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UNITED STATES

NUCLEAR REGULATORY COMMISSION

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MEETING BETWEEN NRC AND MESSRS. LOCHBAUM AND PREVATTE ON EFFECTS OF A LOSS OF SPENT FUEL POOL COOLING AT SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

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Room 2-F-17
Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

Friday, October 1, 1993

The above-entitled meeting commenced, pursuant to notice, at 1:05 p.m.

PARTICIPANTS:

Dick Clark, NRC Projects Manager
Ashok Thadani, NRR/DSSA
Jose Calvo, NRR/DRPE
Conrad McCracken, NRC/SPLB
John White, Region I/DRP
Michael Boyle, NRR/DRPE
Dave Lochbaum, Self Representation
Don Prevatte, Self Representation
George Hubbard, DSSA/SPLB
John Kopeck, NRC Public Affairs
David Shum, NRR/DSSA/SPLB
Michael Blood, AP
Paul Gunter, Nuclear Information and Resource Service
Jon Block, New England Coalition on Nuclear Pollution
Tilda Liu, NRR/PDI-2
Lisa Finnegan, Allentown Morning Call
Joanne Royce, Government Accountability Project
James Kenny, Pennsylvania Power & Light
Herb Woodeshick, Pennsylvania Power & Light
David Ney, Pennsylvania Department of Environmental Resources

PARTICIPANTS [continued]:

Larry Nicholson, NRC Region I
Ken Eccleston, NRR/DRSS/PRPB
Morton Fleishman, NRC/OCMRR
Mark Wigfield, O'Hawy News Service

PROCEEDINGS

[1:05 p.m.]

MR. CLARK: Good afternoon. My name is Dick Clark. I'm the NRC's Office of Nuclear Regulation Project Manager for the Susquehanna Steam Electric Station, which is operated by Pennsylvania Power & Light.

On November 27, 1992, Mr. David A Lochbaum and Mr. Donald C. Prevatte filed a Part 21 report with the NRC asserting that the design of the spent fuel pool cooling system at Susquehanna fails to meet regulatory requirements. The initial Part 21 report has been supplemented by six additional letters.

The purpose of this meeting is to afford Mr. Lochbaum and Mr. Prevatte the opportunity to personally present the technical issues in their Part 21 report to the NRC staff.

Don and Dave, we would really like to welcome you here today. We look forward to hearing from you.

Just for the record, we have previously held two meetings with Pennsylvania Power & Light on this issue, the first of which was attended by the two individuals we are meeting with this afternoon. We have also had the benefit of a number of written submittals by the licensee.

There is an attendance list circulating and when everyone has signed it, we will get copies made for each of you. I would also like to note that this meeting is being transcribed and a copy of the transcript will be available in about two weeks.

Before we start this meeting, I would like to request that we go around the room and have everyone introduce themselves and identify what group they are with. When the introductions are completed, Mr. Thadani, here to my right, our Director of System Safety and Analysis, will make some opening remarks before we turn the meeting over to you, Don and Dave.

At the end of this afternoon's meeting, there will be an opportunity for any observers to offer comments or statements on the issues.

MR. THADANI: I'm Ashok Thadani, the Director of the Division of System Safety and Analysis, NRR.

MR. CALVO: Jose Calvo. I'm the Assistant Director for the Division of Projects in Region I.

MR. McCRACKEN: Conrad McCracken, Chief of the Plant Systems Branch.

MR. WHITE: John White, Region I, Section Chief of the Division of Reactor Projects.

MR. BOYLE: I'm Michael Boyle. I'm the acting Project Director, NRR.

MR. LOCHBAUM: I'm Dave Lochbaum, co-wrote the 10 CFR Part 21 report.

MR. PREVATTE: I'm Don Prevatte. I'm the other person who wrote the 10 CFR 21 report.

MR. HUBBARD: George Hubbard with the Plant Systems Branch.

MR. KOPECK: John Kopeck, NRC Public Affairs.

MR. SHUM: David Shum, Plant Systems Branch.

MR. BLOOD: I'm Michael Blood with the AP.

MR. GUNTER: Paul Gunter, Nuclear Information and Resource Service.

MR. BLOCK: Jon Block, New England Coalition on Nuclear Pollution.

