You can see that we have jet pumps rather than recirculation systems so these nozzles are not there. There is some difference in the coupling. We still have a spray. I can't tell you whether that will be an overhead spray or perimeter spray. That's one of the things that is still under discussion.

By eliminating the control rod velocity limitor one of the things we have been able to do is shrink the total vessel height so the vessel height will be smaller.

Do you have any questions?

MR. EBERSOLE: I would have thought diameter would have been larger, but it isn't?

MR. FOREMAN: I believe the diameter is larger.

MR. MICHELSON: I think you want to watch the -- from the voiding viewpoint when you inject that cold water, when you put that nice little loop at top on the sparger as well. It is just to complicate the problem, I think.

MR. FOREMAN: Certainly that loop up there is designed to be flexible.

MR. MICHELSON: It is also going to be voided as soon as you depressurize the reactor and before cold spray water gets to it.

MR. FOREMAN: We apply all our normal criteria in the design of this that we would into any plant.

MR. EBERSOLE: -- where you built all the substructure under the rods way back in Humbolt Bay?

MR. FOREMAN: We have shootout steel in the present plants. On ABWR the question is whether there will be shootout steel under investigation. I don't think --

MR. EBERSOLE: I never heard that term before. Is that the structure under the rods?

MR. FOREMAN: That's correct.

MR. EBERSOLE: I never heard of that.

MR. FOREMAN: That's maybe a phrase that's used in the field. But you are talking about the girders underneath that prevent the projection more than like six inches.

MR. EBERSOLE: Right.

MR. FOREMAN: I think it is important for you to understand as we went through this design -- I'll go ahead and put up the next slide --

(Slide 23 shown.)

MR. FOREMAN: One of the areas of concern became cost. When we started out with the AET study, and in phase two, we had some very high objectives for ABWR. And the cost just went completely out of sight. And there was -- I don't think there would be any utility that would have bought it at the price.

So in the process of optomizing the design in phase two one of the things we have done is try to cut down that cost so that our objective now becomes to have a higher output plant with greater safety for the same cost.

And one of the ways you do that is by simplifying the design. In the area of ECCS, which is one of the largest parts of the cost of a plant, the question became how many divisions do you want to have? At one time we had design to an N minus two capability.

We designed throughout the plant for double failure. It was just extremely expensive. Our design basis now is N minus one, a risk less than BWR/5.

The three or four division questions are still up in the air. It has not been decided what we will have. Clearly, if you have three divisions -- well, you could have --

Go ahead.

MR. EBERSOLE: What do you mean by N?

MR. FOREMAN: That's the number of failures. N minus one is a single failure, which means that you have four systems and you have single failure, and you have three. If you have four systems and you have two failures, then you still have two. So you can have double failure and still have 100 percent core cooling capability.

MR. EBERSOLE: I wanted to ask if you discriminate between the systems which are on line in constant challenge versus those on standby when you talk about N?

MR. FOREMAN: Yes.

MR. EBERSOLE: So you need more systems if they are always on line?

MR. FOREMAN: I'm sorry?

MR. EBERSOLE: You need more systems if you are always on line because you know they are going to fail?

MR. FOREMAN: No. With N minus two you are able to have one out of service and one failure.

Maybe I said that earlier and misled you.

MR. EBERSOLE: Then what do you have to sustain the operation? One left?

MR. FOREMAN: If you have three divisions you would have one left. Still have two with four divisions.

MR. EBERSOLE: What do you adopt? What is the system?

MR. FOREMAN: The system we have is going to have single failure capability which is either out of service or a failure.

MR. EBERSOLE: Could you express it to me that in the number of systems in a systems which is on duty all the time, not one that's on standby, but on duty.

MR. FOREMAN: Let me try to answer your question in my words.

With ECCS systems you would always want to be able to provide 100 percent.

MR. EBERSOLE: That's the system that you don't use except once in a coon's age. I'm talking about the system that's always on line, like service water. That's a different cat.

MR. FOREMAN: That will be designed to the same single failure criterion that we have now and --

MR. EBERSOLE: What does that mean? How many systems do you have of service water, as a case in point? Don't tell me two.

MR. QUIRK: We are kind of approaching it from the other side of the equation. Instead of defining N, we define the one or the two. If N is the required necessary systems in the plant to safely shut down a cooldown, one is we take away one system -- would result in the loss of a system.

MR. EBERSOLE: I have then as a minimum two systems even though they -- one or the other is on duty all the time? That means if one fails and the other has to go into a transient to meet the new demand it cannot suffer the random single failure demand on challenge?

It is contradictory to the three channel system which says when I set up a transient which I have to meet with residual systems -- mitigating systems?

I think that's contradictory to the current interpretation and to the single failure criterion at large which says that -- I have to complete transients with single failure privileges.

And certainly always if first failure is due to a consequence of an accident being mitigated I must then have the privilege of redundancy after loss of that one.

MR. FOREMAN: Without getting into the fine details of what the system looks like, the top level criterion for the design of ECCS and the wetwell drywell cooling system was that we would ha either three or four divisions, so that question is the one we are addressing right now and -- so when we are going through that design -- the system is going to be -- going to look like it does otherwise, except it is going to have more divisions to meet the demands, to meet the demands that you would place on a two -- the capacity could be less for each division.

MR. EBERSOLE: Staff document about the interpretation and application of single failure criterion, are you going to put out a document that reflects your agreement or disagreement with that? 1977 document.

MR. FOREMAN: Can you address that, Joe?

MR. QUIRK: Tell me more about the document, title of it.

MR. EBERSOLE: It is dated August 17, 1977. It is to the commissions from Lee Gossett(phonetic spelling) and it extends and amplifies the single failure application. I'll give you a copy of this.

MR. QUIRK: You would like our comments?

MR. EBERSOLE: Stick to this or depart from it?

MR. QUIRK: With regard to the ABWR or with regard to BWR/6?

MR. EBERSOLE: Both, if you will offer it.

MR. FOREMAN: It has to be a GE type response.

MR. QUIRK: I would be happy to respond to that.

MR. MICHELSON: Before I get totally lost, when you say you might have a four divisional system, as an example, do you mean that the support systems for each of those divisions is also separate?

MR. FOREMAN: Yes.

MR. MICHELSON: So that means four divisions of water, four divisions of power, et cetera?

MR. FOREMAN: Yes.

MR. MICHELSON: There will be three completely separate trains all the way back?

MR. FOREMAN: Yes.

MR. MICHELSON: The question is on the lower pump, recirculation pump, will you be able to replace seal packages without draining the reactor?

MR. FOREMAN: I can't answer that question.

MR. MICHELSON: I would like a -- I'm sure you have given a lot of thought to it.

MR. FOREMAN: We have plants in operation in Europe.

MR. MICHELSON: Isn't it obvious that you have to drain the reactor? In other words you have to take all the fuel out in order to replace the fuel package?

MR. FOREMAN: That's not obvious to me.

MR. MICHELSON: There is a hole in the bottom of the vessel when you take the package out for -- that's a little bit different situation. This is not the same. So that was my question, how do you replace seal packages? Can you do it without draining the vessel?

MR. FOREMAN: I'm sorry but I can't answer that question.

MR. MICHELSON: What kind of radiation levels does one have to contend with? The Japanese were very worried about radiation -- unless you unload the core.

MR. FOREMAN: I can assure you all of those things are taken into consideration and radiation exposure is very important.

MR. MICHELSON: From experience I think you have to replace seal packages on your present pumps about every three years or thereabouts is my understanding.

MR. FOREMAN: You are talking -- this is a different kind of design. The pump is internal to the vessel.

MR. MICHELSON: But seals are not --

MR. FOREMAN: It has water cooling there. That's included as part of the design.

MR. MICHELSON: The seal eventually will degrade and you will have to replace them about every three years, more or less.

MR. FOREMAN: I can't answer your question. I could even conceive for seal replacement maybe you could take it off the top of the vessel. I don't know.

MR. MICHELSON: I'm sure there is some good answer.

MR. OKRENT: I think we are going to have to move on and not spend more than about ten more minutes on this presentation.

MR. FOREMAN: To answer your question about --

(Slide 24 shown.)

MR. FOREMAN: -- what would you do with four divisions, this is a conceptual design of what four divisions would look like. I don't intend to go into any great detail on this slide at all. You will also notice on this particular slide we had conical containment in our design.

MR. EBERSOLE: What is the international choice of distribution over three versus four versus two? What do Swedes use? Four? Do you know?

MR. FOREMAN: I don't know personally. I would have to go back and look that up.

There are -- in Europe there are some countries that believe in full N minus two capability of being able to take a double failure and one failure and one out of service, et cetera, and their plants are fully designed for that, and when we started this program that was one of our goals.

Now, so you have got -- you have to ask yourself the question: I'm making all this big investment to come up with a new product that's very risky. If I now go to three divisions from four maybe I've eliminated a country that I can sell it in.

MR. EBERSOLE: For instance, core spray pump is never needed, I hope. But the service water pump and component cooling are used all the time.

MR. OKRENT: We are going to have to drop that subject, Jesse.

MR. FOREMAN: The last area I want to show you a deference on was the containment.

(Slide 25 shown.)

MR. FOREMAN: The goal was a working space equal to what is called in Japan an improved Mark I and Mark II containment with an individual hatch to the lower drywell.

One of the goals was to include countermeasures for dynamic loads. We felt it was easily mitigated with reinforced concrete -- directly installed from the containment restraint structure wall.

There are continuing studies going on. Prior to this week we had a conical containment. I was in contact with the project team that's in Japan this week and able to find out that they have gone to -- they have made a decision to go to cylindrical type of containment, so I've changed that particular viewgraph from what you would see in that September 26th letter.

One of the areas that's still under discussion is whether this design will have horizontal rather than vertical vents.

MR. MICHELSON: Is it correct to assume that the refueling is from outside the containment then? It appears that's the design.

MR. FOREMAN: It appears that's the design.

MR. MICHELSON: So now you have to go back to the old questions -- drop things into the core and that sort of thing without containment?

MR. FOREMAN: The design of that upper area you have to take into account turbine missiles -- well, turbine missiles and other things, so you have to decide which pools you are going to fill, which ones you are not going to fill, how you will reinforce that area.

MR. MICHELSON: One of the classical problems is lifting all those internal and dropping them in the process or dropping --

MR. FOREMAN: That's right.

MR. MICHELSON: It looks like they have gone back full circle on some of these questions with this modification of Mark III.

MR. FOREMAN: I'm pretty confident that is going to be a cylindrical containment and certainly that's what GE feels.

If you were to see a containment drawing -- that's why I've taken the judgment that I'll change this figure, but it is always a possibility that we could go back to conical containment. Design is not done. Of course we might go to horizontal vents also.

MR. MICHELSON: Refueled from outside the containment also.

MR. EBERSOLE: Doesn't that design I see there throw back into the picture the potential of suppression bypass as a result of pipe failure, which I thought you had gotten rid of in the Mark III?

MR. FOREMAN: Yes.

MR. EBERSOLE: Don't the Germans use double wall pipe to go preclude that type of thing?

MR. FOREMAN: I can't answer that question. I understand, but I don't know what the Germans do. I didn't know that the Germans used double wall pipe.

MR. MICHELSON: I guess you don't have a weir wall anymore?

MR. FOREMAN: No.

MR. MICHELSON: You are now going to vent every one of those --

MR. EBERSOLE: You know the Brookhaven study that shows how long you last on a suppression bypass?

MR. FOREMAN: No. I'm not aware of it.

MR. EBERSOLE: It is worth looking at.

MR. QUIRK: I missed the opening remarks. Did you talk about a little -- about the Japan influence on some of these features and -

MR. FOREMAN: Yes, I did.

MR. QUIRK: Thank you.

MR. FOREMAN: I've given you another view of what the overall building looks like --

(Slide 26 shown.)

MR. FOREMAN: -- with the cylindrical containment.

MR. MICHELSON: It is really a modified Mark II now, isn't it, more than a Mark III --

MR. FOREMAN: You could call it a modified Mark III also or you could call it a modified Mark I.

MR. MICHELSON: But with the vertical as shown here it looks like it is a Mark II.

MR. FOREMAN: That's correct.

MR. OKRENT: We could call it an ABWR.

MR. MICHELSON: I was just trying to orient my thinking.

MR. EBERSOLE: Are you going to inert this containment?

MR. FOREMAN: Yes.

The current status on t' project is that the design studies are going to continue to the summer of 1985, late summer of 1985, like August or September.

Technical aspects currently under review, the large kind of technical aspects currently under review increasing thermal power capacity from 3845MWT to 3926MWT thermal, Adapting to 52 inch turbine generator, four versus three ECCS divisions and containment design, which I've already talked about.

We also have ongoing studies throughout the plant for optomization improvement and that sort of thing.

MR. OKRENT: Thank you. I think we better terminate this part of the discussion. We will take a ten-minute break and then go into the agenda for today. Reconvene in ten minutes.

(Recess taken.)

MR. OKRENT: The meeting will reconvene.

Let me ask representatives from General Electric, of the various topics which are vaguely identified on the agenda with regard to PRA, which of these are clearly not proprietary in your opinion and which do you think can only be discussed appropriately in a proprietary fashion?

MR. HOLTZCLAW: Can I show you a chart that we had in our modified agenda?

MR. QUIRK: Put it in the overhead.

MR. HOLTZCLAW: We configured our PRA presentations a little bit differently in order to try and get a flow in the presentations and still be responsive to all the items on the ACRS agenda, and we have got some options that I think we can do now as far as presentations go.

What we would like to do is get into the bulk of the PRA proper and the front end portion -- I can't remember the item on the agenda, but it had some specific issue. I think that can be best handled, though, if we can go through just a very brief overview that covers some of the scope and lays some ground work and then go into the presentation core damage probability and uncertainty analysis and the so-called front end of the PRA and the back end of PRA on core melt phenomenon, and go back and pick up some of the specific issues that we had identified in response to your specific areas of uncertainty, because we will be covering a few of those in both of the basis PRA presentations.

And then I can pick up the specifics following that presentation -- those two presentations.

MR. OKRENT: Would you repeat again, then, what it is you are proposing.

MR. HOLTZCLAW: We would like to give an introduction that covers the scope and some of the background and overview, then present -- that's a non-proprietary presentation -- again into the core damage probability and phenomenon and back end of the analysis.

These both contain proprietary information. And then assessment of specific issues, we follow the basic PRA presentations.

MR. MICHELSON: Will there be any discussions of external events then?

MR. HOLTZCLAW: Yes. We could continue on with -- in that fashion, continue on with the rest of the agenda as far as we can get.

MR. OKRENT: As far as time permits.

MR. HOLTZCLAW: Right.

MR. OKRENT: Well, it is all right with me to try it. Why don't we start that way?

When we reach the first subject that is proprietary, would you let me know whether it can be in your mind divided into two portions, which are proprietary and non-proprietary, and if not, just what things are proprietary.

I see many pages in the GE documentation stamped proprietary that are academic in nature. I've seen the same pages in lots of other non-proprietary reports. So I need to get a better definition in my own mind.

First, I don't want to at the moment violate accidentally the agreement of the NRC in proprietary things by not knowing really what is proprietary and what is not.

Why don't we begin and see where we get.

MR. HOLTZCLAW: I would like to give some of the overview of the probabilistic risk assessment we perform for GESSAR II a little bit on the scope of the program and then turn over the presentation to Mr. Larry Frederick, who will present information on core damage probability, which will also include some of the discussions of input into the fault tree and event trees, and then Dr. Deborah Hankins will make a presentation on the back end of the PRA on the subsequent consequence analysis.

I'll be talking here a little bit on the objective. I think we have seen some base results already so I'll be fairly brief in that discussion.

I'll talk a little bit about the scope, the methodology that we utilized, specifically the sites selection we used in the doing of PRA on a standard plant. And give a kind of road map on what the major tasks are that will lead into the two following presentations.

(Slide 27 shown.)

(Slide 28 shown.)

MR. HOLTZCLAW: Again, by way of overview, the objectives of the study were to quantify the safety of standard plant design, provide a comprehensive assessment of public risk resulting from plant operation.

We included in the analysis an evaluation of core damage frequency and the off-site consequences, identified the major contributors to risk and our bottom line results that I think we have talked to this group before in the last day at least, we believe that the likelihood of an accident progressing to core damage is well below the values obtained in the Wash-1400 studies. And in a subsequent presentation we had plants in comparison to well below that value. And the risk as well is below the Wash-1400 results in the interim safety goals.

(Slide 29 shown.)

(Slide 30 shown.)

MR. HOLTZCLAW: Just by way of orientation, the scope was to perform the analysis of the BWR/6 Mark III standard plant design, so-called standard 238 plant.

As we have pointed out, we have included modifications made in response to new reg zero seven three seven Post-TMI modifications as well as modifications dictated by the ATWS rule.

We incorporated in our analysis the so-called ATWS alternate three A. We also made -- incorporated the modifications that were made in response to the Brown's

Ferry partial scram.

Reference Mark III containment, free standing steel containment building, concrete shield building, drywell, primary containment and secondary containment.

There was some decision that had to be made on our part in what we would utilize for reactor site to perform this study and perform the consequence analysis. We ended up deciding on using the reactor safety study site six as far as the site parameters meteorology and demography, so-called Atlantic coastal site. It is in the population of 81 point four million people in the five hundred mile radius.

. In comparison to the RSS sites it was probably most typical of so-called average sites. It was also a site that we did have some good information as far as a specific site in that we had been doing some work on other areas and had picked out a site that was within that region, that is, the New Jersey, Maryland PJM grid experience that we made use of, and this was essentially an input assumption.

As a grid within that site six, and this is the information with regards to its key parameters as far as loss of off-site power probability per year and the recovery capability that's been demonstrated within that site within the first 12 hours. This was factored in as an assumption in the study.

We pointed out as far as the applications of this study to specific site location that the applicant would have to perform or re-perform a consequence analysis for the specific site that reactor would be sited in.

(Slide 31 shown.)

(Slide 32 shown.)

MR. HOLTZCLAW: In terms of the methodology it is fairly consistent with other PRA's that have been done, utilizes base Wash-1400 methodology, utilization of fault tree and event trees.

We did do some specific work in assessing realistic success criterias, and you will be hearing about this in Mr. Frederick's presentation, that is, identifying systems required to reach successful termination of an accident sequence.

We also factored in one of the key results of the Post-TMI experience as far as the BWR owners are involved, that is, the use of the BWR owner's group emergency procedure guidelines. That was a big aid in determining human error response to specific sequences.

One other significant departure from say the Wash-1400 reactor safety studies was the utilization of what we believe to be more realistic product modeling and this factors in the recent or more recent pool scrubbing information from GE test programs. You will also be hearing about this in the subsequent presentation.

Next --

(Slide 33 shown.)

MR. HOLTZCLAW: We have a chart here that I think we showed a year ago and we kind of gave it very much of an overview of this presentation and it is some of the comparisons with the Wash-1400 methods.

There has been a change in the frequency of initiating events. Mr. Frederick will be covering that with a chart in his presentation.

We believe that the fault trees and event trees like many other PRA's that have been done since the -- pointed out we think we have more realistic success criteria. One of these was the credit for feedwater condensate pumps not included in the RSS study.

With regards to ATWS sequences, there was an additional ATWS sequence including -- also factored into the RSS update that was performed by the staff a few years back.

We utilized more release categories in doing the fission product retention and release analyses, and we have been using codes that have been updated since RSS. There has been just a general improvement in that area.

We have also utilized codes that have been developed and utilized internally to General Electric specifically in looking into the success criteria and the thermal hydraulics analyses where the BWR being somewhat indigenous to other reactor designs probably requires a slightly different analysis of thermal hydraulic effects than PWR's do, so there is some divergence with the standard methods available -- since we have to deal with it in the course of our normal design processes.

(Slide 34 shown.)

MR. HOLTZCLAW: This is a very overview-ish view of the results. We will be looking at this in more detail in subsequent presentations. It characterizes the core melt frequency for our study and came out approximately five times ten to the minus sixth.

MR. HATCH: Does that include UPPS?

MR. HOLTZCLAW: No. I'm sorry. I probably didn't preface this. This is really the internal events portion of the PRA. We are going to be having some subsequent presentations on the external events.