MS. LIU: Tilda Liu, Project Director for NRR.

MS. FINNEGAN: Lisa Finnegan with the Allentown Morning Call.

MS. ROYCE: Joanne Royce, Government Accountability Project.

MR. KENNY: Jim Kenny, Pennsylvania Power & Light Company.

MR. WOODESHICK: Herb Woodeshick, Pennsylvania Power & Light Company.

MR. NEY: Dave Ney. I'm with the Department of Environmental Resources in Pennsylvania.

MR. NICHOLSON: I'm Larry Nicholson with the NRC Region I.

MR. THADANI: First, let me thank you for the effort that you have put in. I know it's been a considerable effort on this issue and to bring this issue to our attention. We do appreciate that and we want to thank you for that.

As Dick mentioned in his opening remarks, we met with Pennsylvania Power & Light and we received from them their assessment of this issue. As you probably know, we're reviewing that information.

We're also very anxious to hear from you. So I look forward to this discussion in the next two hours, to try and make sure we understand your perspective on this issue, the analyses and assessments that you have done. I assure you we are going to take this information that you give us very seriously. We will review it very carefully.

It will be your information and the information we received from Pennsylvania Power & Light and some of the studies that we're doing ourselves that will form the basis of our final conclusion.

It would be very helpful, if it is possible, as you go through your discussion and presentation, if you can kind of give us your perspective on the safety significance of the issue. After all, I think that should be what should drive us.

But other than that, we're very anxious to hear from you.

MR. PREVATTE: Thank you very much. We appreciate everyone being here. We appreciate the introduction. As was mentioned, the purpose of this meeting is for Dave and myself to make a presentation to the Nuclear Regulatory Commission.

There were two primary issues that we originally intended to address. One was the technical issue with regard to the spent fuel pool at Susquehanna. The second was Pennsylvania Power & Light's handling of this issue when we were contract employees.

We have agreed with the NRC. As of this week, we were requested not to present the second issue in this public forum and we have agreed to that, under the condition that at some point in the future, when they've had additional time to consider it, that it would also be presented in a public forum.

Our technical concerns are as follows. If you would like to follow along in the handout, it's the page after the cover letter. Susquehanna Steam Electric Station has serious design defects for handling of loss of fuel pool cooling events associated with design basis loss of coolant accidents.

For design basis loss of coolant accident conditions, defects in the plant design could produce potentially catastrophic results, as follows; fuel meltdown outside containment, failure of all reactor building safety-related systems and result in core meltdown, failure of all containment systems in the plant, and result in off-site radiation doses thousands of times greater than the 10 CFR 100 limits.

We consider this to be a very high risk accident. You asked that we address the safety significance. Risk in the nuclear power business is defined as a product of probability and consequences. In this particular accident, the probability is the same as the design basis LOCA, ten-to-the-minus-six per year.

The reason it's the same is because it is mechanistically caused by LOCA. The failure of the fuel pool cooling system is caused by the LOCA. The consequences of this accident are much worse than a LOCA, as described above. You have failure of containment. You have an off-site dose significantly higher than what 10 CFR 100 allows for anything that was ever analyzed. The product of those two things gives you a risk that's much, much higher.

Because the reactor building is inaccessible post-LOCA, the operators can't get inside to intervene and do anything that might prevent this accident from happening. They also can't get to the instrumentation that normally monitors the fuel pool conditions. Those conditions are temperature and level.

That instrumentation is inside the reactor building and they can't get to it. Therefore, they are unable even to monitor what's going on or to do anything about it. Basically, all they can do is stand back and watch it happen.

Most BWRs in this country have the same basic containment design, the BWRs with the Mark I and Mark II containment. That's approximately a third of the plants in the United States. We are concerned that this particular accident scenario that we've discovered is likely to be common to all of those plants.

At this point, Dave will give you an outline of how we believe the accident scenario would progress.

MR. LOCHBAUM: The primary area of dispute we had with PP&L on this issue was whether or not the DBA LOCA, with a boiling spent fuel pool, was within the design basis of the Susquehanna plant. I think we can show that it is.

The licensing basis is defined as all accidents and events described in the FSAR, plus all of the mechanistic results that occur because of those accidents and events. The design basis LOCA is one of the accidents described in the Susquehanna FSAR.

The spent fuel pool cooling system at Susquehanna is not designed for post-LOCA hydrodynamic and environmental conditions and will fail mechanistically as a result of the accident.

To go over briefly what hydrodynamic loads are, this is a typical Mark II containment. It's not necessarily Susquehanna's, but it's similar. In a design basis LOCA, you have a failure of some piping, generally the recirc piping, inside the drywell. Following that failure, the energy, the steam that is released into the drywell, pushes the nitrogen atmosphere of the drywell down through the vent piping, into the suppression pool.

This atmosphere exits below water level and forms essentially a huge bubble that lifts up the water in the suppression pool. This is non-condensible. So eventually it makes its way through the water. The water then falls back down to the containment and gives the entire building a fairly good shake, on the same magnitude as a seismic event. So, essentially, the entire building sees the equivalent of an earthquake-type loading.

Because the fuel pool cooling system piping and the associated service water system piping is not designed for these loadings, they probably -- they may or probably will not survive that event.

In any event, the fuel pool cooling system pumps are not Class 1E power. They will automatically load-shut on a LOCA signal. So the system is designed to shut down fuel pool cooling in an accident and it does.

Once the fuel pool cooling system fails, for the Susquehanna FSAR, the only design consequence is the spent fuel pool blown. There's an RHR fuel pool cooling assist mode in the plant, but its only design function is during a refueling outage. It has no accident role.

We'll also be able to show you why they couldn't use it even though it's installed.

MR. THADANI: Can I ask a question?

MR. LOCHBAUM: Sure.

MR. THADANI: Have you done calculations of the dynamic loads that would be generated due to these non-condensibles?

MR. LOCHBAUM: No.

MR. THADANI: I agree with you that you will get -- clearly, you will get swell and the drop and you'll get some loads, but I was curious if you had done any calculations.

MR. LOCHBAUM: PP&L has looked at those and they said that there's a moderate risk to the service water system piping and the hangers in the fuel pool cooling system are not designed for those loads.

MR. PREVATTE: There have been extensive studies done on that. Craftwork Union did an extensive study for PP&L on these hydrodynamic loads. The results of those studies showed that they are real and they are very significant. They're on the same order of magnitude as the seismic loads which the plant has to be designed for.

Therefore, if this piping is not designed for that load -- this system -- and it is not. All the safety-related systems in the building are designed for these loads. This system is not. Therefore, it cannot be assumed to take that pressure.

MR. THADANI: I understand the design basis. My question was more in the sense of when something is designed for it or something is expected to fail are two different things. So I was curious if you had done those calculations.

MR. LOCHBAUM: PP&L did analyze the result in this issue and they're indicating it as a moderate risk to service water system piping.

Once the spent fuel pool cooling system fails --

MR. CALVO: Excuse me. Those loads impact the service water system pipe. That's the water that you're taking out of the --

MR. LOCHBAUM: Fuel pool cooling system heat exchangers. That's what cools those heat exchangers.

MR. PREVATTE: And that's a non-safety-related system.

MR. CALVO: Can you put the picture back again?

MR. LOCHBAUM: Sure. This one might be a little bit better for that. The fuel pool cooling system heat exchangers are located here. The service water system enters and leaves to remove the heat from the fuel pool cooling system heat exchangers.

This piping is generally routed through the lower portions of the reactor building. That's why it's going to see higher loadings than the fuel pool cooling system piping, which is located at the upper levels of the reactor building.

MR. CALVO: I see.

MR. LOCHBAUM: Once the fuel pool cooling system fails, the only design consequence is a boiling spent fuel pool.

MR. PREVATTE: One of the points that Dave made also was that another mechanism for failure in the LOCA is environmental conditions. This system is also not designed for the LOCA environmental conditions; that is, temperature and radiation. Post-LOCA, the environment in the reactor building gets pretty nasty. The radiation levels go up to around 5,000 R per hour. In localized spots, it's even higher than that. The typical temperature in the reactor building is about 135.

I believe we can show also that those numbers may be non-conservative. So this equipment is not designed for those conditions and it cannot be assumed to operate post-

LOCA.