We believe our results have attributed to a number of features of the BWR. Consistent with other recent studies it is probably a more accurate assessment on the time to core damage.

I think we have highlighted the BWR capability for water delivery in the multiple systems and I think that was obviously verified in this study. Also hand in hand with that would be capability is the utilization of more realistic success criteria.

And the bottom -- loss of containment integrity I think is another importance aspect because in some of the sequences it obviously provided the operators with longer response times and had an impact on human reliability.

(Slide 35 shown.)

MR. HOLTZCLAW: This is a chart that I think has been shown in a couple other presentations already in the last two days, chimney charts that shows the types of sequences and what their relative contributions were in core damage.

Again, showing the significance of loss of off-site power initiator and its dominance as far as the core damage frequency goes. We again will be focusing on this chart when we do a more thorough discussion of the results, and it is going to figure also in our discussion of design modications, because it is a primary input to that study.

MR. CAMP: For this discussion is there any difference between core damage and core melt?

MR. HOLTZCLAW: No. I'm sorry.

(Slide 36 shown.)

MR. HOLTZCLAW: I want to lay out one fairly overview chart here on what you will be seeing in the next two presentations.

MR. MICHELSON: Before you do that, can you tell me just briefly the reason why an inadvertent opening of safety relief valve is a much higher frequency of core damage than a small break loca, which is kind of the size that we are dealing with?

MR. HOLTZCLAW: What was the -- maybe Larry can answer this.

MR. FREDERICK: Largely in the initiating frequency. IORV, particularly in models originally in the PRA, has a higher frequency of occurrence than a pipe break, small break loca, but it is a much higher --

MR. MICHELSON: Because of frequency of occurrence?

MR. FREDERICK: That's correct.

MR. EBERSOLE: But because it is suppressed, it goes under the pool, what is the terminal event? How does it proceed to cause so much trouble?

MR. FREDERICK: It doesn't particularly cause a lot of trouble but there is not an automatic scram with an IORV. The operator must manually scram the reactor. So there is an operator error in the analysis.

Does that answer your question?

MR. EBERSOLE: Yes. I'll turn around on that a while.

MR. HATCH: One further question, how is the loss of heat removal different from the station blackout where you would expect also to have a loss of heat removal?

MR. HOLTZCLAW: As far as this chart goes? I'm sorry? Go ahead and answer it.

MS. HANKINS: These are divided up in terms of initiating event. This is initiated by a transient or loss of heat removal. In other words you have sufficient core cooling -- the station blackout you do not have core cooling.

In my presentation I'll be describing how we divided events up in terms of which goes first, the core or the containment. Heat removal sequence of the containment fails before you lose core cooling.

MR. HOLTZCLAW: TQUV for loss of off-site power initiators or that class of events and the other over here as a TW.

MR. EBERSOLE: In that inadvertent RSV opening the discharge has to the suppression pool and the operator, you say, the operator failure to respond is the problem.

So aren't there inumerable indications that he has got -- that he has got to do something but not necessarily too fast?

MR. HOLTZCLAW: The best thing to do would be wait until we go in to the individual.

MR. QUIRK: The answer is to his concern that in and of itself inadvertent open safety relief valve is not a real significant event without dire consequences. It is only if that's the first of many subsequent failures, which you assume all the way to core melt.

MR. EBERSOLE: So that's a composite event? I would rather not think of SRV as being particularly important.

MR. QUIRK: I don't like to give the impression to people that the blackout is a dominant event to core damage. It isn't. We have systems to mitigate it, but it is the first step. It is part of communication problem.

MR. MICHELSON: I'm a little surprised that a scram itself doesn't lead to core damage when you start compounding all of these, and I thought it would even have a little bit of length of the bar -- because of a scram you got into trouble. But it is just -- the first step would be a scram. The second step is stuck open relieve the third step operator fails to respond.

Isn't there a finite probability of that scenario compared with your small break loca, for instance?

MS. HANKINS: Manual shutdown is one of the initiating events. Mr. Frederick in his presentation will be describing all the initiating events.

MR. MICHELSON: Why doesn't the scenario I propose appear as a bar because I think it has got probably higher core melt probability maybe than a small break? That's why we are trying to eliminate scrams.

MR. FREDERICK: All of these initiating events are initiated by scrams with the exception of the manual shutdown. The turbine trip is a scram. MSIV closure events, isolation event, is a scram. All of these are scrams. And the initiating event frequency based on operating field operating data as collected based on --

MR. EBERSOLE: I thought you said the --

MR. FREDERICK: RSV is the exception to that but the turbine trip MSIV closure and loss of feedwater are all scrams.

MR. EBERSOLE: If I look at that on the probability particular basis certainly the inadvertent opening SRV must be far more probable than any of the rest of them?

MR. FREDERICK: No, it is not. Turbine trip is by far --

MR. EBERSOLE: It is not even up there.

MR. FREDERICK: Well, this effects core damage frequency.

MR. EBERSOLE: I'm looking --

MR. FREDERICK: We are getting into the presentations I'll be making in just a minute.

(Slide 37 shown.)

MR. HOLTZCLAW: This is kind of a general road map of the next two presentations.

Mr. Frederick will be talking about frequency of core damage and Dr. Hankins will be covering release and the consequence portion of the PRA. That's broken up even further, I guess, into the steps that we went through in the next chart. It shows the various blocks and --

(Slide 38 shown.)

MR. HOLTZCLAW: -- where we utilize the standard plant configuration date, went through the process of identifying and quantifying the accident initiators and then factored in the success criteria in order to put together the accident event trees.

Then identify, quantified, classified sequences and constructed containment event trees and that interacts with the fission product transport analysis utilized as the core damage and containment analysis as input and then does the release portion of the analyses where consolidated into release categories, define the frequency of fission product release. Then out here in the last box did the consequence analysis and the evaluation of plant risk.

What I would like to do now is turn the presentation over to Mr. Frederick to cover the evaluations of core damage.

I think in terms of what Dr. Okrent was asking, I don't recall how your slides go, Larry, as far as do we have the bulk of the proprietary information in any one location?

MR. FREDERICK: Very close to the beginning we will have success criteria which is proprietary. So we can close off.

MR. QUIRK: I recommend that we close the meeting at this time, Doctor Okrent.

MR. OKRENT: Well, before we get into that, let me ask the staff, when will we receive all of the consultant reports in, I guess what you would call them, final form with regard to the GESSAR II PRA or when will we receive some of them in final form and others --

Have you got a schedule?

MR. SCALETTI: Doctor, much of the information from Brookhaven you have in draft form. Many of the reports are not finalized and the schedule for finalizing these probably would be from two to three weeks to a couple of months and so you won't have them in the near term. We have turned over all the documents that we have relating to -- with the exception, I guess --

Do we have external events documents?

MR. RUBIN: I don't know.

MR. SCALETTI: You have what we have so far.

MR. OKRENT: Since I haven't had the --

MR. SCALETTI: Excuse me. I'm sorry.

MR. OKRENT: Since I haven't yet had the benefit -- I better put quotes on that -- of seeing all of these documents, I can't tell, for example, whether a draft document is a relatively complete document or it is a document missing a front and a back, which I sometimes have also seen coming from the staff with six months delay to the final document on other reviews.

MR. SCALETTI: I believe the internal events that you have is a relatively complete document from Brookhaven. Again, with regard to whether these -- I don't know if you are asking or whether you are going to ask it not -- these documents do not identify proprietary information.

Now, one of the -- we have not gone through them yet to identify this information nor has, I guess, General Electric. Maybe one or two of the documents they have, but -- the reason when we sent this stuff to the committee on October 5, we put a note on the package that indicated that some of this information may contain proprietary information. Some of these documents -- it has not been identified as yet.

MR. OKRENT: I see. Now, I'm just trying to understand now. General Electric mentioned a moment ago success criteria are proprietary. I guess I'm -- give me an example of one -- not the answer, but a kind of success criterion that's proprietary.

MR. FREDERICK: The success criteria were based on and derived from transient analysis using General Electric codes which were developed by General Electric and are proprietary. The determination of whether one pump, given a certain set of conditions, whether one pump will adequately cool the core is determined by a transient analysis using General Electric proprietary codes.

Does that answer the question?

MR. OKRENT: I guess I understand that answer. What are some of the other kinds of things then that are considered to be proprietary specifically?

MR. QUIRK: In general, there are two broad reasons why some information is proprietary.

The first reason is safeguards related information and one can view fault trees, for example, as a very specific quantification of the vulnerabilities and capabilities of an actual plant. And some utility customers and GE maintain that for that reason this information is safeguards related and as such should be withheld.

Another --

MR. OKRENT: Let's not leave that point, because it is a non-trivial point. It is my impression that the staff has not taken the position that fault trees and event trees in PRA's for specific plants should be kept in a -- should be handled in a security fashion, as relating to that kind of issue, because I was seeing PRA after PRA with that material in it.

Can the staff advise me of what their position is and how they arrived at it? There may be a technical position and there may be a legal position here. I'm not sure whether the two are the same.

MR. THOMAS: I'm not sure we can add an awful lot to it. You are correct in your observations, at least as far as our practice has been we have not considered vulnerabilities to be information that should be withheld. I'm not sure we have Larry focused on it, but in our practice we haven't.

MR. OKRENT: As a matter of fact, you published Wash-1400?

MR. THOMAS: Yes.

MR. EBERSOLE: I can't see how the numbers could be proprietary. I can see the method by which you achieved the numbers.

MR. QUIRK: Let me address that.

In our PRA from day one we have made available the bottom line numbers, the core damage probability numbers and the events that we have shown, and the consequence analysis of fatalities and such. And that information has been made available because we think it is an important conclusion of the study and the public has a right to know

And other information has been been made available [illegible]

What ın essence we are withholding are the fault trees for what I will regard as safeguards related. Event trees are success criteria, which I think are commercially related. And I didn't get into developing that thought but I was saying earlier there are two reasons, safeguards and the second was going to be commercial.

And what I mean by commercial, typically a lot of people have done PRA's and they used standard methodology and standard assumptions, some of which are not representative of the BWR systems, and they get what I'll refer to as scary results. Well, in our PRA we use standard methodology but we took into account actual BWR capabilities that maybe others that aren't as familiar with the BWR may not.

And we put in some of our research results and these considerations led to more realistic results. And these results were developed with GE resources and I think are commercially available.

MR. EBERSOLE: Do you mean the numbers are available?

MR. QUIRK: The methodology.

MR. EBERSOLE: By which you got the numbers?

MR. QUIRK: Yes.

MR. EBERSOLE: But not the numbers proper.

MR. QUIRK: It depends on how far down you go. The overall -- when you get into each specific sequence I think that's where we say from this point on --

MR. EBERSOLE: The way I hear it, you have developed certain impediments to progression of an accident, maybe sequential things that you do that you would rather not divulge at large?

MR. QUIRK: Yes. I can give you a couple of examples.

In our PRA work we factor in credit for the fuel channels. The analyses don't. You know, in the PWR regime the flow can kind of divert and go around and this results in a higher hydrogen production. So when we take into account the fuel channels, it causes blockage and lower hydrogen production and can change the answer.

Another example of factoring in specific BWR, which resulted in an opposite effect, a conservative effect that hadn't been factored in before, is the guide tubes. Failure of the guide tubes would result in earlier vessel melt through which would give us worse results earlier. And so there are some of these examples with our detailed system knowledge and expertise we sharpen the pencil and it can alter the answer considerably.

MR. OKRENT: I must confess the example about failure of the guide tubes seems to me to have been a topic that was discussed in open meetings when one talked about different ways in which a core melt might progress and vessel failure might go, so I'm hard put to see the basic ideas being proprietary.

MR. QUIRK: It is not the basic idea. We just talked about it. But what I'm trying to say is that we have taken the standard methodology, for example, March code, and we have modified it to reflect unique specific BWR capabilities and have altered that methodology.

MR. OKRENT: Well, I guess the staff has accepted in the past that computer codes developed in part or entirety by vendors can be kept proprietary even if sometimes they represent a modification of something previously developed at a national lab for -- sort of for free.

But we are trying here to understand the rationale for keeping different parts, and in fact the entire PRA, pretty much proprietary and at the moment I've heard this matter on success paths, which I'll grant is something that could come out of an individual calculation -- I must confess, compared to disadvantages I can see in keeping this PRA proprietary out, I'm not convinced that there is a large commensurate commercial disadvantage that way.

Could you help me understand what other kinds of things you feel are really proprietary, and leaving aside this -- what I'll call security question -- which is a very troublesome one, no doubt. But nevertheless the NRC has followed certain policy in that regard and --

MR. FREDERICK: Dr. Okrent, we feel at General Electric -- we feel GE is in the best position to do probabilistic risk assessments on boil water reactors. There are many other companies that would like to do them. They have a right to do that. They would like very much to be able to have the entire -- the entire analysis that we have done. We presume they would like to have this and to be able to use it or quote from it. We can't and don't want to prevent them from doing probabilistic risk assessments, but we don't have to make the results of our analysis available to them. It is a commercial consideration.

MR. OKRENT: Well, your agency, you are well aware, the proposed severe accident safety policy of the NRC hinges much of its review on the PRA, and if it is essentially all proprietary, this pretty much means that the bulk of the public, including those who could interpret this and critique it and review it with sophistication, but the bulk of the public besides, have no access to what is a vital part of the way in which a judgment is arrived at concerning the adequacy of the safety.

And it seems like there is, you know, there is a mixture of different kinds of questions here and not easy to balance one against the other. If there were a question of spending 50 dollars here and 48 dollars here and that was it, we would know what to do. But that's why I'm trying to see really where the true proprietary features or the more important proprietary features, or whatever it is, lie and where some may be of lesser importance.

MR. FREDERICK: Mr. Okrent, I think possibly the whole is greater than the sum of the parts in this case. The entire analysis altogether is of value.

I've been in this division of General Electric for 15 years and during that entire time I've seen an expenditure of resources by General Electric in the collection of data, in the performance of reliability analysis and models of plant safety functions as well the other functions, and there is considerable investment there.

MR. QUIRK: Dr. Okrent, this may be presumption on my part, but in my opinion, when you look at the -- not all the proprietary information -- not all the PRA is proprietary. We have recently met with the NRC staff and have developed a version which is just pulling out all the non-proprietary information in the PRA that was formerly proprietary.

Let me explain.

If the significant part of a document is proprietary, and intermingled rather in the past, rather than issue two sets of documents our habit was to stamp it all proprietary and issue it accordingly. In today's environment that's not desirable anymore and we have taken that document and extracted the parts of it that are not proprietary and we are ready to issue that. So that information will soon be available to the public.

A summary form is already available to the public that they can look at and get the bottom line results. My point is this, with that version and with the staff's fairly detailed version, I think that information would overwhelm the needs of the public to be included, you know, in how we arrived at what we did and what the basis was.

I think it is more than ample, more than sufficient. As I said, that may be presumption on my part but somewhere along the line you have got to drop the line and protect a lot of work that we have done at our expense and I think that we haven't made that determination lightly and I think the available information --

MR. MICHELSON: I would like to comment on the security aspect.

I appreciate fully the need for certain commercial protection but I am really at a loss in the security area keeping in mind that the plants that have vulnerability to sabotage are those in existence, and not on paper, and the paper plant is long way down the road.

We are not protecting presently the existing plants which could have a potential sabotage vulnerability. I can go to a number of PRA's and find out all I need to know.

They haven't been and therefore the horse is already out of the barn. And so why and how you could argue that protecting this information from a security viewpoint for some future way down the road plant is beyond my comprehension.

Could you help me a little by how you can really rationalize there is a security matter involved here, keeping in mind that the plant's vulnerability --

MR. QUIRK: I just think that you are providing a road map on the most vulnerable areas of the plant.

MR. MICHELSON: I have no doubt I could fully agree with you but I've already got the road map. I can go to any number of PRA's and get that road map.

MR. QUIRK: This design differs from the Wash-1400.

MR. MICHELSON: True, but the one that's out there today, not one that you are going to build and may be in operation ten years from now. By that time the potential saboteur will be just as smart about that plant. There are a number of easy ways of getting it. So I don't think you have got much of an argument.

Commercial argument, I don't have a problem with the security argument, why they aren't protecting the present plants from that viewpoint by controlling information.

MR. SCALETTI: The staff -- when we agreed that this information was proprietary was solely from the standpoint of the commercial consideration and not safeguards.

MR. MICHELSON: I have no problem. I'm not questioning that at all. If you started claiming security, then I've got a real problem understanding it.

MR. EBERSOLE: With the advent of the UPPS system I suggested the staff looks into constraining the details of design to make it less well known what might be safe shutdown potentials of the plant which are not so highly exposed should never have been spread across the public. You should have a place to go that's secure, which had not been pre-advertised as a vulnerable part of the design.

MR. OKRENT: I guess that's probably as far as we can get on this subject now.

Let's see. The request is that we begin now with the proprietary session, so Mr. Major has asked that GE verify whether people sitting on my left are all from GE or from the ACRS, as I see it, whether the staff can do the same on their side.

MR. ROSENTHAL: We have one person -- I would like to just offer this to GE -- Dr. Swanson, who has been a contractor to the NRC and assigned proprietary withholding agreements in the past fiscal years and we cannot today find out if he has signed a proprietary agreement for this fiscal year.

He is a contractor to the accident evaluation branch for this year. I'm sure he would be glad to leave his watch with you, but -- he is also employed in the past on matters such as our RDA contract, on A-45. So he is a current contractor to AUB, this fiscal year -- he will be. The paper is floating. He would normally have access. I just cannot confirm. We can't confirm today that it is in force.

MR. QUIRK: Well, he is under current contract or will be soon with the NRC and doesn't that cover the proprietary information agreement?

Okay, I accept.

MR. OKRENT: Well, let me --

(Discussion held off the record.)

MR. OKRENT: Let me hear what GE thinks, with their arrangement.

ABWR DESIGN

A PRESENTATION TO ACRS SUBCOMMITTEES ON GESSAR II/
RELIABILITY & PROBABILISTIC ASSESSMENT

LOS ANGELES, CALIFORNIA

GENERAL ELECTRIC COMPANY
OCTOBER 18-19, 1984

ABWR DESIGN

AET FEASIBILITY STUDY (PHASE I)

O OBJECTIVE OF STUDY

PERFORM FEASIBILITY STUDY THROUGH CONCEPTIONAL DESIGN OF THE NEW TYPE BWR BY ADVANCED ENGINEERING TEAM (AET)

O MEMBERS OF STUDY

INTERNATIONAL COOPERATION OF 5 BWR MAKERS

- HITACHI, TOSHIBA (JAPAN)
- GE (U.S.A.)
- ASEA ATOM (SWEDEN)
- ANSALDO MECCANICO NUCLEARE (ITALY)

O PERIODS OF STUDY

FROM: JULY 1978

TO: JUNE 1979

ABWR DESIGN

STUDY OF A-BWR DEVELOPMENT

PHASE II & III

O OBJECTIVE OF STUDY

DESIGN AND STUDY PERFORMANCE OF ADVANCED BWR (A-BWR) BASED ON RESULTS OF AET FEASIBILITY STUDY.

- BASIC DESIGN AND OPTIMIZED DESIGN
- TEST AND DEVELOPMENT

O MEMBERS OF STUDY

- BASIC DESIGN AND OPTIMIZED DESIGN

  JOINT STUDY OF TEPCO AND 3 MAKERS (GE, HITACHI, TOSHIBA)
- TEST AND DEVELOPMENT

  JOINT STUDY OF 6 ELECTRIC POWER COMPANIES AND 3 MAKERS

O PERIOD OF STUDY

- BASIC DESIGN FROM JULY 1981 TO JUNE 1983
- OPTIMIZED DESIGN ABOUT ONE AND HALF YEARS FROM THE FIRST OF FISCAL YEAR 1984
- TEST AND DEVELOPMENT FROM JULY 1981 TO 1986

ABWR DESIGN

OBJECTIVES OF ABWR DEVELOPMENT

0 STANDARDIZATION OF PLANT DESIGN

0 IMPROVEMENT OF OPERABILITY AND LOAD-FOLLOWING CAPABILITY

0 IMPROVEMENT OF SITING EFFICIENCY AND FINANCIABILITY

0 ESTABLISHMENT OF 1300MWE CLASS PLANT

0 IMPROVEMENT OF PLANT CAPACITY FACTOR MORE THAN 80%

0 REDUCTION OF EXPOSURE
200 MAN-REM/REACTOR YEAR

0 IMPROVEMENT OF RELIABILITY AND SAFETY

ABWR DESIGN

TECHNICAL FEATURES OF A-BWR

- O LARGE PLANT OUTPUT
  ELECTRICAL OUTPUT 1300MWE CLASS
- O REACTOR RECIRCULATION SYSTEM WITH INTERNAL PUMP
- O FINE MOTION CONTROL ROD DRIVE MECHANISM
- O IMPROVED CORE AND FUEL
- O EMERGENCY CORE COOLING SYSTEM
- O REINFORCED CONCRETE CONTAINMENT VESSEL

MAJOR FEATURES OF INTERNAL PUMP SYSTEM

O NO RECIRCULATION PIPE, VALVE, NOR EXTERNAL PUMP → NO ISI OF RECIRCULATION PIPE → REDUCED EXPOSURE

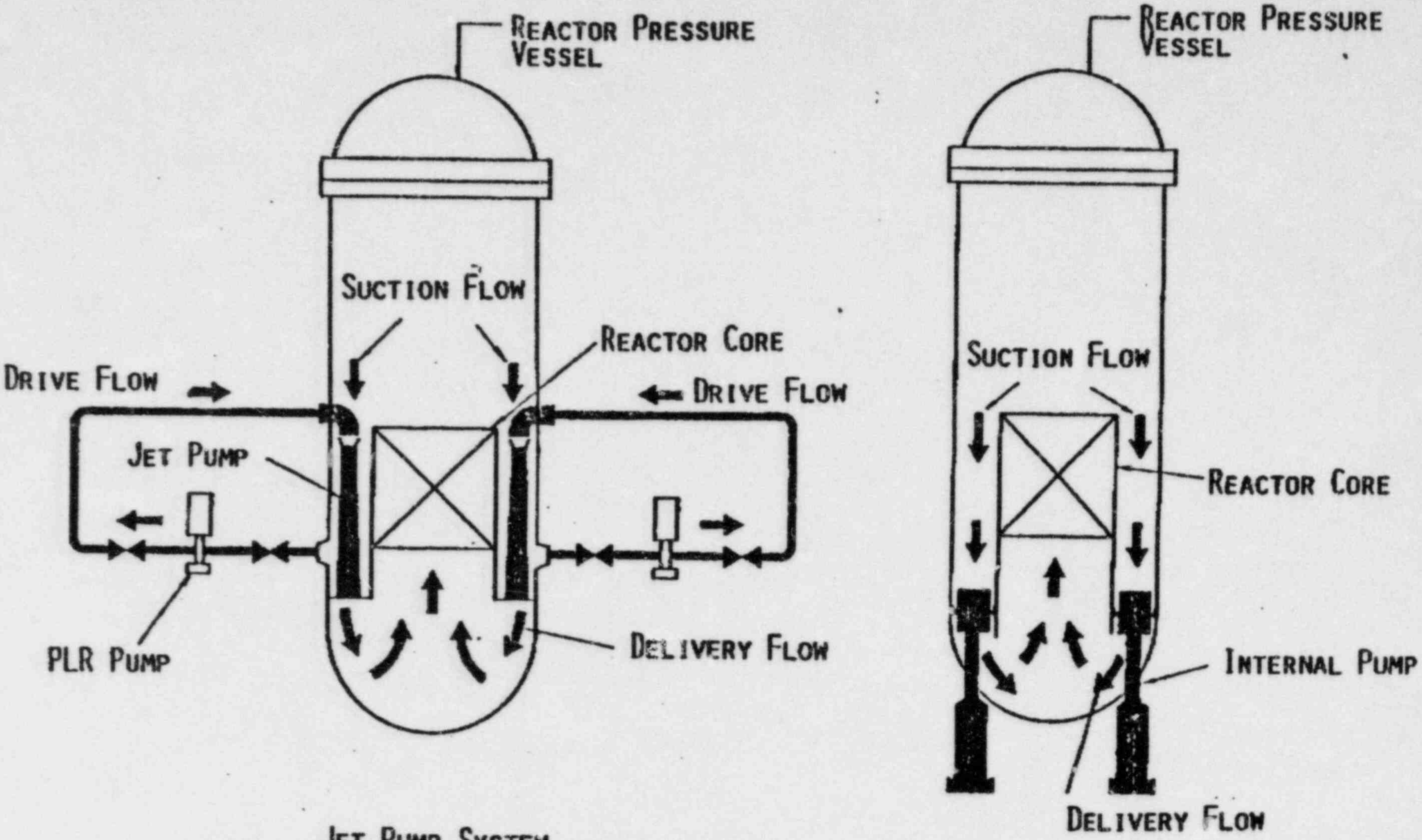
O NO RECIRCULATION PIPE, VALVE, NOR EXTERNAL PUMP → NO EXPOSURE FROM RECIRCULATION PIPE → REDUCED EXPOSURE

O NO RECIRCULATION PIPE BREAK → IMPROVED SAFETY

O NO RECIRCULATION PIPE, VALVE, NOR EXTERNAL PUMP → COMPACT CONTAINMENT → IMPROVED FINANCIABILITY

O NO RECIRCULATION PIPE, VALVE NOR JET PUMP → REDUCED POWER FOR SYSTEM → REDUCED STATION POWER, IMPROVED FINANCIABILITY

O SMALLER INERTIA OF INTERNAL PUMP (M.G SET → THYRISTOR INVERTER) → IMPROVED CONTROL RESPONSE



JET PUMP SYSTEM

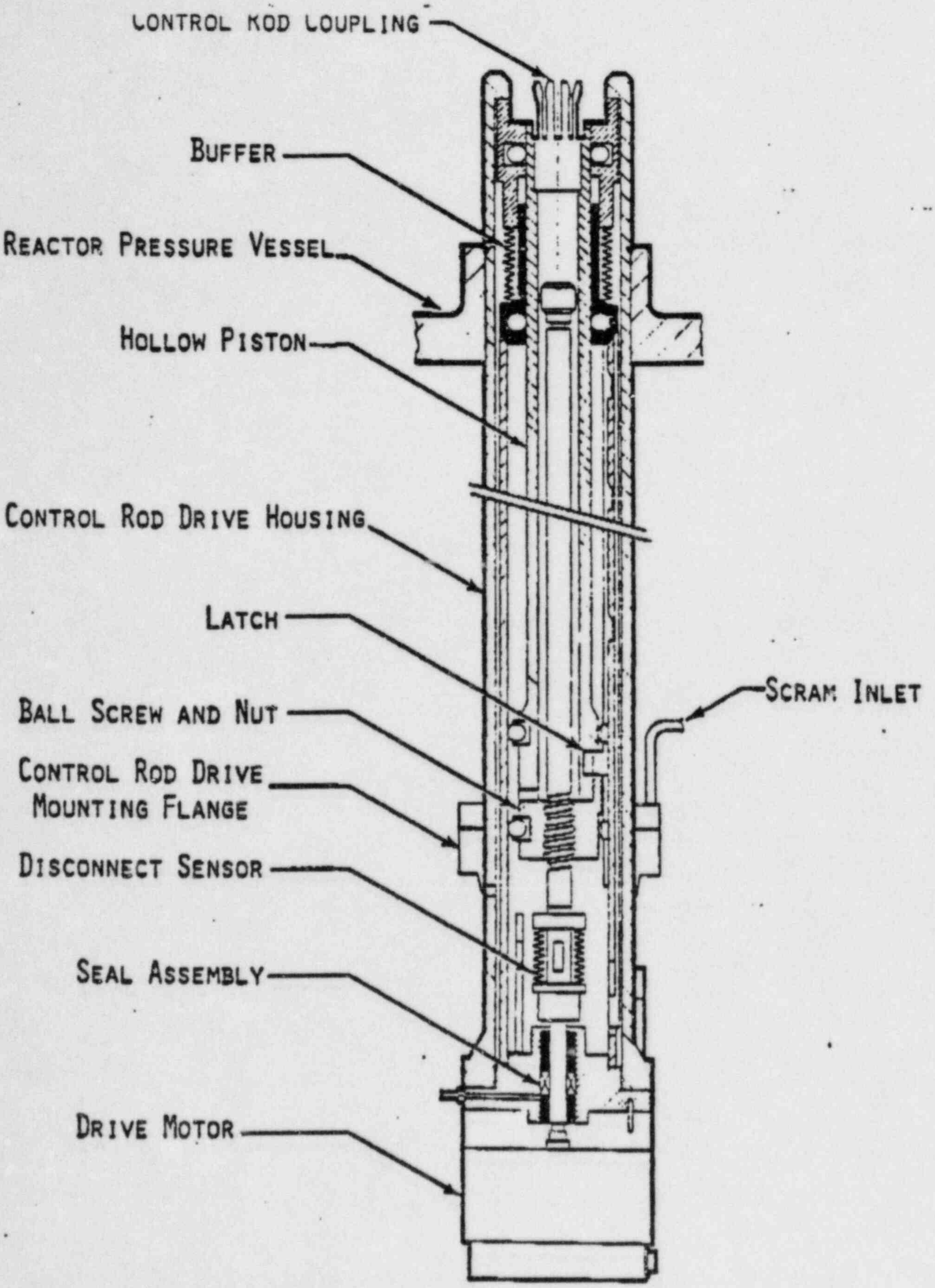
INTERNAL PUMP SYSTEM

COMPARISON OF RECIRCULATION SYSTEM

ABWR DESIGN

FMCRD FEATURES

- DIVERSITY OF DRIVE MECHANISMS
  - ELECTRIC POWER DRIVE
    - FINE MOTION NORMAL DRIVE
    - BACKUP SCRAM
  - HYDRAULIC DRIVE
    - SCRAM
  - IMPROVED SCRAM RELIABILITY BY DIVERSITY
- CONTROL ROD FINE MOTION DRIVE
  - IMPROVED FUEL INTEGRITY
- ELECTRIC POWER GANG MODE DRIVE
  - REDUCED START-UP TIME AND IMPROVED PLANT CAPACITY FACTOR
- NO WEAR SEAL
  - INSPECTION-FREE FOR DRIVE MECHANISM BODY
  - REDUCED INSPECTION REQUIREMENTS AND EXPOSURE
- AVOIDS REACTOR COOLANT INFLOW AT SCRAM
  - REDUCED EXPOSURE DURING ANNUAL INSPECTION

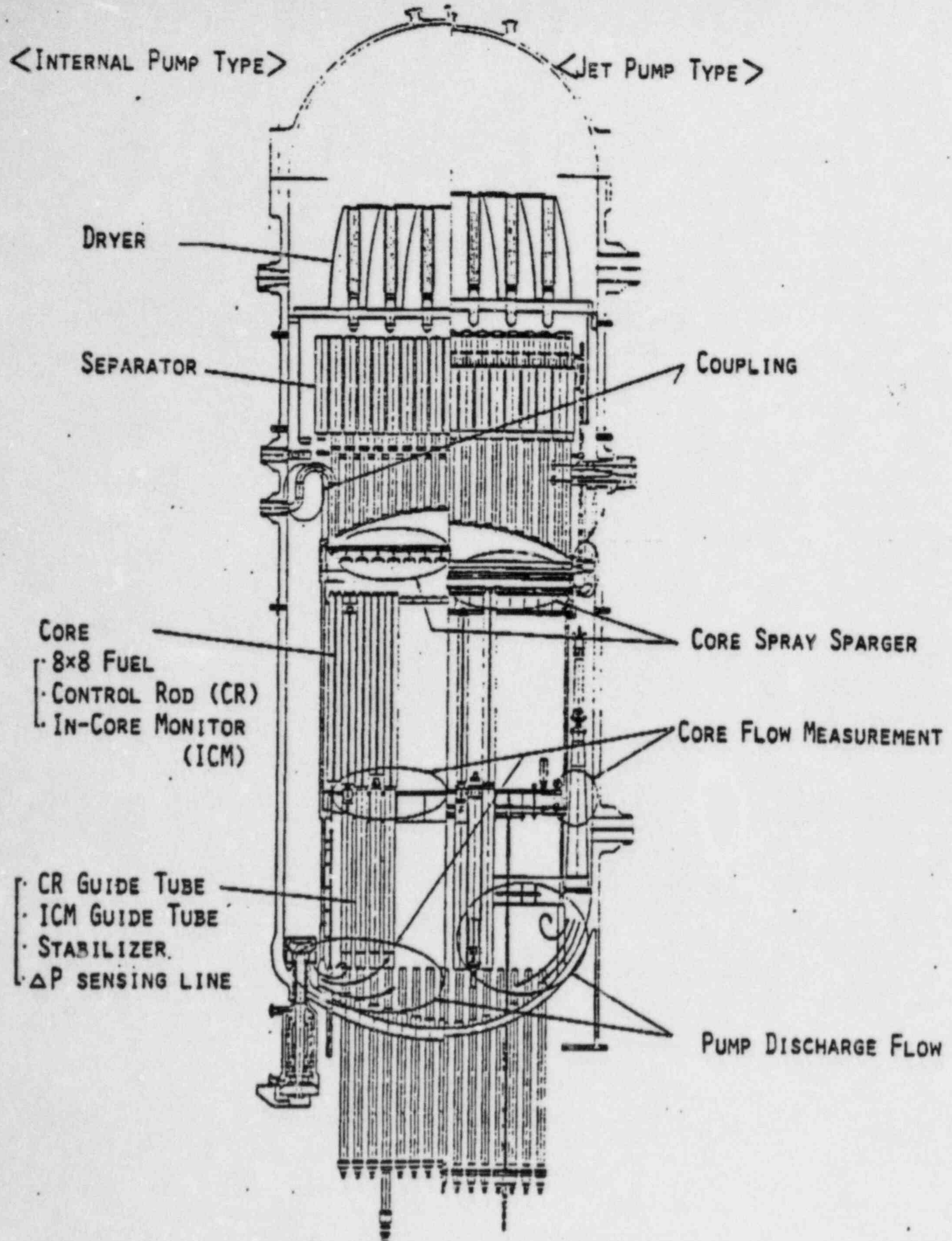


STRUCTURE OF FINE MOTION CONTROL ROD DRIVE (FMCRD)

ABWR DESIGN

IMPROVED CORE AND OPTIMIZED FUEL

- ADEQUATE REACTIVITY DIFFERENCE BETWEEN UPPER AND LOWER FUEL
- LOW REACTIVITY FUEL ASSEMBLIES AROUND THE CONTROL ROD FOR OPERATION
- INCREASED THERMAL MARGIN CAUSED BY ADOPTION OF FMCRD
- CORE FLOW CONTROL WITH WIDE FLOW WINDOW
- LONG OPERATION CYCLE CAUSED BY THE HIGH-ENRICHED FUEL DESIGN



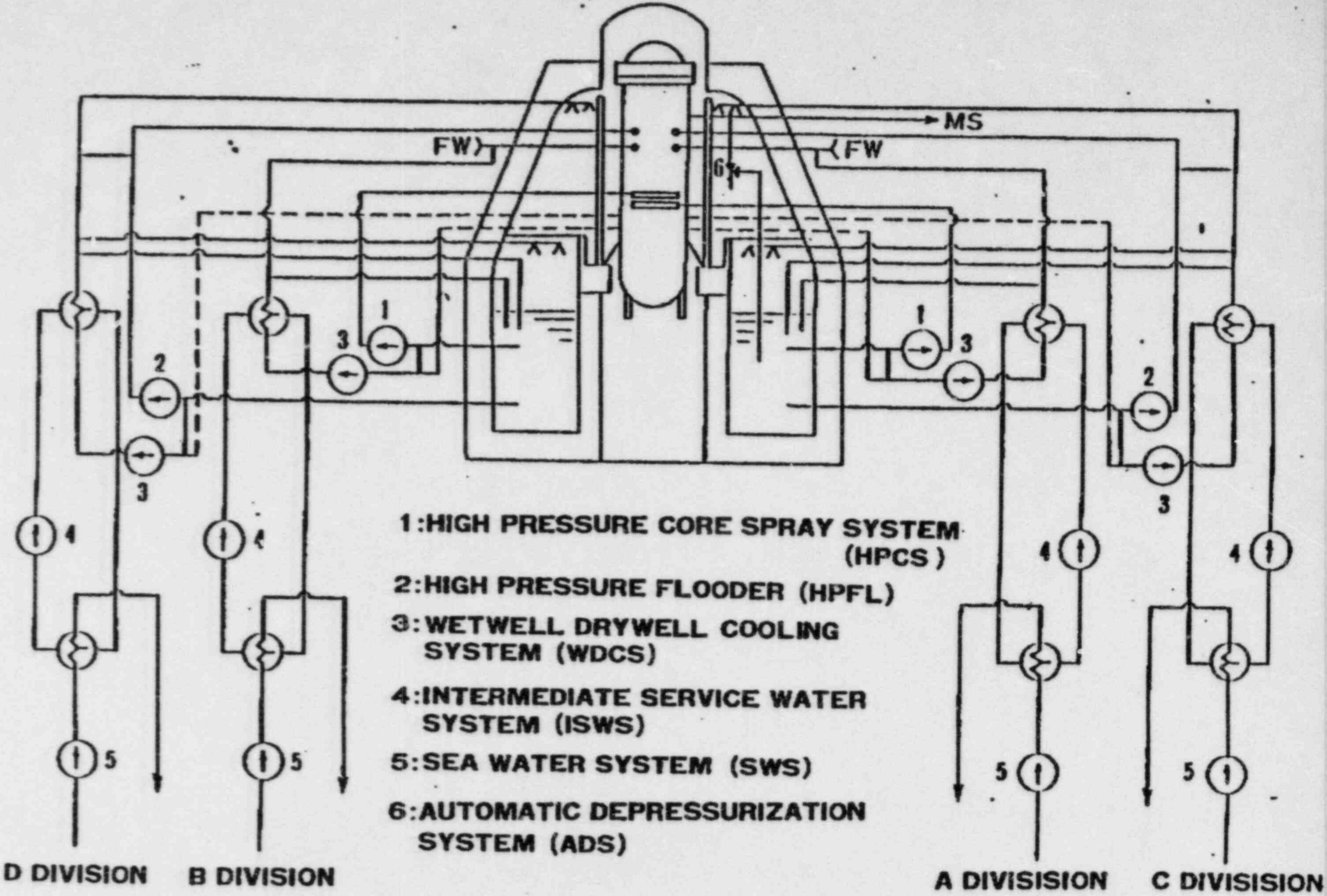
COMPARISON OF PRESSURE VESSEL AND IN-CORE STRUCTURE BETWEEN A-BWR AND BWR-5

ABWR DESIGN
ECCS/WDCS

3 OR 4 DIVISION ECCS/WDCS

1. DESIGN BASIS IS N-1
   (SINGLE FAILURE)

2. RISK IS LESS THAN BWR-5



ECCS/WDCS (4 DIVISION)

ABWR DESIGN

REINFORCED CONCRETE CONTAINMENT VESSEL

O TYPE:

PRESSURE SUPPRESSION TYPE CONTAINMENT

O MATERIAL:

REINFORCED CONCRETE CONTAINMENT WITH STEEL LINER

O MAINTAINABILITY

- WORKING SPACE EQUAL TO IMPROVED MARK I AND MARK II CONTAINMENT
- INDIVIDUAL HATCH TO LOWER DRYWELL

O COUNTERMEASURE FOR DYNAMIC LOAD:

EASILY MITIGATED WITH REINFORCED CONCRETE

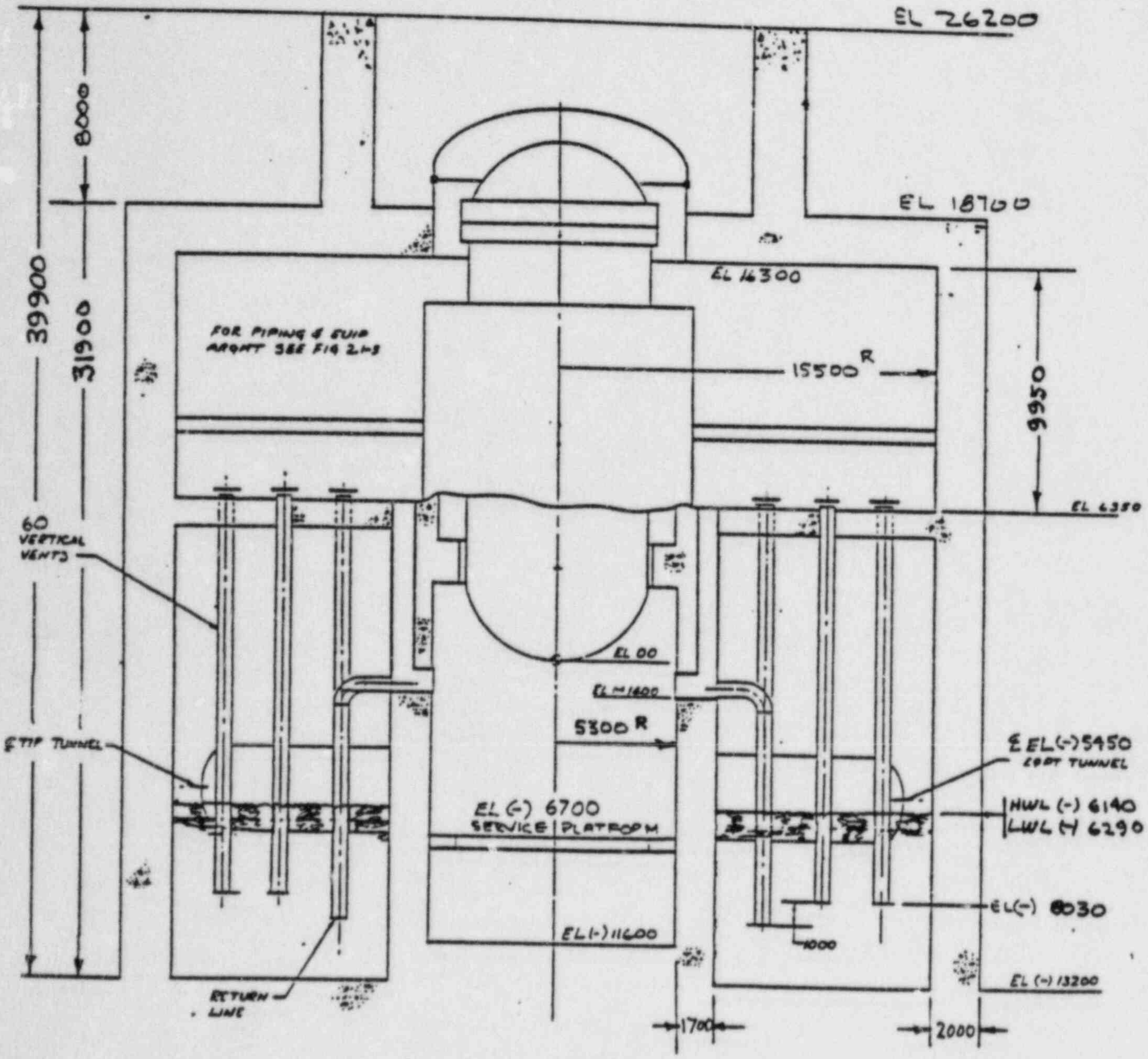
O PIPE WHIP:

DIRECTLY INSTALLED FROM CONTAIMENT RESTRAINT STRUCTURE WALL

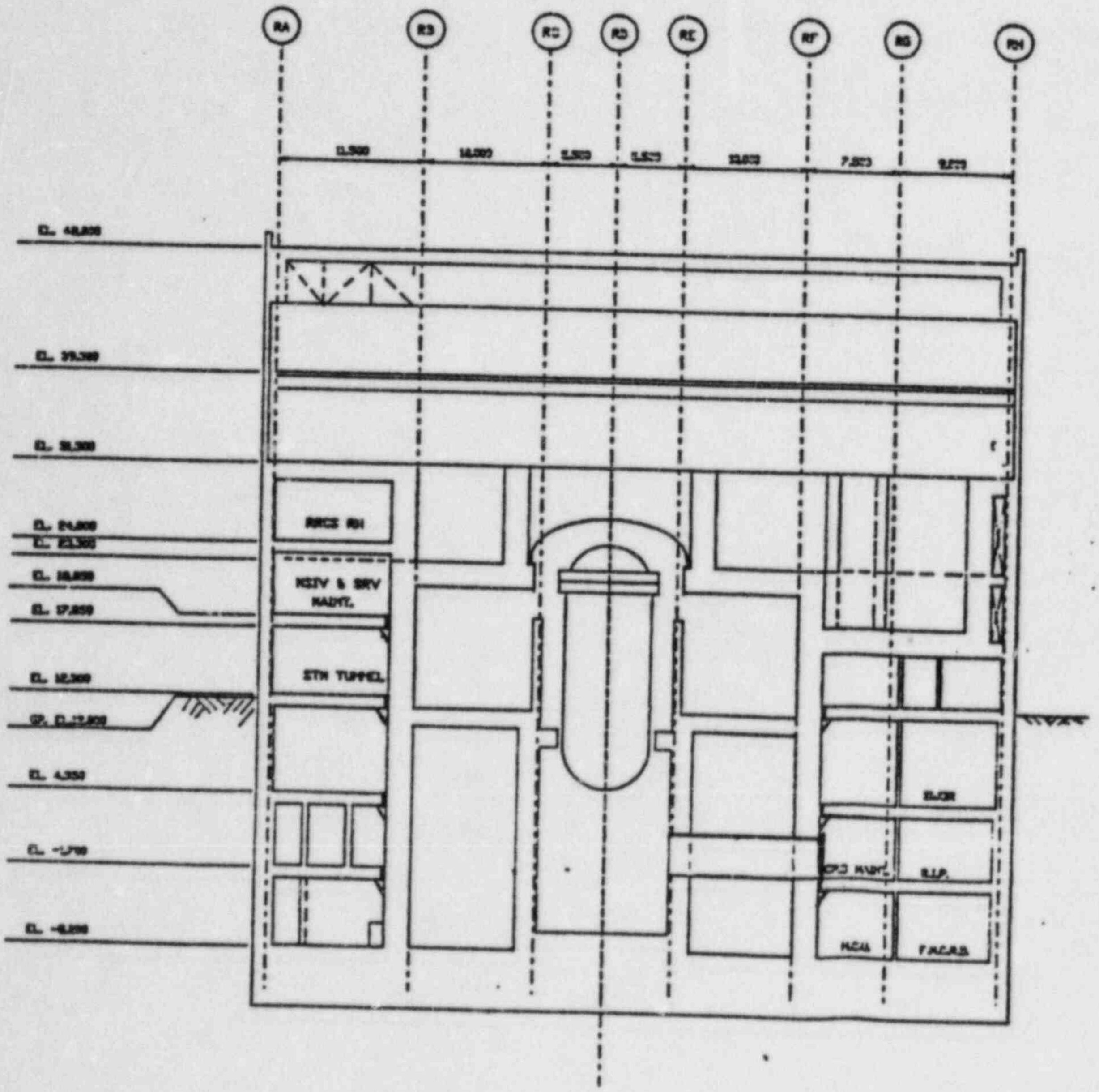
O CONTINUING STUDIES

HORIZONTAL VERSUS VERTICAL VENTS

ABWR CYLINDRICAL
CONTAINMENT DESIGN



ABWR CYLINDRICAL
CONTAINMENT DESIGN



ABWR DESIGN

CURRENT STATUS

- DESIGN STUDIES CONTINUE TO SUMMER 1985
- TECHNICAL ASPECTS CURRENTLY UNDER REVIEW
  - INCREASING THERMAL POWER CAPACITY FROM 3845 MWT TO 3926 MWT
  - ADAPTING TO A 52 INCH TURBINE GENERATOR
  - FOUR VERSUS THREE ECCS DIVISONS
  - CONTAINMENT DESIGN

GESSAR II PROBABILISTIC RISK ASSESSMENT

A PRESENTATION TO ACRS SUBCOMMITTEES
ON GESSAR II/RELIABILITY & PROBABILISTIC ASSESSMENT

LOS ANGELES, CALIFORNIA

GENERAL ELECTRIC COMPANY
OCTOBER 18-19, 1984

OVERVIEW

PROBABILISTIC RISK ASSESSMENT (PRA)

- OBJECTIVE
- RESULTS
- SCOPE - DEFINITION OF BWR/6 REACTOR PLANT
- PRA METHODOLOGY
- SITE SELECTION
- MAJOR TASKS OF THE PRA

GESSAR II PROBABILISTIC RISK ASSESSMENT

- OBJECTIVE - QUANTIFY THE SAFETY OF THE BWR/6 STANDARD PLANT DESIGN
  - Provide a Comprehensive Assessment of Public Risk Resulting From Plant Operation
  - Identify Major Contributors to Risk
- RESULTS
  - Likelihood of an Accident Progressing to Core Damage Well Below WASH-1400 and Interim Safety Goal
  - Plant Risk Well Below WASH-1400 and Interim Safety Goals

GESSAR II PROBABILISTIC RISK ASSESSMENT

- SCOPE - DEFINITION OF BWR/6 REACTOR PLANT
  - Standard 238 Plant, Plus
    - Post-TMI Modifications
    - ATWS Alternate 3A + BF Modifications
  - Reference Mark III Containment
    - Free Standing Steel Containment
    - Concrete Shield Building
    - Drywell, Primary Containment and Secondary Containment
  - RSS Site #6 Meteorology and Demography
    - Atlantic Coastal
    - 31.4 Million People (In 500 Mile Radius)
  - Electric Grid Reliability Based on the PRA - N.J. - MD (PJM) Grid Experience (In Site #6)
    - .05 Losses of Off-Site Power/Year
    - 99.5% Recovery Within 12 Hours

GESSAR II PROBABILISTIC RISK ASSESSMENT

- METHODOLOGY
  - MODIFIED WASH-1400 (FAULT TREES/EVENT TREES)
  - CONSISTENT WITH PRA PROCEDURES GUIDE (NUREG/CR-2300)
  - REALISTIC SUCCESS CRITERIA
  - EMERGENCY OPERATING PROCEDURES
  - REALISTIC FISSION PRODUCT MODELING

MAJOR DIFFERENCES

- COMPARISON WITH WASH-1400 (RSS) METHODS
  - Reduction in Frequency of Initiating Events
  - More Comprehensive Fault Trees
  - More Comprehensive Event Trees
  - More Realistic Success Criteria
  - Credit for Feedwater and Condensate Pumps
  - Additional ATWS Sequence Included
  - More Release Categories
  - Improved Codes

COMPARISON OF ESTIMATED FREQUENCY OF CORE DAMAGE

REACTOR	FREQUENCY OF EVENT PER REACTOR YEAR
RSS BWR/4 MARK I	3×10^{-5}
BWR/6 MARK III	5×10^{-6}

- CORE DAMAGE FREQUENCY BELOW NRC GUIDELINE (1×10^{-4}/YEAR)
- ATTRIBUTED TO MANY SAFETY FEATURES
  - LONG TIME TO CORE DAMAGE
  - MULTIPLE AND DIVERSE HIGH AND LOW PRESSURE PUMPS AND POWER SUPPLIES
  - REALISTIC SUCCESS CRITERIA
  - MULTIPLE HEAT SINKS AND HEAT REMOVAL MODES
  - SIMPLE ONE LOOP OPERATION
  - LONG TIME TO LOSS OF CONTAINMENT INTEGRITY BY STEAM OVERPRESSURIZATION

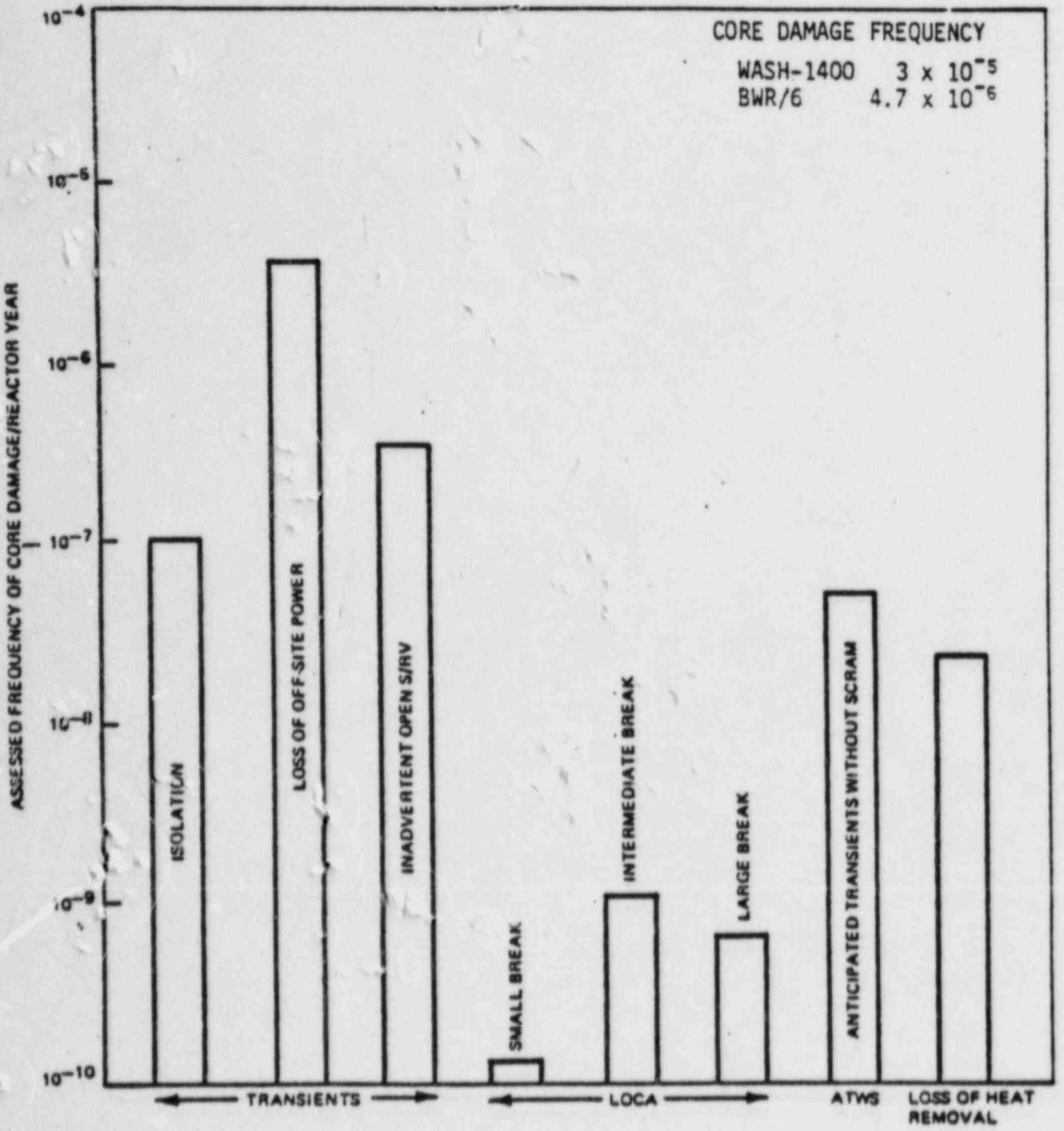
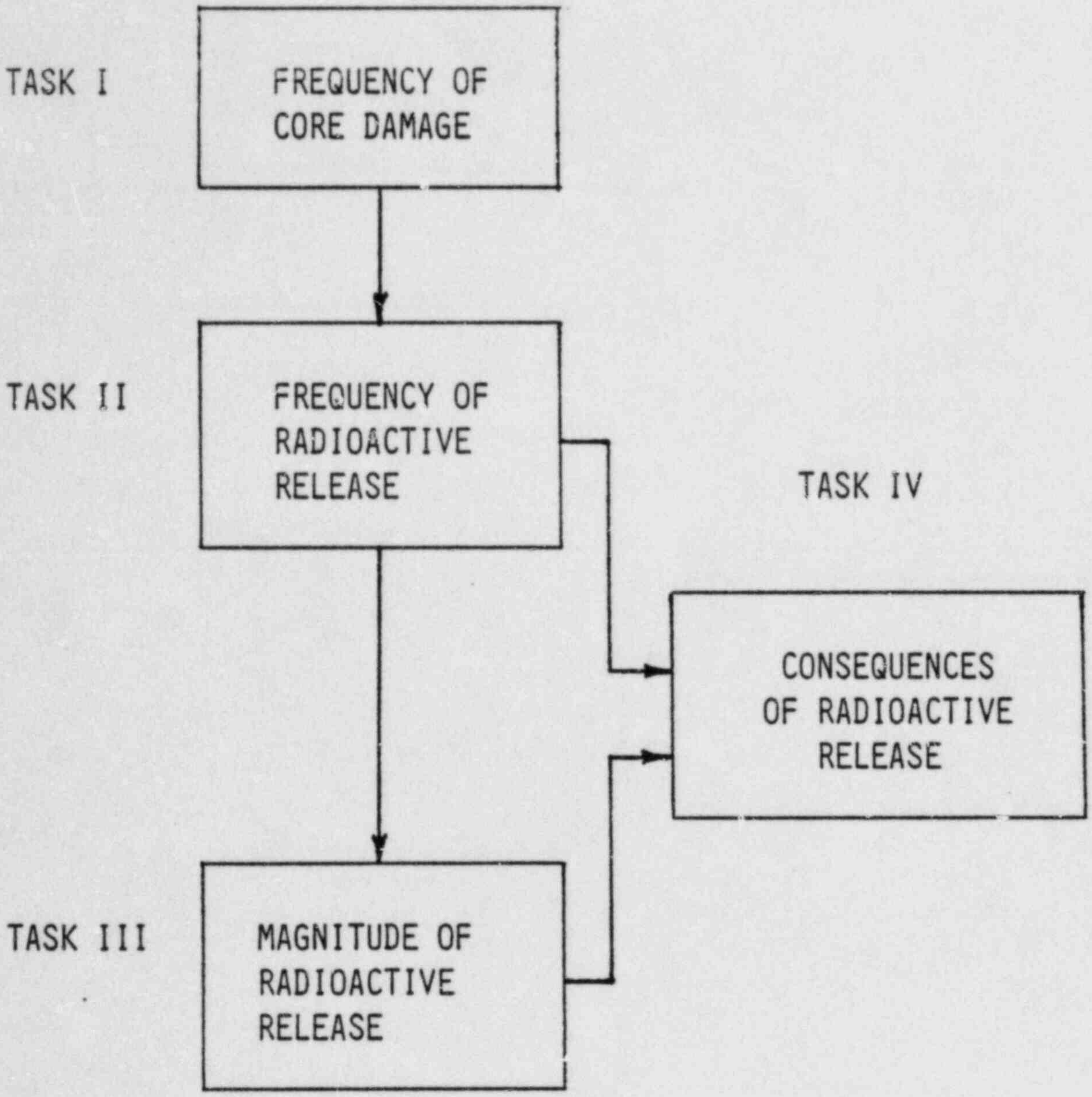
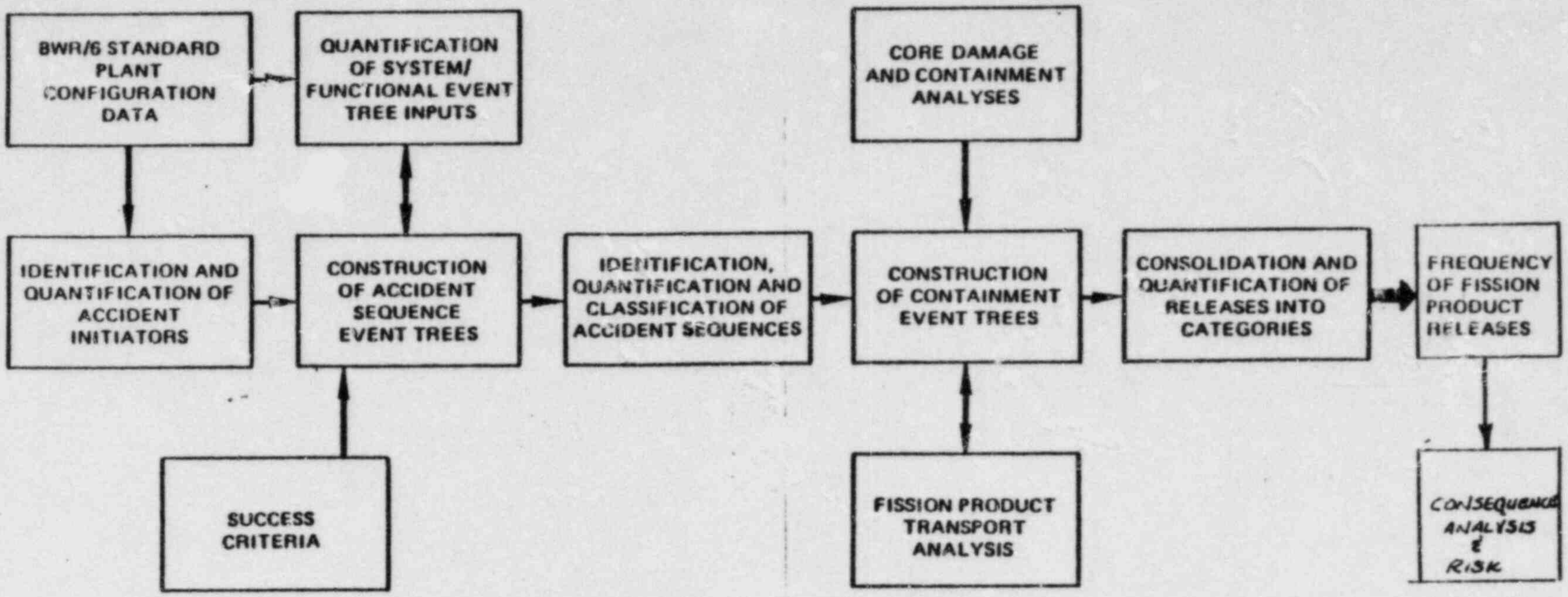