What we've shown so far is that the fuel pool cooling system has the potential of failing as a result of the LOCA, nothing else. One of the points that PP&L has made to the NRC, to the press, and to their own people, very emphatically, they have stated that our position was that it required concurrent accidents in order for this to happen. That is absolutely not true. You don't need a LOCA and a LOOP for this to happen. You don't need a LOCA and a single failure. You only need a LOCA. The LOCA itself can cause this to happen.

Now, it is true that the licensing basis for Susquehanna includes these other events, LOCA/LOOP, LOCA/single failure, LOCA with a seismic event/single failure. All these other things are included in the licensing basis. In fact, these other events alone can cause a failure of the fuel pool cooling system.

The point here is that it only takes a LOCA to cause it. These other events are also within the licensing basis. So these other events combined with a LOCA, combined with a boiling spent fuel pool are all within the licensing basis.

It is not specifically addressed in the FSAR in those terms, but the LOCA is described in the FSAR. This is a mechanistic result of the LOCA. Therefore, it is a part of the licensing basis for this plant.

Dave will now continue with the presentation, describing some of the consequences of what happens when you lose fuel pool cooling.

MR. LOCHBAUM: The emergency service water system at Susquehanna is designed to provide makeup water to the boiling spent fuel pool. Basically, we show this on a P&ID. The service water system pumps water from the ultimate heat sink or spray pond through some manual valves into the spent fuel pool in either unit. There are two loops. It's redundancy. It's a safety-related system.

The significance of this is Regulatory Guide 1.13 allows the fuel pool cooling system to be a non-safety-related system, provided there is a safety-related makeup system in the design that's available.

MR. CALVO: The fuel pool cooling pumps. The pipes that you assume that fail are those pipes -- the service water pipes.

MR. LOCHBAUM: This, too. This is not seismic, not hydrodynamic loads. This piping here is not seismic, not designed for hydrodynamic loads. This piping from here to here to the RHR system is seismic, but the rest of this piping, all this --

MR. CALVO: Where is the route that you think that you can put water into the pool?

MR. LOCHBAUM: With this system? It's essentially the same as over here.

MR. CALVO: I must go back to the emergency service water into the RHR, which is also seismically qualified.

MR. LOCHBAUM: That's correct.

MR. CALVO: And they will not be able to -- they won't see those loads that you postulated.

MR. LOCHBAUM: They'll see him. They're just designed to handle them.

MR. CALVO: Designed to handle them.

MR. LOCHBAUM: That's right. For the design basis LOCA, with the required Regulatory Guide 1.3 source terms resulting from the accident, the reactor building at Susquehanna will be inaccessible for many days following the event.

The significance of that is, if you go back to this P&ID, these ESW makeup valves are located in the reactor building. They will not be accessible due to high radiation levels following an accident.

MR. THADANI: Again, did you do calculations using Reg Guide 1.3 assumptions?

MR. LOCHBAUM: PP&L did.

MR. THADANI: And they concluded that -- what would the dose be in that room?

MR. PREVATTE: It's around 5,000 R per hour. The lethal dose is around 450 R.

MR. LOCHBAUM: We'll also address their time-motion study a little bit later in the presentation.

MR. THADANI: Because I had thought they had given us different estimates.

MR. LOCHBAUM: We'll cover that. The conclusion, again, is the ESW makeup valves cannot be opened because of the radiation exposure being too high.

It says makeup water cannot be provided to the pool in the boiling spent fuel pool. Eventually, this pool will become uncovered and the irradiated fuel in the pool will suffer severe damage. This fuel is located outside primary containment. So there's fewer barriers to prevent the release of the radioactivity to the atmosphere, to the environment.

Again, even the consequences of the release of this radioactivity into the environment has not been analyzed. It would be much more severe than what is currently analyzed at Susquehanna.

Don will pick up.

MR. PREVATTE: One of the consequences of this boiling spent fuel pool is that the safety-related equipment in the reactor building is not qualified for the conditions that the boiling spent fuel pool will create.

If we can flip back for just a moment to the diagram of the reactor building. Susquehanna is divided into three ventilation zones. Everything on the refueling floor, which is essentially everything from this level up, is Zone 3. In Unit 1, everything from that level down is Zone 1. In Unit 2, it's called Zone 2.