Figure 7.1-1 Distribution of Core Damage Events



MAJOR TASKS OF THE PRA



Overview of Methodology for Assessing Frequency of Core Damage and Fission Product Releases

CONTAINS
Proprietary
INFORMATIO

CORE DAMAGE PROBABILITY

A PRESENTATION TO ACRS SUBCOMMITTEES
ON GESSAR II/RELIABILITY & PROBABILISTIC ASSESSMENT

LOS ANGELES, CALIFORNIA

GENERAL ELECTRIC COMPANY
OCTOBER 18-19, 1984

CORE DAMAGE PROBABILITY

- OVERVIEW
- ACCIDENT SEQUENCE EVENT TREE INPUTS
  - SUCCESS CRITERIA
  - INITIATING EVENT FREQUENCIES
- ACCIDENT SEQUENCE PROBABILITY DETERMINATION - EXAMPLE
- FAULT TREE INPUTS
- DOMINANT ACCIDENT SEQUENCES
- SUMMARY OF RESULTS
- CORE DAMAGE UNCERTAINTY ANALYSIS
- CONTAINMENT EVENT TREES

OVERVIEW

DETERMINATION OF CORE DAMAGE FREQUENCY

- 0 METHODS
  - o WASH-1400
    - Accident Initiator
    - Fault Tree
    - Event Tree
  - o PRA Procedures Guide (NUREG/CR-2300)
- 0 CODES
  - o WAM Code (Fault Tree Quantification)
  - o Event Tree Quantification Code
- 0 DATA SOURCES
  - o GE and Industry Data

ACCIDENT SEQUENCE EVENT TREE INPUTS

TYPICAL SUCCESS CRITERIA

0 INITIATING EVENTS: TRANSIENT

0 SYSTEMS NEEDED FOR CORE COOLING

RCIC OR,

HPCS OR,

1 FEEDWATER PUMP OR,

$$3\ \text{S/RV} + \left\{ \begin{array}{l} \text{LPCS OR,} \\ \text{1 OF 3 LPCI OR,} \\ \text{1 CONDENSATE PUMP} \end{array} \right.$$

0 SYSTEMS NEEDED FOR CONTAINMENT HEAT REMOVAL

1 OF 2 RHR LOOPS

OR

MAIN CONDENSER

TYPICAL SUCCESS CRITERIA

INITIATING EVENT: Turbine Trip Without Scram

SUCCESSFUL COMBINATIONS:

1 SLC + RCIC + RHR

or, 1 SLC + HPCS + Main Condenser

(Combinations Require L8 Trip or Manual Action, RPT and Feedwater Runback).

RESULTS PRESENTED IN A MANNER THAT RELATES DIRECTLY TO EVENT TREES

GENERAL ELECTRIC
PROPRIETARY INFORMATION

Table A.1-1

SUMMARY OF BWR/6 STANDARD PLANT INITIATING EVENT FREQUENCIES

Initiating Event	Frequency (Events/Year)		Where Used-Event Tree Section in Appendix C
1. Reactor Shutdown		2.77	C.2
A. Planned Shutdown	1.96		
B. Other Scrams	0.81		
2. Turbine Trip		1.32	C.3
3. Isolation		1.97	C.4
A. Feedwater Failure	1.18		
B. Immediate Isolation	0.79		
4. Loss of Offsite Power (LOOP)		0.05	C.5
5. Inadvertent Open Safety/Relief Valve (IORV)		0.4	C.6
TOTAL		6.51	

ACCIDENT SEQUENCY PROBABILITY DETERMINATION

EXAMPLE

- O REACTOR SHUTDOWN EXAMPLE

- O FUNCTIONAL FAULT TREE
  - o TREATMENT OF DEPENDENCIES BETWEEN SYSTEMS

- O SYSTEM FAULT TREE
  - o TREATMENT OF DEPENDENCIES WITHIN SYSTEM

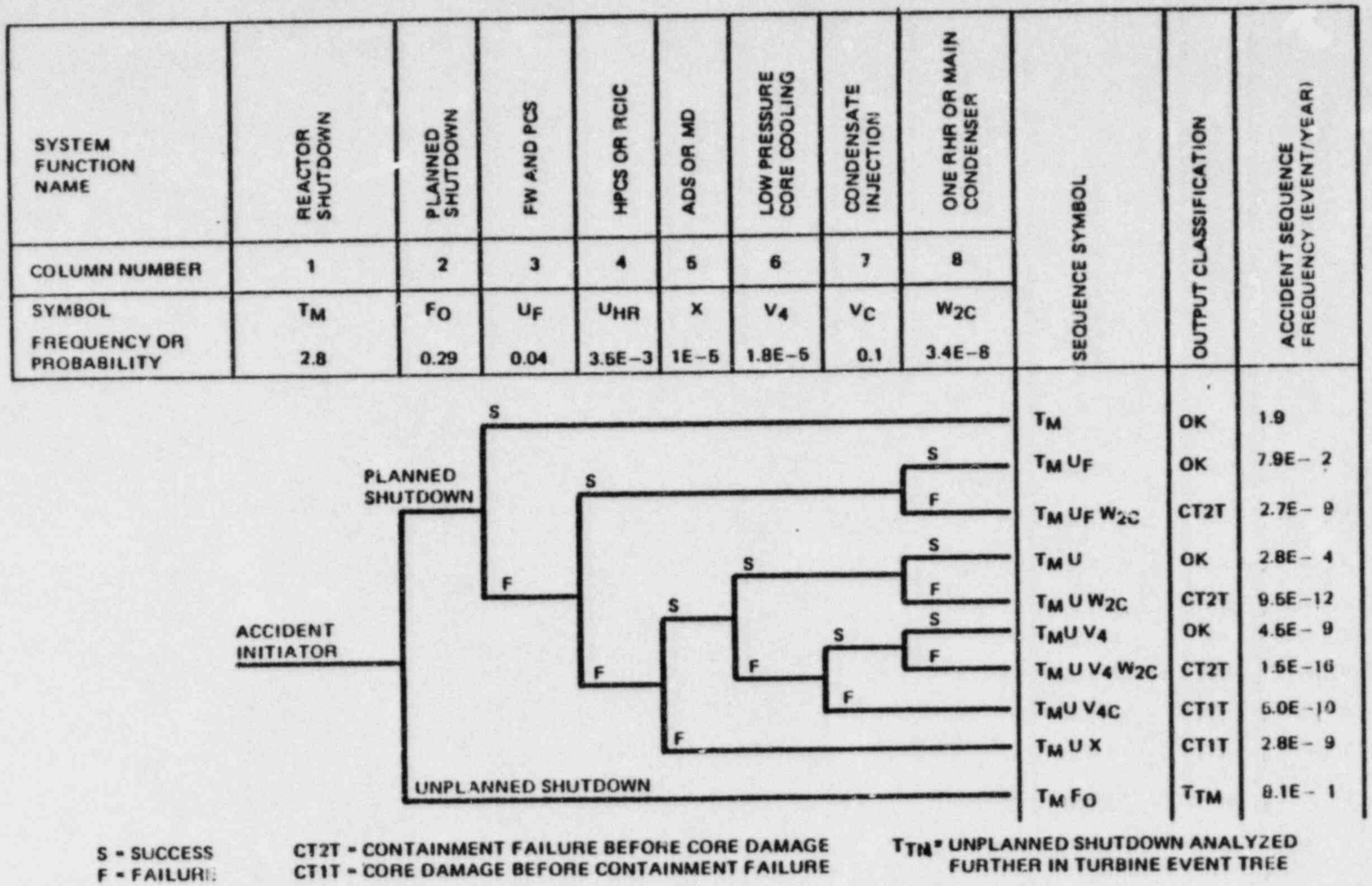
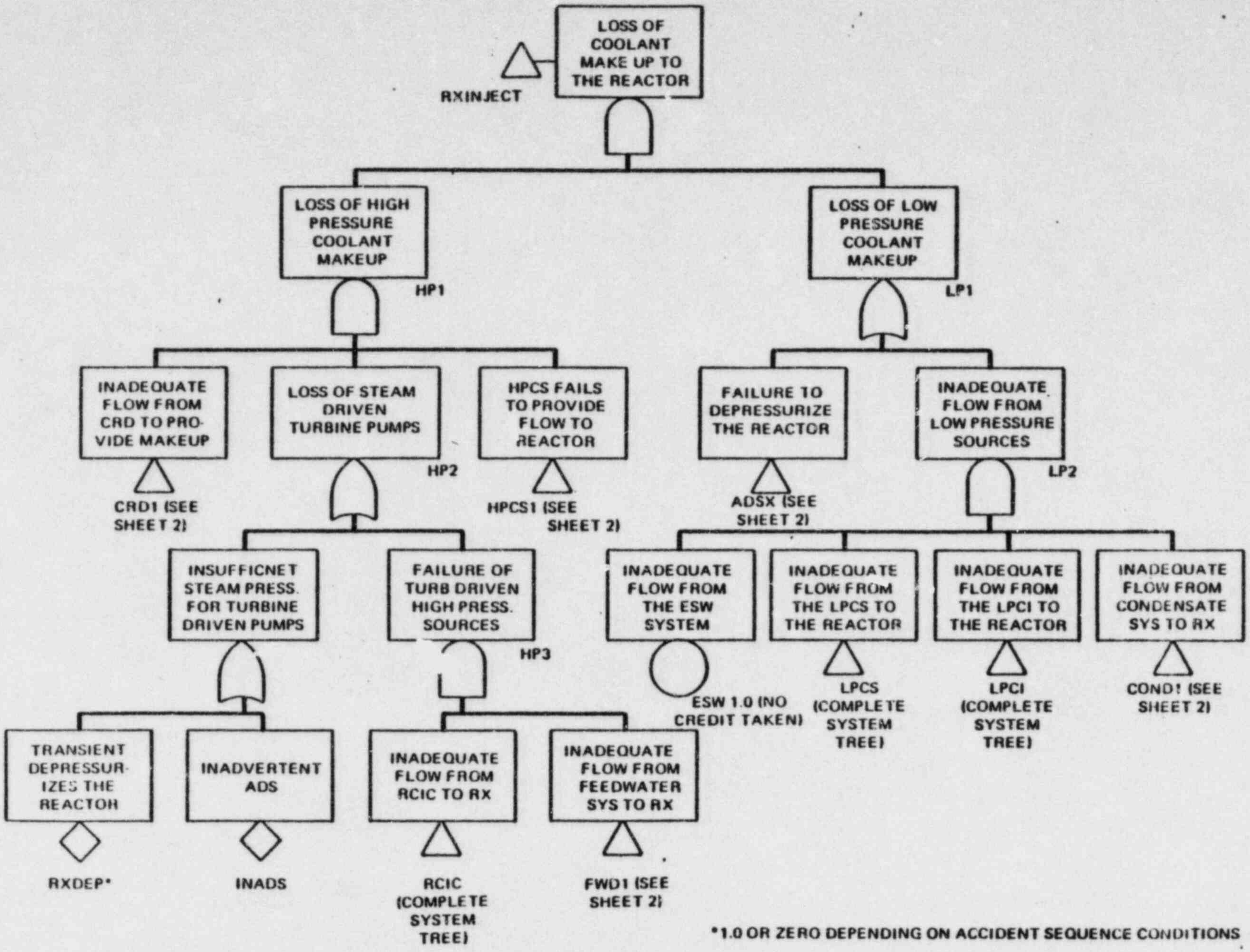


Figure C.1-2. Example Event Tree - Reactor Shutdown



*1.0 OR ZERO DEPENDING ON ACCIDENT SEQUENCE CONDITIONS

Figure D.1-1. Functional Fault Tree for Reactor Coolant Makeup
(Sheet 1 of 2)

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION
Class III

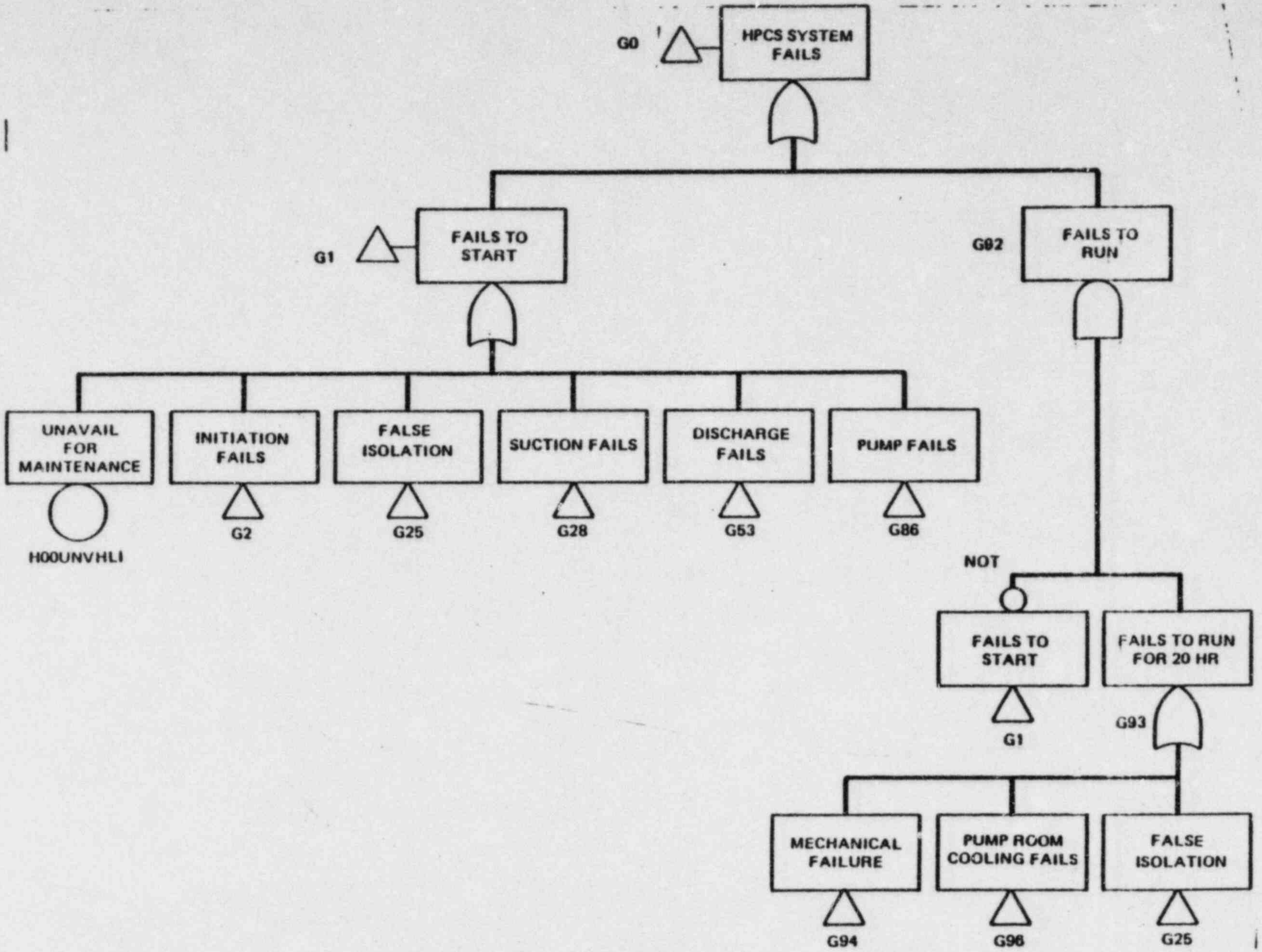


Figure D.2-1. HPCS (High Pressure Core Spray System) (Sheet 1 of 15)

FAULT TREE INPUTS

- O HARDWARE FAILURE RATE INPUT
- O TREATMENT OF HUMAN ERROR
- O COMMON CAUSE FAILURE
- O UNAVAILABILITY DUE TO MAINTENANCE
- O PROBABILITY OF RECOVERY
  - o Loss Of Off-Site Power
  - o Residual Heat Removal System
  - o Power Conversion System

FAULT TREE INPUTS

- HARDWARE FAILURE RATE INPUT
    - Data Are Generic
    - Mean Failure Rates
    - Data Sources
        - NRC Data
        - GE Data
        - WASH 1400
        - IEEE 500
        - Government Industry Data Exchange Program (GIDEP)
        - Nuclear Power Reliability Data System (NPRDS)
        - MIL-HDBK-217C

FAULT TREE INPUTS

TREATMENT OF HUMAN ERROR

- USE "HANDBOOK OF HUMAN RELIABILITY ANALYSIS"
  A. D. SWAIN/H. E. GUTTMANN

- CONSIDER STRESS AND DEPENDENCY

  EXAMPLE:

  - Human Error Rate for Failure To Follow Established Procedure x = 0.01
  - Stress Level Is Moderate (2x)
  - Level of Dependence Between Tasks if Moderate (15%)

- TREATED AS A COMPONENT IN FAULT TREE

- SPECIFIC EXAMPLE

  - Miscalibration of Sensors

FAULT TREE INPUTS

COMMON CAUSE FAILURE

- O HOW TREATED
  - o Equipment or Signal Commality
    - - Component in Fault Trees
  - o Divisional Services, i.e., Common Power Supplies or Service Water
    - - Component in Fault Tree
  - o System Dependency
    - - Component Functional Fault Tree
  - o On-Site or Off-Site Power Unavailability
    - - Estimated From Experience
  - o Human Errors
    - - Handbook of Human Reliability Analysis

Table A.4-1
ESTIMATED
SYSTEM UNAVAILABILITY UPON DEMAND, $\bar{A}_m$ ATTRIBUTED TO ON-LINE MAINTENANCE

	As Reported by Operating BWR/4's				Projected	
System	Number of Events	Total Hours	Average Time (hrs)	Reported $\bar{A}_m$/Year	Total Hours	Expected A_m%/Year
HPCI	22	507.0	34.5	0.69	NOT APPLICABLE[1]	
RCIC	20	605.5	30.3	0.82	817.4	1.1
LPCS[2]	7	86.1	12.3	0.06 (per loop)	135.3	0.1
LPCI[2]	14	213.2	15.2	0.15 (per loop)	304.6	0.2 (per loop)
RHR[2]	11	291.0	26.4	0.20 (per loop)	423.3	0.3 (per loop)
D/G[3]	14	111.5	8.0	0.05 (per D/G)	159.3	0.1 (per D/G)
HPCS	——— NOT APPLICABLE[1] ———				334.0	0.5

NOTES:

1 In BWR/6, HPCS replaces HPCI
2 Two loops/system
3 Four D/G's

SUMMARY OF RESULTS

FREQUENCY OF CORE DAMAGE

INITIATING EVENT	Frequency Of Core Damage (Events/Year)	
Transients		
	Loss Of Offsite Power	88.0%
	Inadvertent Open S/R	9.4%
	Loss of Feedwater/ Isolation	2.4%
	Others	0.2%
LOCA		0.1%
TOTAL	$4.7\text{x}10^{-6}$	100%
WASH-1400	$3.\text{x}10^{-5}$	

DOMINANT CORE DAMAGE SEQUENCE

FREQUENCY

INITIATING EVENT	SYSTEM FAILURE IN SEQUENCE	FREQUENCY (EVENTS/YEAR)
Loss Of Offsite Power	HPCS RCIC 3 LPCI LPCS	$4.0\text{x}10^{-6}$
	HPCS RCIC ADS	$1.3\text{x}10^{-7}$
	TOTAL	$4.1\text{x}10^{-6}$

GESSAR II INTERNAL EVENT PRA
UNCERTAINTY ANALYSIS

METHODOLOGY

- O DEFINITION OF UNCERTAINTIES

  1. INPUT PARAMETERS (DATA VALUES)

     - o EVENT INITIATING FREQUENCIES
     - o COMPONENT FAILURE RATES
     - o TEST AND MAINTENANCE OUTAGE TIMES
     - o HUMAN ERROR RATES
     - o RECOVERY RATES

  2. PHENOMENOLOGICAL MODELING (SUCH AS HYDROGEN COMBUSTION PHENOMENA)

     - o CONTAINMENT EVENT TREE BRANCH PROBABILITIES

- O METHOD OF ANALYSIS

  1. MONTE CARLO SIMULATION METHOD USED

  2. COMPUTATIONS PERFORMED USING "SPASM" CODE

  3. BASED ON THE PROPAGATION OF UNCERTAINTIES IN THE MEAN VALUE ESTIMATES THROUGH FAULT AND EVENT TREES

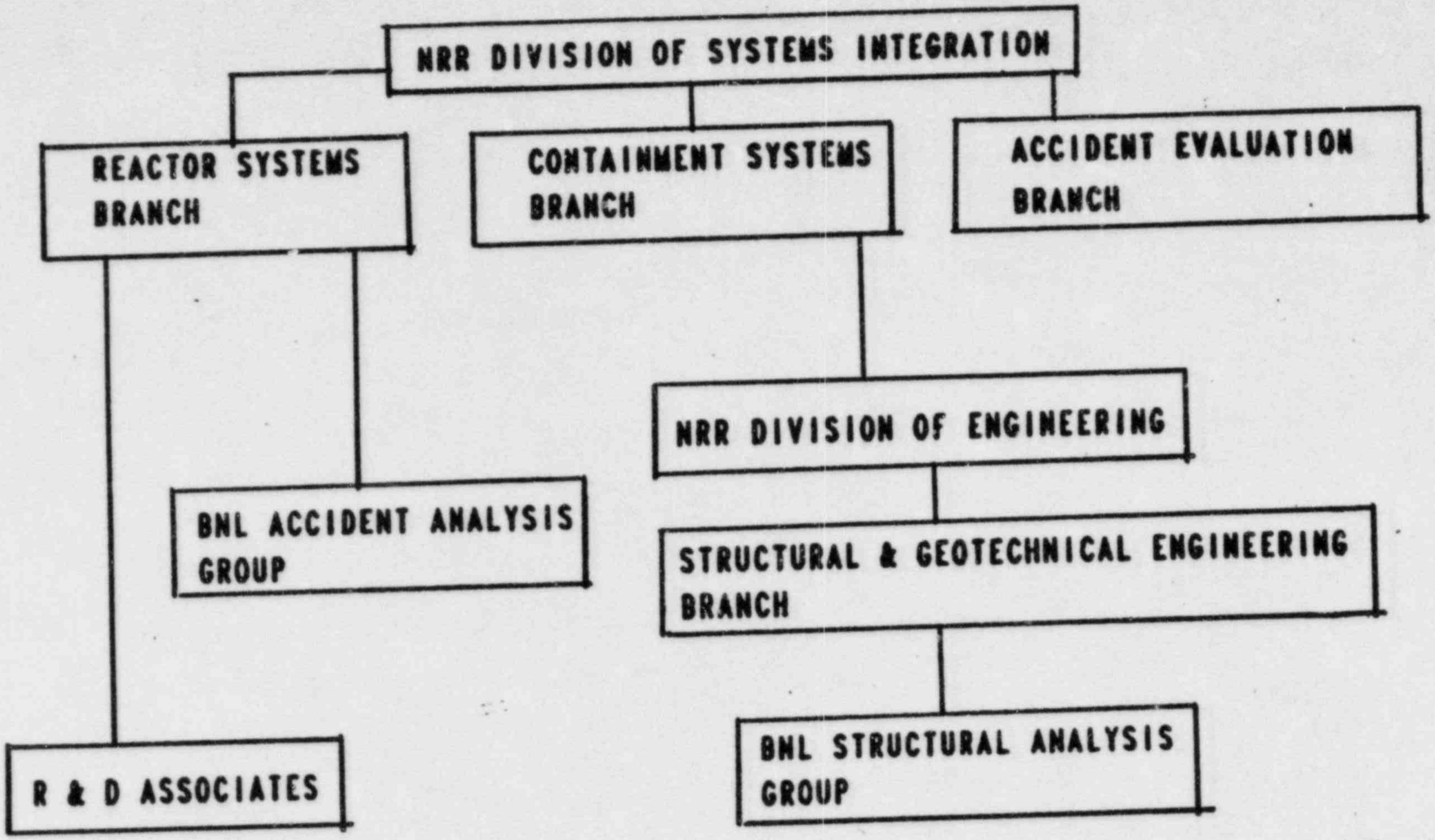
UNCERTAINTY ANALYSIS RESULTS

PARAMETER	CORE DAMAGE FREQUENCY (EVENTS/REACTOR YEAR)
MEAN	3.17E-6
MEDIAN	2.44E-6
5% CONFIDENCE LIMIT, X.05	8.91E-7
95% CONFIDENCE LIMIT, X.95	7.72E-6
ERROR FACTOR = $\sqrt{\frac{X_{.95}}{X_{.05}}}$	= 3

CONTAINMENT EVENT TREES

- 0 PROVIDE THE MODEL FOR PROPAGATION OF ACCIDENT SEQUENCES THROUGH THE CONTAINMENT
- 0 PROVIDE INPUTS TO CORRAL FOR CALCULATION OF RADIOACTIVE RELEASE TERMS
- 0 PROVIDE FREQUENCY INPUTS TO CRAC FOR CALCULATION OF RISK

REVIEW PARTICIPANTS



NRR DIVISION OF SYSTEMS INTEGRATION

REACTOR SYSTEMS BRANCH

CONTAINMENT SYSTEMS BRANCH

ACCIDENT EVALUATION BRANCH

NRR DIVISION OF ENGINEERING

BNL ACCIDENT ANALYSIS GROUP

STRUCTURAL & GEOTECHNICAL ENGINEERING BRANCH

BNL STRUCTURAL ANALYSIS GROUP

R & D ASSOCIATES

REVIEW METHODS

- GENERALLY SIMILAR METHODS/TOOLS AS WAS USED IN THE INDIAN POINT/ZION PRA (MARCH, CORRAL, CRAC) WITH THE FOLLOWING EXCEPTIONS :
  - FISSION PRODUCT RELEASE AS FUNCTION OF CORE TEMPERATURE
  - PERMANENT RETENTION OF FISSION PRODUCTS IN PRIMARY SYSTEM
  - HIGH SUPPRESSION POOL DECONTAMINATION FACTORS (DF) USED

- ASTPO & QUEST INDICATE HIGH UNCERTAINTY IN SOURCE TERMS, THEREFORE A RANGE OF SOURCE TERMS WAS USED. ALSO, TOTAL PLANT RISK NOT REPORTED.

PREDICTED CONSEQUENCE RESULTS

- STAFF PREDICTS VERY SMALL AVERAGE EARLY FATALITIES AND ONLY IF HIGH RANGE FISSION PRODUCT RELEASES ARE ASSUMED.

- PREDICTED LATENT FATALITIES WERE ALSO LOW.

REASONS FOR LOW CONSEQUENCES

1. MK-III CONTAINMENT FEATURES RESULT IN RELATIVELY LOW FISSION PRODUCT RELEASE FOR ALL INTERNAL EVENT SEQUENCES.

2. IMPROVED METHODOLOGY WAS USED IN ANALYSIS.

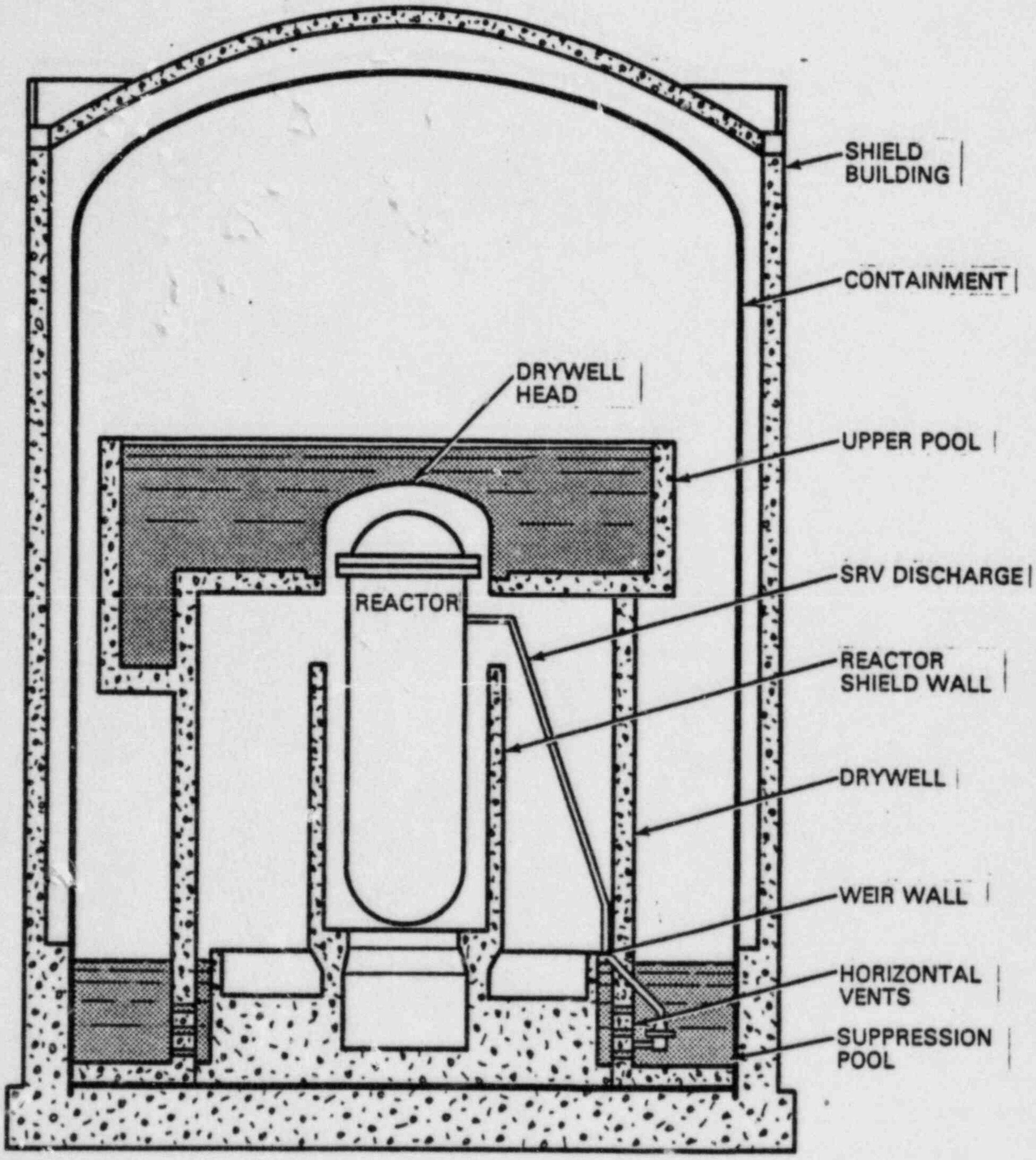


Figure 15.4 GESSAR II MARK III CONTAINMENT

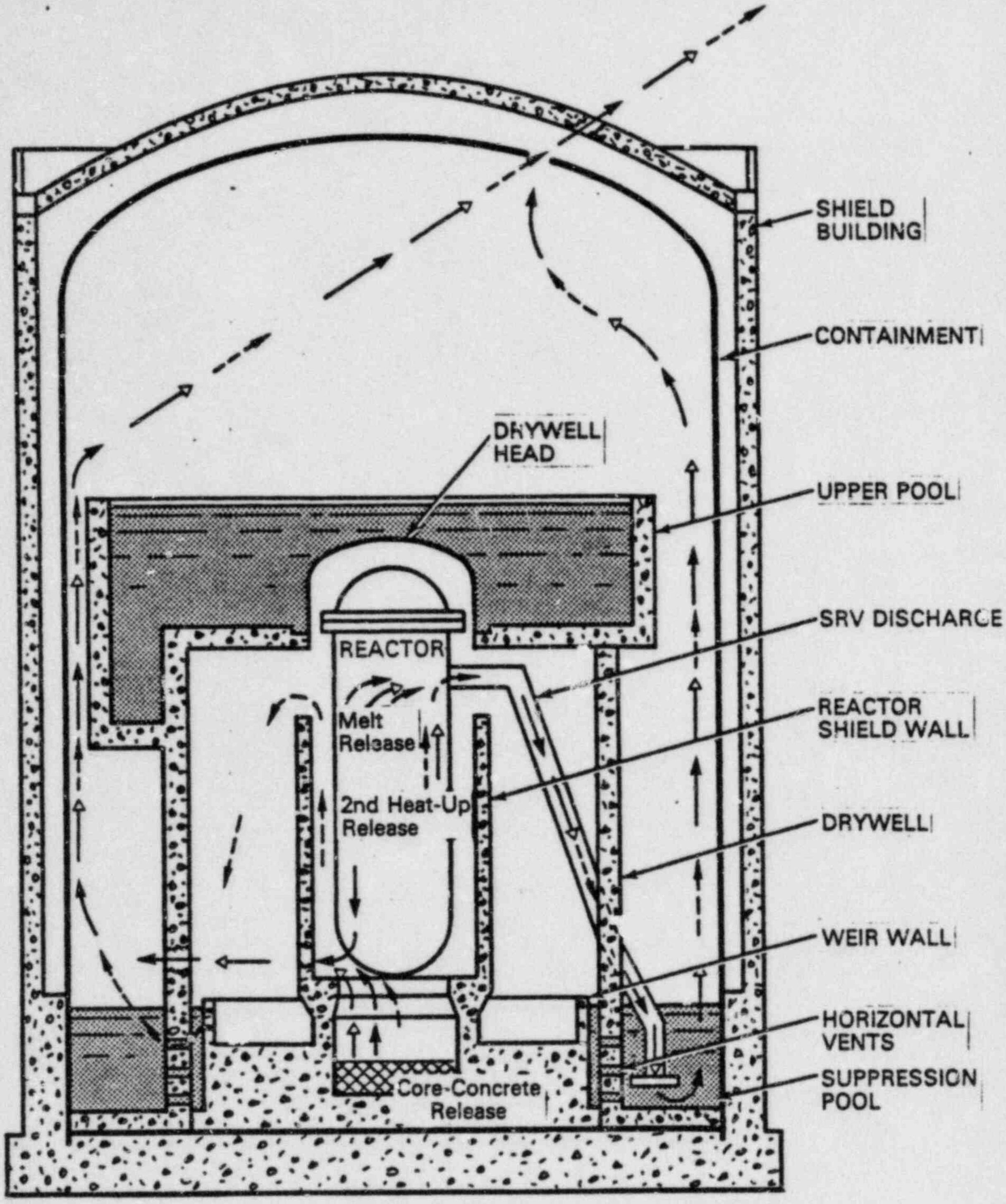


Figure 15.1 GESSAR II MARK III CONTAINMENT

SUPPRESSION POOL SCRUBBING CODE

——→ Least Scrubbing

——▷ Intermediate Scrubbing

– – → Greatest Scrubbing

10-19
pm

CORE DAMAGE PHENOMENA, FISSION PRODUCT TRANSPORT AND CONSEQUENCE ANALYSIS

A PRESENTATION TO ACRS SUBCOMMITTEES
ON GESSAR II RELIABILITY & PROBABILISTIC ASSESSMENT

LOS ANGELES, CALIFORNIA

GENERAL ELECTRIC COMPANY
OCTOBER 18-19, 1984

- CLASSIFICATION OF CORE DAMAGE SEQUENCES
- CORE DAMAGE AND CONTAINMENT RESPONSE
- CONSOLIDATION OF CONTAINMENT RELEASE SEQUENCES
- CALCULATION OF FISSION PRODUCT RELEASE
- CONSEQUENCE ANALYSIS
- RISK RESULTS

CRITERIA FOR CLASSIFICATION OF CORE DAMAGE SEQUENCES

- THE TIME OF LOSS OF CONTAINMENT INTEGRITY RELATIVE TO THE TIME OF CORE DAMAGE
    - 5 CLASSES, CORE DAMAGE LEADS TO LOSS OF CONTAINMENT (ODD ROMAN NUMERAL)
    - 5 CLASSES, LOSS OF CONTAINMENT MAY LEAD TO CORE DAMAGE (EVEN ROMAN NUMERAL)
- PIPE BREAK SIZE OR TYPE OF TRANSIENT
    - CORE DAMAGE PHENOMENOLOGY (E.G., TIMING, HYDROGEN GENERATION RATE)
    - FISSION PRODUCT RELEASE PATH (E.G., VENTS OR SRV's)
- DURATION OF TIME FROM ACCIDENT INITIATION TO LOSS OF CONTAINMENT INTEGRITY OR CORE DAMAGE
    - CORE DAMAGE PHENOMENOLOGY (TIMING EITHER FAST OR SLOW)

LIST OF ACCIDENT CLASSES

Class	Initiating Event	Accident Sequence	RPV Release Path	Class Unique Characteristics
I_T	All transients (represented by loss of offsite Power event)	Core damage then loss of containment integrity	S/RVs	Loss of all core cooling
$I_{SB/IB}$	Small break (SB) or Intermediate break (IB) LOCA in drywell	Core damage then loss of containment integrity	S/RVs and Vents	Loss of core cooling (except one CRD), flow through SRV's and vents
I_{LB}	Large break (LB) LOCA in Drywell	Core damage then loss of containment integrity	Vents	Releases directly to drywell, flow only through vents
II_T	Isolation transient w/ loss of heat removal but w/adequate core cooling	Loss of containment integrity then core damage	S/RVs	Loss of heat removal may lead to core damage
II_A	No scram, followed by SLC injection w/loss of heat removal, but adequate core cooling	Loss of containment integrity then core damage	S/RVs	Faster containment Pressurization than w/Class II_T or II_L
II_L	Drywell LOCA w/loss of heat removal but w/ adequate core cooling	Loss of containment integrity then core damage	Vents	Releases directly to drywell, flow only through vents
III (ATWS)	For all initiators: No scram and no core cooling but w/SLC injection	Core damage then loss of containment integrity	S/RVs	No core cooling (except CRD). Slower effect of negative activity than w/Class I.
IV (ATWS)	For all initiators: No scram and no SLC injection, but w/ adequate core cooling	Loss of containment integrity then core damage	S/RVs	Blowdown @ ~15% power leads to earlier loss of containment integrity than II_A
V	LOCA outside the primary containment	Core damage then loss of containment integrity	Directed to secondary containment	No containment isolation
VI	LOCA outside the drywell but inside the primary containment	Loss of containment integrity then core damage	Directed to primary containment	Inadequate containment vacuum breakers or spray

1 - CRITERIA 1 LOSS OF CORE OR CONTAINMENT FIRST
2 - CRITERIA 2 BREAK OR TRANSIENT
3 - CRITERIA 3 SPEED OF EVENT PROGRESSION

GENERAL ELECTRIC PROPRIETARY INFORMATION

SEVERE ACCIDENT COMPUTER CODES

- TRANSIENT CODES QUALIFIED WITH TEST DATA
  - SAFE (VESSEL INVENTORY & SYSTEMS)
  - REDY (TRANSIENT NEUTRONICS-HYDRAULICS, POINT)
- PRA CODES ADAPTED FOR GESSAR PRA
  - MARCH-CORRAL (UPDATED CONTEMPT-CORRAL)
  - CRAC (OFFSITE CONSEQUENCES)
- OTHER PHENOMENOLOGY CODES
  - POOL DECONTAMINATION FACTORS
  - CONTAINMENT RESPONSE TO HYDROGEN GENERATION AND COMBUSTION
  - CONTAINMENT STRUCTURAL ANALYSIS

SEQUENCE OF CORE DAMAGE

	TIME
O INITIATING EVENT/CORE UNCOVERY	
o Turbine Trip, MSIV Closure	0.0
o Water Level Diminishes	
o Steam Cooling	
O CORE DAMAGE	
o Fuel Pins Overheat	45 Min.
o Onset of Metal-Water Reaction	
o Core Melt and Significant Fission Product Release	50 Min.
o Possible Containment Failure by Hydrogen Combustion	
O RPV MELT-THRU	
o Melten Corium Attacks Instrument Tube or Guide Tube	100 Min.
o Corium Pours Thru Penetration Into Pedestal Cavity	
O CORE-CONCRETE REACTION BEGINS	
o Molten Corium Ablates Concrete	~100 Min.
o Production of Non-Condensable Gases From Concrete	
o Additional Fission Product Release as Gases Sparge Thru Corium	
o Containment Failure by Non-Condensable Gas Overpressurization	
O EVENT TERMINATED	
o Pedestal Cavity Flooded	24 Hrs.

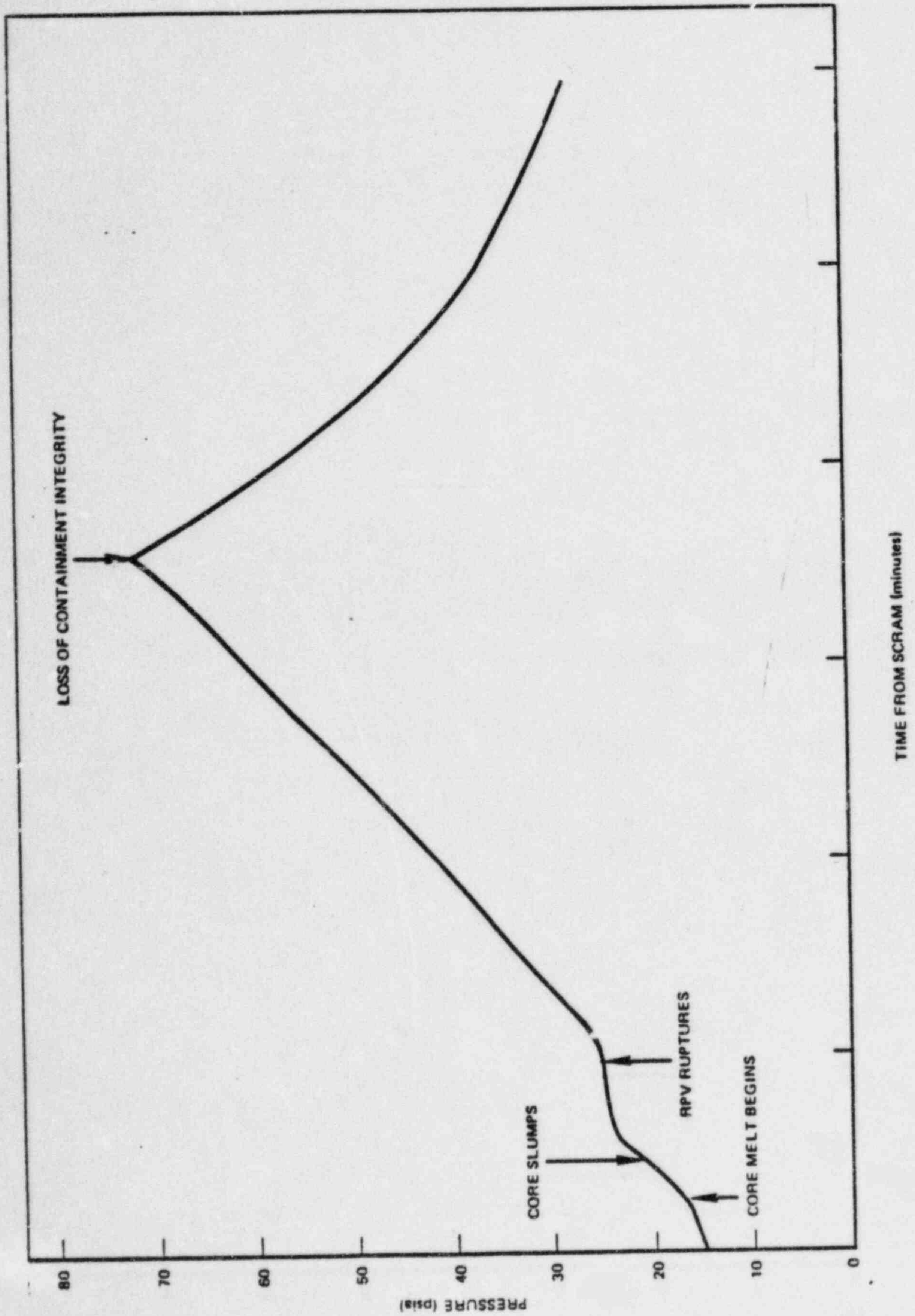
MARCH MODEL FOR BWR APPLICATION

0 INTIALIZATION OF MARCH ANALYSIS

- WATER LEVEL MODIFIED - CONSISTENT WITH GE SAFE CODE
- ATWS POWER LEVEL DETERMINED BY HPCS/RCIC FLOW RATES, GE REDY CODE
- DECAY POWER MODIFIED, INCLUDE HEAVY ELEMENTS, 1979 ANS DECAY POWER

0 ADJUSTMENTS TO MARCH INPUTS

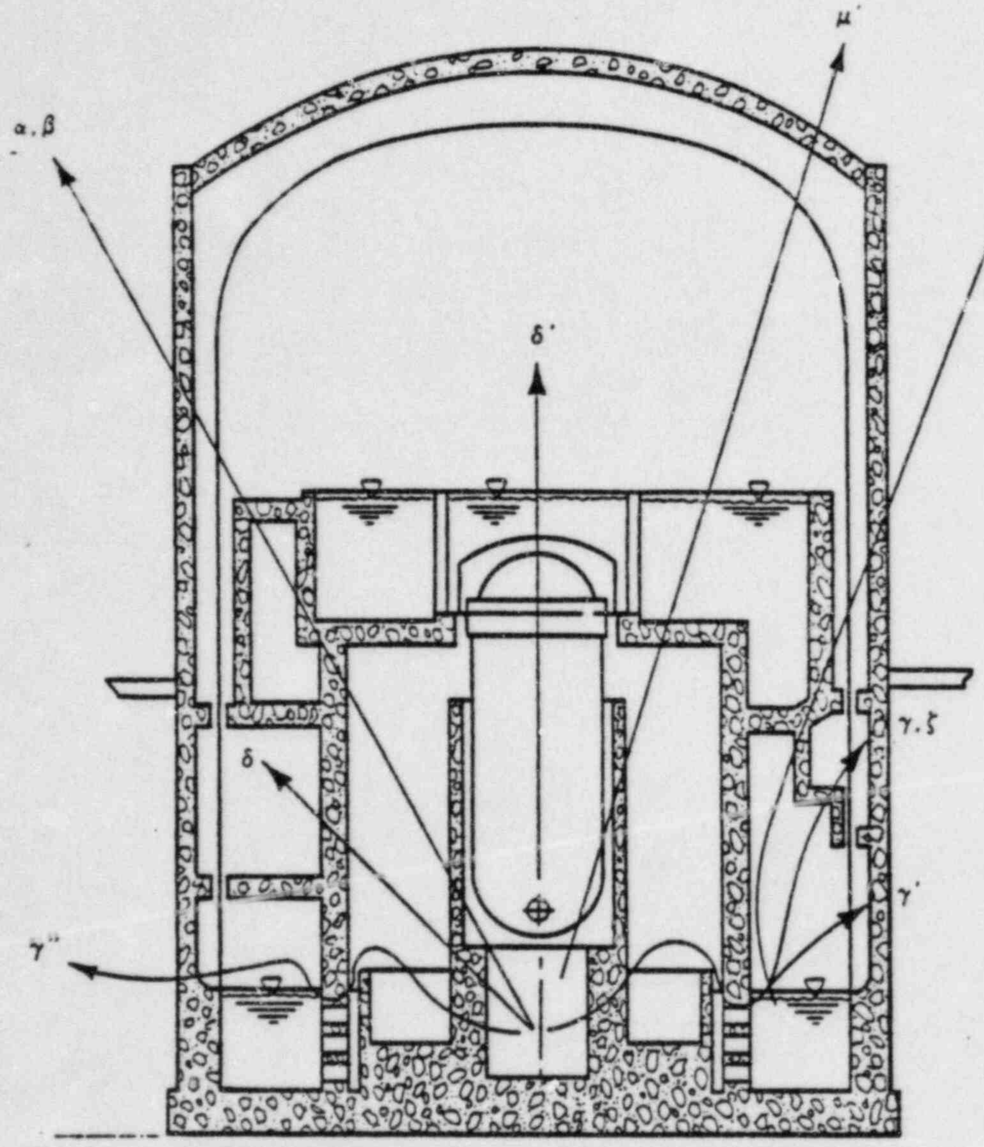
- SRV SIMULATION
- EPG CONTROLLED DEPRESSURIZATION
- SUPPRESSION POOL HEAT SINK
- FUEL CHANNELS INCLUDED
- CORE MELT/SLUMP PATTERN



Containment Pressure Time History for Class I Transient (Class I_T)

REFERENCE MARK III CONTAINMENT

- DESIGN
  - Concrete Drywell
  - Free Standing Steel Containment
  - Concrete Shield Building
- FAILURE MODES
  - Overpressure Failures (Static)
    - Hydrogen Combustion (Loss of Core Cooling)
    - Steam Generation (Loss of Heat Removal)
    - Non-Condensible Gases (Core-Concrete)
  - Dynamic Failures
    - Hydrogen Detonation
    - Steam Explosions - Mechanistically Precluded
- FISSION PRODUCT RELEASE PATHWAYS DETERMINATION
  - Accident Sequence (e.g., Containment LOCA)
  - Containment Failure Modes
  - Equipment Failures (e.g., Vacuum Breaker Failure)
  - Shield Building/SGTS Capability



α	IN-VESSEL STEAM EXPLOSION	δ	CONTINUOUS BURN CAUSES LOSS OF DRYWELL INTEGRITY
β	STEAM EXPLOSION IN CONTAINMENT	δ'	CONTINUOUS BURN CAUSES RPV PIPING DAMAGE
γ	STATIC OVERPRESSURE CAUSES LOSS OF CONTAINMENT INTEGRITY	μ	LOCAL DETONATION CAUSES LOSS OF CONTAINMENT INTEGRITY
γ'	CONTINUOUS BURN CAUSES LOSS OF CONTAINMENT INTEGRITY	μ'	GLOBAL DETONATION CAUSES LOSS OF DRYWELL AND CONTAINMENT INTEGRITY
γ''	GLOBAL COMBUSTION CAUSES LOSS OF CONTAINMENT INTEGRITY	ζ	CONTAINMENT LEAKAGE

FIGURE 2-3. POTENTIAL BWR/6 MARK III CONTAINMENT RELEASE PATHS

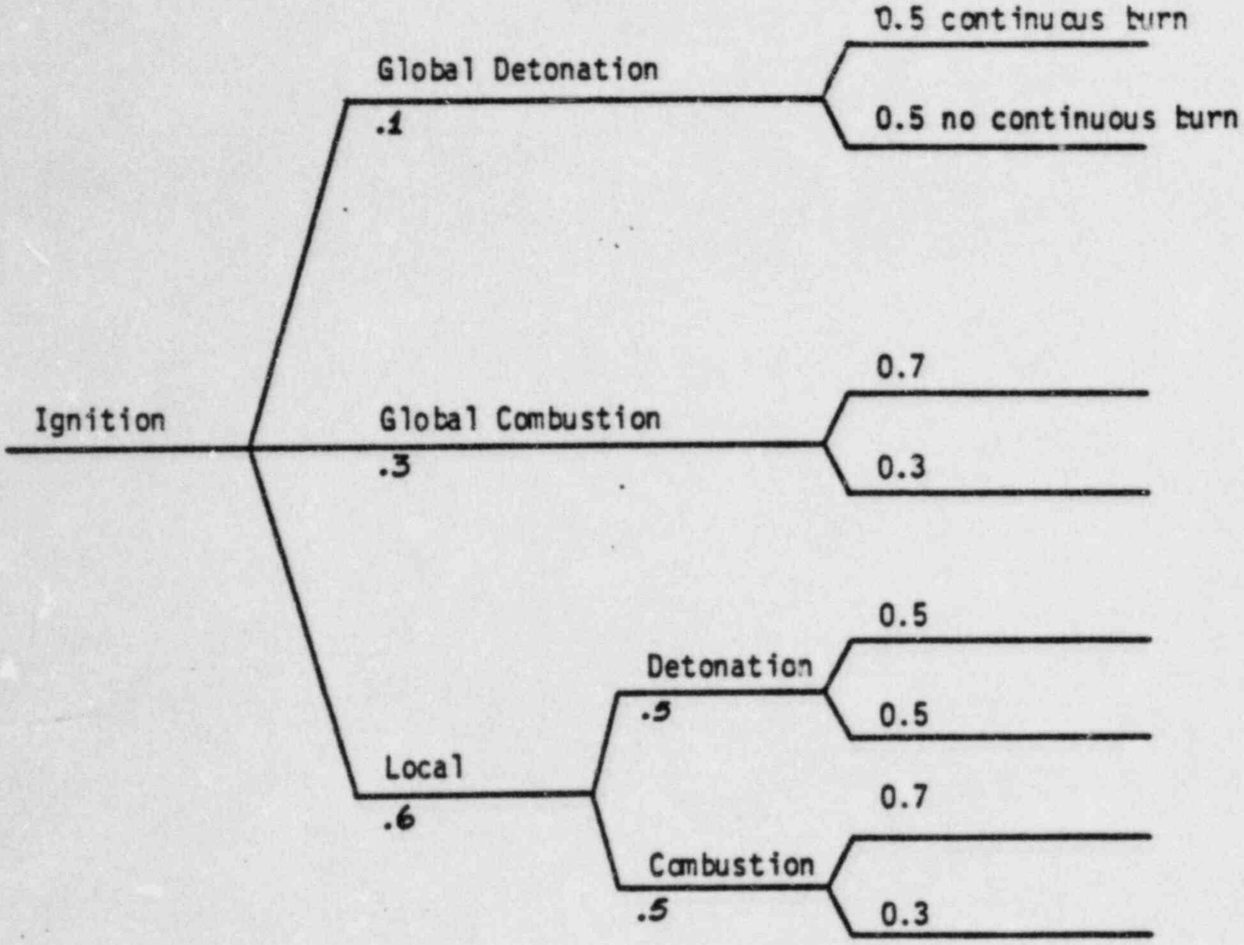
EVALUATION OF RISKS DUE TO HYDROGEN COMBUSTION

- INPUT DATA REQUIRED
  - Rate and Total Hydrogen Release
  - Location and Duration of Release
  - Mixing Assumed (Local and Global)

- EVALUATION OF POTENTIAL SOURCES OF IGNITION OF MIXTURE
  - AC Power
  - DC Power
  - Auto-Ignition

- EVALUATION OF RELATIVE PROBABILITIES OF COMBUSTION OR DETONATION DEPENDS ON
  - Hydrogen Concentration
  - Extent of Combustibility
  - Amount and Nature of Diluents, Ignition Sources Available

TIME: 0<t<1hour



Continuous burn multipl[illegible]
(release time factor) .5

PROBABILITY OF:

ignition = .95
global detonation = .1
global combustion = .3
local detonation = .3
local combustion = .3
continuous burn = .31

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

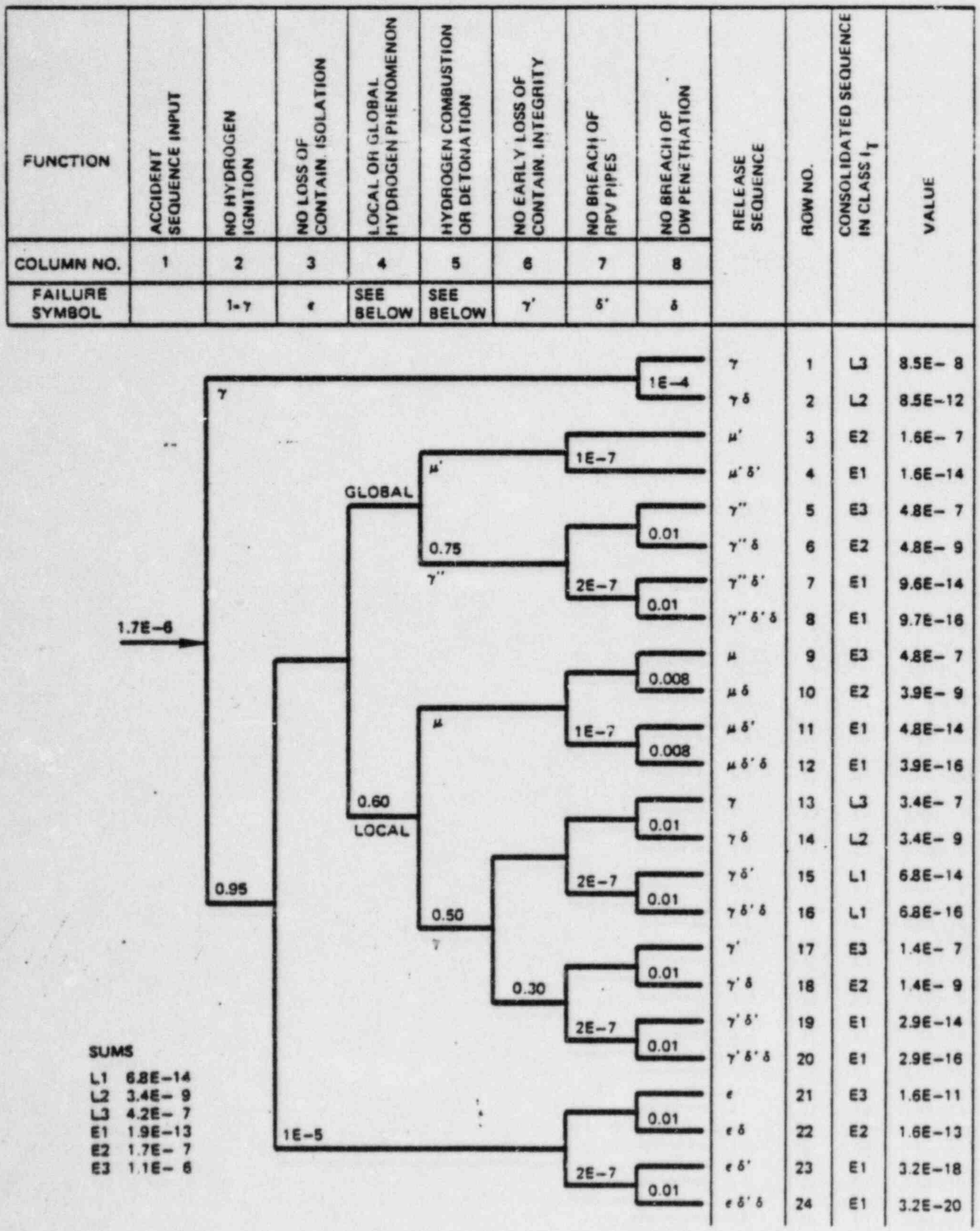
STEAM EXPLOSIONS

- DEFINITION: VIOLENT MIXING OF MOLTEN CORIUM AND WATER THAT RESULTS IN LOSS OF VESSEL OR CONTAINMENT INTEGRITY

- BWR FEATURES PRECLUDE STEAM EXPLOSIONS
  - Distributed Core Support From Below (No Core Plate Collapse)
  - Forest of Guide Tubes (Inhibits Mixing)
  - Structures Above Core (Disperse Material, Attenuate Energy)