During a loss of coolant accident under the original design of the plant -- and we'll get into some of the plans that PP&L has for changing that a little later. But under the original design of the plant, in a loss of coolant accident, the unit suffering the loss of coolant accident would automatically -- the ventilation system would automatically cross-connect Zone 3 with the accident unit.

Let's say for a moment that this is Zone 1, an accident unit. Zone 3 is where the fuel pool is. So if the fuel -- and the normal ventilation system during an accident shuts down and your safety-related ventilation systems start.

One of the safety-related ventilation systems is called the recirculation system. Basically, it recirculates all the air in the reactor building in Zone 1 -- in this case, it's a Unit 1 accident -- and Zone 3. If it happens to also be a loss of off-site power, the other unit is also cross-connected. So you have all three zones cross-connected.

But just to keep it simple, let's just talk about, for a moment, just the loss of coolant accident and no loss of off-site power.

Whatever is happening on the refueling floor when you're in a recirculation mode is going to tend to be happening everywhere else in the building, because this recirculation system is designed to keep the atmosphere well mixed and well recirculated.

If you have boiling going on, for the design basis case where you had a full fuel pool, you would have as much as 26 million Btu's per hour of heat would be going into the reactor building atmosphere.

Normally, this heat is going into the spent fuel pool system and out through the service water system to the cooling tower. But when you have a loss of the spent fuel pool cooling system, that heat now goes into boiling water and the heat eventually ends up in the reactor building.

The safety-related equipment in the reactor building has not been designed to handle this heat load. To give you an understanding of what the order of magnitude here is, I did the reactor building heat load calculation for Susquehanna as a part of the Power Uprate Project.

The heat load under the design basis accident conditions for this equipment in this area is 5.2 million Btu's per hour. For this particular condition here where we have a boiling spent fuel pool, we add an additional 26 million Btu's per hour that was never accounted for in the original design.

It doesn't take a rocket scientist to figure out that that's a pretty big change in the heat load. As I said before, the typical qualification temperature for safety-related equipment in the building is around 135 degrees. Bechtel did calculations back in 1979 that showed that for this boiling spent fuel pool case, the temperature was around 180 degrees. That is a significant change in temperature and the equipment is not qualified to take it.

In addition to the heat load that's going into the room or into the building, there's also the makeup water that's going into the building. For the design basis fuel load case, the full fuel pool case, there may be as much as 5 million gallons of water put into the building over the course of a 30-day accident.

That water leaves the fuel pool, much of it does, in the form of vapor, which goes up into the Zone 3 atmosphere, and in overflow. If you don't happen to have the valve set exactly at the boiling rate of the fuel pool, whatever excess there is overflows and goes into the building.

That was never accounted for in the original design. As PP&L has finally indicated in some of their submittals, it produces significant water levels in the basement of the building. However, their submittals -- the volumes that they have estimated are not based on a full fuel pool with all the heat load, the total heat load that would have to be accommodated.

They are also based on assuming that the operator sets the valve precisely at the right position. These valves aren't calibrated. They aren't preset. So the only thing the operator can do is set the valve either wide -- and he doesn't have much time to set the valve, by the way. He's operating in a 5,000 R environment.

MR. LOCHBAUM: Also, these valves have never been tested.

MR. PREVATTE: With this water here, PP&L has already conceded that one of their core spray pumps fails in this accident event and other unspecified safety-related equipment. We don't know what they're conceding on that.

But another thing that must be considered is that none of the HVAC ductwork in the reactor building that this vapor is all going through, none of it is designed to handle water. If you consider what's happening here, if I have a boiling spent fuel pool, the temperature in Zone 3 is going to tend to be the highest temperature location in the building and you're going to have 100 percent humidity.

As soon as the air is drawn into the ductwork and recirculated to the rest of the building, which is relatively cooler, it's going to condense and you're going to have condensing water inside your ductwork. The negative effects of that can be several.

One, they can collect in the ductwork and cause it to fall down. It's not designed to hold up water. It can also block the ductwork, and this is safety-related ductwork. It's supposed to be performing a function at this point in time. If it blocks it, the equipment will not get the cooling it's supposed to get.

It's also going to be pouring out at the places where it's not watertight, and nothing in the building has been analyzed to see what are the effects of water pouring out of this ductwork.

MR. LOCHBAUM: The other safety significance of the water blockage is that the ductwork is also used as the inlet for steamline gas treatment system. That's the only system that provides filtration of radioactivity released to the environment. So that system could potentially fail also due to the blocked ductwork.

MR. PREVATTE: Another thing that has to be considered is that in this building there are a number of safety-related coolers that have to operate in order to provide cooling to the safety-related equipment; the core spray pumps, the RHR pumps, all your safety-related switchgear and load centers.

Those coolers, none of them are designed for latent heat cooling. What that means is if you have a very humid atmosphere, which you will have here if this thing is boiling, you're going to have condensation going on in those coolers. There are a couple of problems with that. One is the coolers don't put out the cooling effect that they're supposed to and the other is there's no provision in the coolers for handling this condensate.

The condensate itself may cause a problem in the location of the coolers with regard to safety-related equipment.

One other point on this. PP&L has made -- I may be getting ahead of myself on this one, with regard to isolation.

MR. LOCHBAUM: We'll get it later.

MR. PREVATTE: We'll get that later. We'll come back to this later with regard to the position PP&L has taken on this.

MR. LOCHBAUM: At this point, we were going to cover some fault trees. I don't have slides made up for those, but they're attached to the handout we provided. The first page is the one right after page 13. It's a fault tree that at the top says "DBA LOCA Event."

I'd like to go through that just to show -- this outlines this accident of concern. The first page essentially describes the Susquehanna design with examining the use of equipment on the accident unit and whether it will be available.

The initiating event is a design basis LOCA. The design basis LOCA results in a direct loss of fuel pool cooling due to the automatic load shed logic. This logic sheds non-Class 1E loads from their power supplies so that the safety-related equipment has sufficient power to come up to speed in the time it needs to. So the design basis LOCA immediately results in a loss of fuel pool cooling.

The first box is is the reactor building accessible. As we stated earlier, if you assume Reg Guide 1.3 --

MR. THADANI: Could I ask you?

MR. LOCHBAUM: Sure.

MR. THADANI: Just so I can understand better. With the automatic load change logic, is that assumed that off-site power is available or not available?

MR. LOCHBAUM: Either way. Either way, the fuel pool cooling pumps will not have any power to them.

MR. CALVO: But if you have the off-site power -- okay. So the capability is with off-site power or without off-site power.

MR. LOCHBAUM: Either way.

MR. CALVO: You're sequencing the load whether you have off-site power. The fact that you got a LOCA signal, you say okay, if I've got off-site power, I'm still going to sequence the load.

MR. LOCHBAUM: That's correct.

MR. CALVO: But if I have off-site power, I may be disconnecting the fuel pool equipment, but I may have the capability later on to connect it if I want because the off-site power is available.

MR. LOCHBAUM: That's correct. We'll go through that, also.

MR. CALVO: All right.

MR. LOCHBAUM: If the reactor building is accessible -- let me step back, first. If you assume the required Reg Guide 1.3 accident source terms, the accident reactor building is not accessible due to high radiation levels. In that case, you'd immediately go to the second page.

We'll assume that the reactor building is accessible and see how far we can get here. The next is is non-Class 1E power; essentially, is off-site power available. If the answer to that question is yes, then you circle over to the right. The first question is the fuel pool cooling system and service water piping intact.

As we stated earlier, the hydrodynamic loading is beyond the design basis for both of this piping. So we cannot assume that it will survive the design basis LOCA.

But if you can't, then you go over to the left and circle down. We'll just take the path assuming that you can. The next question is is the fuel pool cooling system and service water components operable.

As Don discussed earlier, these components will experience very high radiation, very high temperature, and very high humidity environment following an accident that they are not designed for. These systems are not single failure-proof. So the failure of any component can render the system inoperable.

So, again, for the system -- for these components to be available, they would have to be operating beyond design basis.

MR. THADANI: But this is sort of dependent on the timing, isn't it?

MR. LOCHBAUM: Yes. If the answer to both of those questions is yes, then you can initiate the fuel pool cooling system to using the affected -- or the accident unit's fuel pool cooling system to restore fuel pool cooling. If the answer to any of those questions is no or if you don't have Class 1E power, then you go down and look at can you use the RHR loop in the accident unit.