- ANALYTICAL EVALUATION
  - Energy Required >500 MJ
  - Melt Mass Required to Mix Intimately 36,000 kgms
  - Resulting Coarse Fragmentation Size >3 Times Greater Than Space Between Guide Tubes
  - Required Fine Fragmentation Mixing Energy Greater Than Available Thermal Energy

- EXPERIMENTAL DATA EVALUATION
  - Analysis Verified Against Sandia Data
  - Artificial Trigger Required in Sandia Corium Experiments
  - Application to Reactor Conditions Shows No Loss of RPV or Containment



RELEASE SEQUENCE	ROW NO.	CONSOLIDATED SEQUENCE IN CLASS I_T	VALUE
γ	1	L3	8.5E- 8
$\gamma\delta$	2	L2	8.5E-12
μ'	3	E2	1.6E- 7
$\mu'\delta'$	4	E1	1.6E-14
γ''	5	E3	4.8E- 7
$\gamma''\delta$	6	E2	4.8E- 9
$\gamma''\delta'$	7	E1	9.6E-14
$\gamma''\delta'\delta$	8	E1	9.7E-16
μ	9	E3	4.8E- 7
$\mu\delta$	10	E2	3.9E- 9
$\mu\delta'$	11	E1	4.8E-14
$\mu\delta'\delta$	12	E1	3.9E-16
γ	13	L3	3.4E- 7
$\gamma\delta$	14	L2	3.4E- 9
$\gamma\delta'$	15	L1	6.8E-14
$\gamma\delta'\delta$	16	L1	6.8E-16
γ'	17	E3	1.4E- 7
$\gamma'\delta$	18	E2	1.4E- 9
$\gamma'\delta'$	19	E1	2.9E-14
$\gamma'\delta'\delta$	20	E1	2.9E-16
ϵ	21	E3	1.6E-11
$\epsilon\delta$	22	E2	1.6E-13
$\epsilon\delta'$	23	E1	3.2E-18
$\epsilon\delta'\delta$	24	E1	3.2E-20

SUMS

L1	6.8E-14
L2	3.4E- 9
L3	4.2E- 7
E1	1.9E-13
E2	1.7E- 7
E3	1.1E- 6

CT1-Pa: Loss of Offset Power (LOOP)
$\leq$ 60 Minutes*

BWR/6 MARK III CONTAINMENT RELEASE CATEGORIES FOR CONSOLIDATED RELEASE SEQUENCES

Class I_T, For Containment Event Tree: CT1-P_b

Relative Degree of Release Scrubbing	Timing of Containment Release		
	E **EARLY** @ Core Damage Stage	I **INTERIM** @ RPV Melt-through Stage	L **LATE** @ Loss of Containment Integrity Stage
1. Some Suppression Pool Scrubbing of Some of the Releases			
2. Most Scrubbing of All Releases until RPV meltthrough some scrubbing afterward	$\varepsilon\delta$ $\varepsilon\delta'$ $\varepsilon\delta'\delta$	μ' $\mu'\delta'$ $\gamma''\delta$ $\gamma''\delta'$ $\gamma''\delta'\delta$ $\mu\delta$ $\mu\delta'$ $\mu\delta'\delta$ $\gamma'\delta$ $\gamma'\delta'$ $\gamma'\delta'\delta$	$\gamma\delta$ $\gamma\delta'$ $\gamma\delta'\delta$
3. All Continuous Scrubbing of All Releases	ε	γ'' μ γ'	γ

ASSESSED FREQUENCY OF RELEASE CATEGORIES

#	Release Category	Class	Cause for Dominant Release Sequence	Frequency (Event Per Reactor Year)
1	I-T-L3	I-Transients	Slow Pressurization	6.9×10^{-7}
2	I-T-E2	I-Transients	H_2 Global Detonation	2.1×10^{-7}
3	I-T-E3	I-Transients	H_2 Global Combustion	1.3×10^{-6}
4	I-T-I2	I-Transients	H_2 Global Detonation	7.6×10^{-7}
5	I-T-I3	I-Transients	H_2 Global Combustion	1.7×10^{-6}
6	I-T-L2	I-Transients	Slow Pressurization with local combustion	4×10^{-9}
7	I-SB-L3	I-SB LOCA	Slow Depressurization	4×10^{-10}
8	I-LB-L3	I-LB LOCA	Slow Depressurization	2×10^{-10}
9	I-S/LB-E1	I-SB/LB LOCA	H_2 Global Detonation	7×10^{-10}
10	I-S/LB-E3	I-SB/LB LOCA	H_2 Global Combustion	1×10^{-9}
11	I-S/LB-L1	I-SB/LB LOCA	Slow pressurization with local combustion	5×10^{-12}
12	II-T-B3	II-Transients	Loss of containment integrity and core cooling leads to Core Damage	2×10^{-8}
13	II-L-B3	II-LOCA	↓	3×10^{-10}
14	II-A-B3	II-ATWS	↓	2×10^{-11}
15	IV-F3	IV-ATWS	↓	5×10^{-8}
			TOTAL	4.7×10^{-6}

CALCULATION OF FISSION PRODUCT RELEASE

- RELEASE FROM FUEL

- PASSIVE FISSION PRODUCT RETENTION MECHANISMS - ALWAYS AVAILABLE

  - AGGLOMERATION
  - GRAVITY SETTLING
  - CONDENSATION/PLATEOUT
  - POOL SCRUBBING

- ENGINEERED SAFETY FEATURES (ESFs) - ACTIVE SYSTEMS

  - AVAILABILITY TREATED PROBABILISTICALLY
  - ISOLATION SYSTEM
  - CONTAINMENT SPRAYS
  - STANDBY GAS TREATMENT

- DECONTAMINATION FACTORS (DF)

- CORRAL CODE

  - SEVERAL COMPARTMENTS (E.G., VESSEL, DRYWELL)
  - NATURAL MECHANISMS (EXCEPT AGGLOMERATION)
  - ESFs (SOME MODELED, OTHERS INPUT AS DF)
  - OUTPUT RELEASE FRACTIONS

- FISSION PRODUCT RELEASE FRACTIONS (RF)

  - INPUT TO OFFSITE DOSE

  - $DF_{OVERALL} \cong DF_{\substack{NATURAL \\ MECHANISMS}} \times DF_{POOL} \times DF_{ESF}$

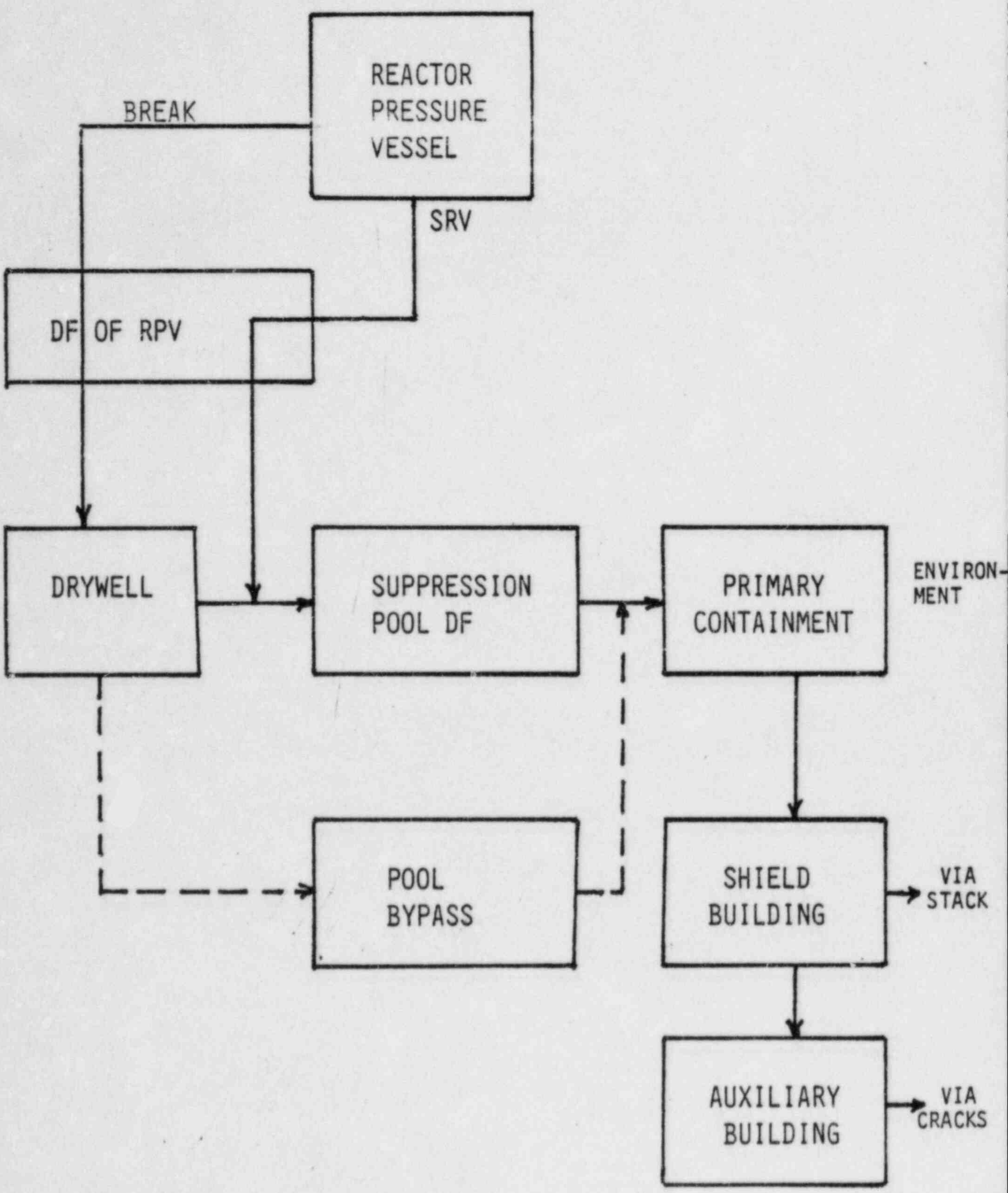
  - $RF \cong \frac{\text{CURIES RELEASED FROM FUEL}}{DF_{OVERALL}} = \text{RELEASE TO ENVIRONMENT}$

SOURCE TERM (FUEL RELEASES) FOR CORE DAMAGE EVENTS

- FUEL RELEASES CALCULATED FROM ORNL MODEL (NUREG-0772, APPENDIX B)

- FISSION PRODUCT RELEASES FUNCTION OF CORE TEMPERATURE AND TIME

- CORE TEMPERATURE PROFILE FROM MARCH CALCULATION



BWR MARK III CORRAL CODE

DF CALCULATION FOR LEAKAGE PATHS

- NO DF FOR PRIMARY CONTAINMENT OR SHIELD BUILDING CRACKS
- NO DF FOR DETONATION CAUSED DRYWELL CRACKS
- USED CORRAL ANNULUS MODEL FOR LPCI GUARD PIPE
- MOREWITZ PLUGGING MODEL FOR NORMAL LEAKAGE PATHS
  - NUREG-0772 P 7.35
  - Empirical Models Developed From Tests in Wet and Dry Environments
- PROCEDURE
  - Calculate Aerosol Loading in RPV and Drywell
  - Amount Leaked Prior to Plugging (M)

$M = KD^3$ Where

D = Crack Diameter
$K = 50\ \text{g/cm}^3$

- EXAMPLE

0.25 Inch Instrument Line From RPV
D = 0.64 CM
RPV Aerosol Source Term - 102 KGM
M (GMS) = 12.8 GMS

$$DF = \frac{102{,}000\ \text{GMS}}{12.8\ \text{GMS}} = 8{,}000$$

- NO CREDIT TAKEN FOR LARGER CRACKS
- REALISTIC CREDIT FOR SMALL CRACKS

SUMMARY OF DECONTAMINATION FACTORS FOR BWR/6 PROBABILISTIC RISK ASSESSMENT

LOCATION	PARTICULATES	ORGANIC IODIDE	NOBLE GASES
PRIMARY SYSTEM BEFORE CORE SLUMP			
SMALL BREAK OR TRANSIENTS	10.	1.0	1.0
OTHERS	1.0	1.0	1.0
RPV AFTER CORE SLUMP	1 - 20	1.0	1.0
SUPPRESSION POOL			
SUBCOOLED	600 - 10,000	1.0	1.0
SATURATED	600 - 10,000	1.0	1.0
SGTS	1.0	1.0	1.0
CONTAINMENT SPRAY	VARIABLE (C)	1.0	1.0
DRYWELL (B) CRACKS	VARIABLE (A)	1.0	1.0
DRYWELL/CONTAINMENT	PLATEOUT CALCULATED BY CORRAL	1.0	1.0

NOTE: (A) MAY VARY FROM DF=1 TO INFINITY AND IS A FUNCTION OF BREAK SIZE, GEOMETRY, AND AEROSOL GENERATION RATE

(B) FROM PRE-EXISTING DRYWELL PENETRATIONS

(C) ACCIDENT SEQUENCE DEPENDENT CALCULATED BY CORRAL CODE

GENERAL ELECTRIC PROPRIETARY INFORMATION

CONSEQUENCE ANALYSIS METHODOLOGY

- CRAC CODE ADAPTED FROM 1977 NRC VERSION OF CRAC USED FOR WASH-1400

- CODE MODIFICATIONS
  - Corrected Height of Plume Release Error
  - Baseline for Chronic Dose Calculations Extended From 10 to 30 Years
  - Corrected Summation of Ingestion Dose Pathways for Chronic Calculations

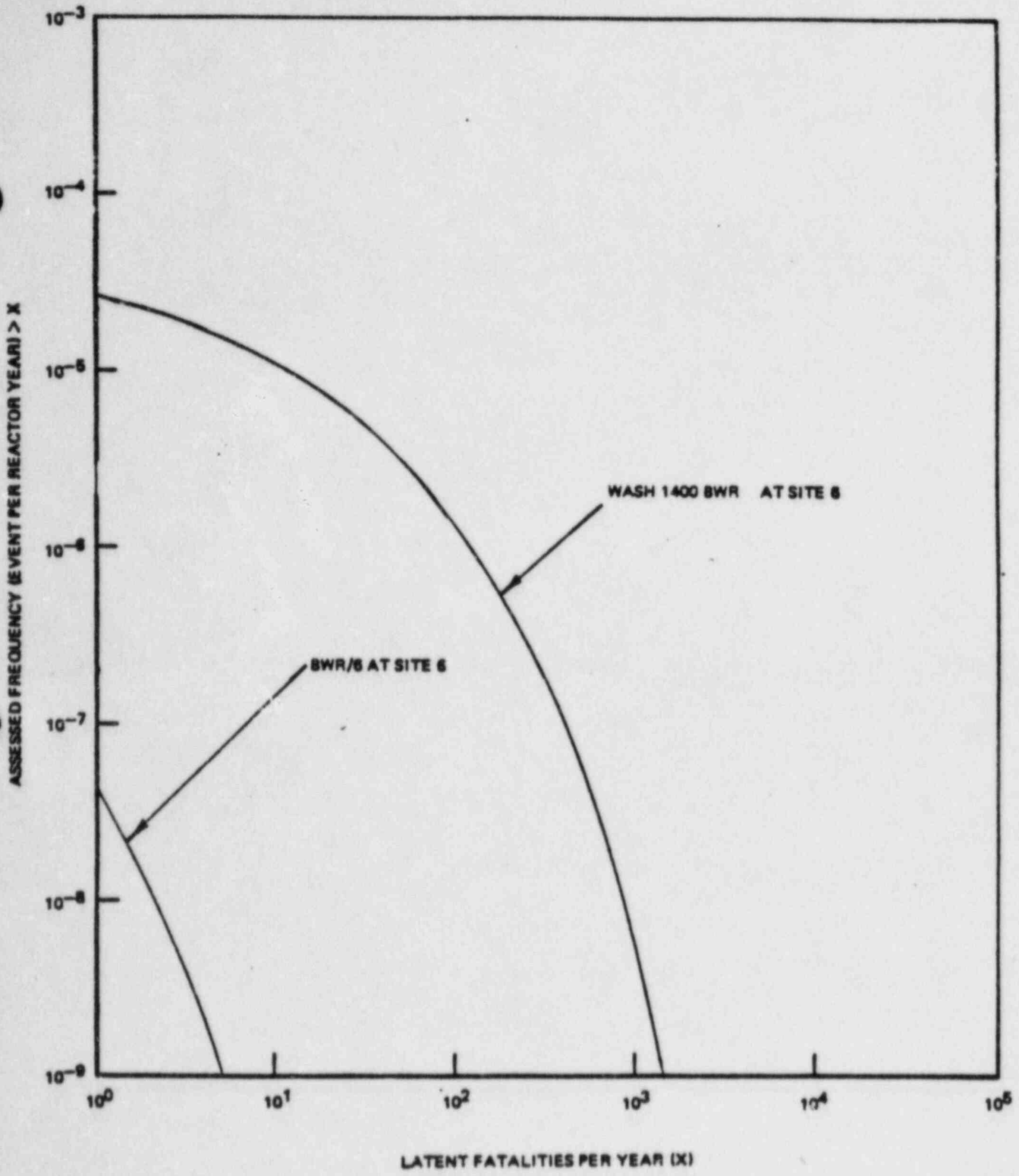
- EFFECT OF MODIFICATIONS ON WASH-1400 RESULTS (WITH WASH-1400 INPUTS)
  - Reduce Acute Fatalities by 5%
  - Overall Reduction of 10-20% in Chronic Doses

DOSE MODEL

- THRESHOLD FOR EARLY FATALITIES
  - 320 Rem to Bone Marrow
  - Same as WASH-1400
- INITIALLY LINEAR CHRONIC DOSE MODEL WAS USED FOR LATENT CANCERS
  - Central Estimate Used in WASH-1400
  - Linear Factor of 2 Times Central Estimate
- FINAL RESULTS CALCULATED USING CENTRAL ESTIMATE DOSE MODEL

SITE DESCRIPTION

- SITE SELECTION PROCEDURE
  - WASH-1400 BWR Accident Releases Run at All Six Sites
  - Results Compared to Composite Site
  - Site 6 Results Closest to Composite Results
- SITE 6 DESCRIPTION
  - Atlantic Coastal Site
  - Average Rainfall and Weather Stabilities
  - Population
    - Composite of All Atlantic Coastal Site Populations
    - Third Largest in Overall Population (81 Million Within 500 Miles)
  - Average Grid Reliability
- COMPARISON TO WASH-1400 COMPOSITE SITE
  - No Data For Duplication of WASH-1400 Curve
  - GE CRAC Code Different Than WASH-1400 Version
  - Site 6 Curve Most Representative of Composite Site Curve



Comparison of Risk for the
WASH-1400 BWR and BWR/6

ESTIMATED CORE DAMAGE AND RISK COMPARISON

	Event	Assessed Frequency of Event Per Reactor Year	Risk (Per Year)	
			Early Fatalities	Latent Fatalities[b]
I.	CORE DAMAGE			
	RSS BWR/4 Mark I @ composite site	$\sim 4\times10^{-5}$ $\sim 4\times10^{-5}$	$\sim 1\times10^{-5}$[a] 2.4×10^{-5}[c]	$\sim 5\times10^{-2}$[a] 2.5×10^{-2}[c]
	RSS BWR/4 Mark I @ site #6[c]	$\sim 4\times10^{-5}$	7.8×10^{-6}	2.1×10^{-2}
	BWR/6 Mark III @ site #6[c]	5×10^{-6}	~ 0	1.7×10^{-5}
II.	U.S. NATURAL BACKGROUND RADIATION	Continuous	0	814

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[a] With WASH-1400 Methods (calculated from the reported curves).

[b] The total accident-caused fatalities over the lifetime of the exposed population or the calculated excess cancers in the same population from one year of background radiation.

[c] Computed with the GE CRAC Code.