The first box is is a loop available. To date --

or at the time of the 10 CFR 21 report, PP&L had not shown that an RHR loop will be available following an accident. RHR is used for many post-accident functions, like core cooling and containment cooling, suppression pool cooling.

If you have to assume single failure of one of those RHR loops could be out of service, or RHR pumps, so that there may not be an RHR loop available for this function. If there is an RHR loop available, the second question is are the RHR fuel pool cooling components operable. These valves -- I haven't talked about these.

The RHR fuel pool cooling system is used down here. It can take suction from a skimmer surge tank, fuel pool, through an RHR pump, to an RHR heat exchanger, using RHR service water as the coolant, and return to the fuel pool. So that's the cooling loop.

To do that, you have to close a manual valve to the fuel pool cooling system and open three manual valves to the RHR fuel pool cooling system, and then you could initiate fuel pool cooling.

MR. CALVO: That pipe that comes from the skimmer tank.

MR. LOCHBAUM: Yes.

MR. CALVO: Is that one Category 1?

MR. LOCHBAUM: This pipe?

MR. CALVO: Yes.

MR. LOCHBAUM: This pipe to here are all Category 1.

MR. CALVO: That one that is not Category 1 is the one that goes down that way.

MR. LOCHBAUM: On this diagram, it would be everything from this side of the valve and on this side of the valve. It's seismic down to the valve into the RHR system.

MR. CALVO: I see.

MR. LOCHBAUM: It's seismic Class 1 to the RHR system and to the valve.

MR. CALVO: I see.

MR. PREVATTE: By the way, a point that Dave just made, we went by it very quickly, in order to initiate this mode of cooling, you have to operate manual valves. Again, those valves are in the reactor building. The operators can't get to them.

MR. LOCHBAUM: These three valves that have to be operated in order to allow fuel pool cooling were removed from the PP&L in-service inspection program in 1987 and 1988, because PP&L determined they didn't provide a safety function. So they haven't been tested. At the time of our report in November of 1992, they hadn't been tested in almost five years.

If the fuel pool cooling system assist components of RHR are available and operable, the next question is do you have adequate net positive suction head for the RHR pump. The RHR pump takes suction from the skimmer surge tank. The skimmer surge tank, according to PP&L, is not sized properly or large enough to allow -- to maintain sufficient net positive suction head when you start the pump.

In pre-op tests, they drained the tank dry and the pump got heavy vibration, cavitation. They could not demonstrate the system could work.

MR. PREVATTE: Do we want to mention here the calculation they had done?

MR. LOCHBAUM: I'm not that familiar with the calculation. You'd have to cover the calculation. One other point I'd like to make here, though, is that the pre-op test that was done was done with the fuel pool cooling water and the skimmer surge tank water relatively cool. It was on the order of 80 degrees or perhaps less.

By the time they got around to initiating this loop of RHR, you've already lost the fuel pool cooling system for some period of time. The water is going to be sufficiently higher than 80 degrees. It's not analyzed for any operation above 125 degrees.

It's very doubtful, extremely doubtful that you'd have enough suction head with this water above 125 degrees, like you'll see following an accident when you go to align that system.

Assuming you did have enough net positive suction head to start the pump, the next question is do you have enough ultimate heat sink capacity to handle the heat load that's now going to be going through the heat exchangers and out to the ultimate heat sink. It has not been designed for that.

This heat load is approximately 13.2 -- the design heat load is on the order of 13 million Btu's per hour. It's not been included in the PP&L spray pond analysis. There was a preliminary spray pond analysis performed by PP&L that showed that they did have sufficient capacity. That was based on several assumptions, such that both RHR spray pond loops had to be operable.

In addition to both loops being operable, they had to go out and optimize the spray pattern in order to remove the decay heat from the fuel pool cooling system.

Even if the spray pond has sufficient capacity, the next question is do you have an accident source term present. If the answer to that is yes, then you can't use RHR because you'll be taking and pumping water with -- very high radioactivity level water, pumping it up to Zone 3, where it's not been analyzed.

MR. PREVATTE: It's essentially bypassing your primary containment when you do that.

MR. LOCHBAUM: If you initiate that with accident source terms, you'll be pumping water essentially -- well, the piping, we've already seen water with accident-laden source terms. You'd then be using the same piping to pump water for the spent fuel pool around. So some of these accident source terms will get up here, where they have not been analyzed for.

If the answer to that question is no and there's no accident source terms present, then you can use the RHR fuel pool cooling assist mode on the accident unit to restore fuel pool cooling. If the reactor building is not accessible or if the fuel pool cooling cannot be restored using the accident unit's systems, then we look at the second chart, the one at the top that says "B."

MR. THADANI: Dave, can I ask you a question?

MR. LOCHBAUM: Yes.

MR. THADANI: I like the approach you've used here because I always like logic charts. It's easier to follow through. I kind of break this issue in two parts. One is the design-base world, which tends to be non-mechanistic, as you know, and a more or less mechanistic world, which is what I think you're trying to do here systematically.

Have you tried to assign various pathways some probabilities to get a really good understanding of the end in terms of -- you used early on one-in-a-million-years accident with high consequences.

I was kind of curious to see if you took it further in this context to see what a revised estimate might be.

MR. LOCHBAUM: No. I looked at it from the design standpoint. I guess getting back to the question earlier, which was do we expect the fuel pool cooling system to fail.

MR. THADANI: Yes.

MR. LOCHBAUM: We don't expect to have a LOCA.

MR. THADANI: No, no. I'll give you an example of the reason I asked. The design-base accident world is kind of a sensible way to design things, but when you get into the issues of what would really happen if an accident were to take place, then I think you have to do what I think you're trying to do here, systematically understand.

The example I would use would be that if I had a loss of coolant accident, I don't expect the core to be damaged. I have ECCS. I'm supposed to make sure the cladding temperature doesn't get above 2,200 degrees.

But what I do non-mechanistically, I assume that I'm going to have a very large source term in the containment, which, in fact, wouldn't be there if I were to go at it mechanistically.

So when I think probabilities, in my mind, I am outside the design base world and I'm trying to track an accident systematically down the way; say, okay, here's the frequency of an event and then I start looking for conditional probabilities of what else has to go on.

In that context, then I would want to look to see really what are the real conditions. Are the real conditions such that there is a very large source term that people can't go in or are the real conditions the source term is really pretty small and people can go in and take actions? It's that sense.

What I think you've done here is what I would call first part. It's a different way of developing event trees and logic diagrams, but you haven't taken the quantification to that.

MR. LOCHBAUM: No. Basically, because I don't have a background in PRA. Also, the example I'd use, I graduated the quarter that Three Mile Island happened. With that accident at Susquehanna, you couldn't get into those buildings. So from a real standpoint --

MR. PREVATTE: I think that's a very good point there, what Dave just said about Three Mile Island. That was not an accident that was supposed to happen, either. But the rad levels in those buildings, the core damage was significantly greater than what is required to be assumed in Reg Guide 1.3.

So if we look at it from an analytical point of view, strictly an analytical point of view, our analyses show that the core temperature probably won't exceed about 1,600 degrees. But I think that the NRC recognized after Three Mile Island that you could, in fact, get core meltdown, because it happened there.

In NUREG 0737, after Three Mile Island, the NRC came along and reiterated the requirement to the licensees that they must assume Reg Guide 1.3 source terms. It is realistic. It's not pie in the sky.

MR. LOCHBAUM: I'd like to pick up on that second page where we go through and look at whether the equipment on the adjacent unit can be used to support the accident unit.

The first question is is non-Class 1E power available. If the answer is yes, then you can go over and try to use the fuel pool cooling system on the adjacent unit. The adjacent unit, I guess, by definition, would be if the accident is on Unit 1, we'd be trying to use the Unit 2 equipment. It's the adjacent unit. And vice versa if the accident were on Unit 2.

The first question is are the fuel pools cross-tied. If the answer to that question is no -- and I'll show you why here in a minute. Looking at the two fuel pools, you have one fuel pool, second fuel pool, and a shipping cask storage pit between them, with gates that can be opened, such that the water is one continuous volume with these gates removed.

PP&L, at the time of our report in November, tried to maintain these gates installed whenever possible due to a drainage event they had a few years ago. They only opened these gates at the time during refueling to connect all three volumes. So in an accident, either one, if these gates are removed, this fuel pool communicates with this fuel pool and you might -- you will be able to use the fuel pool cooling system on the adjacent unit to cool both pools, provided -- and that's the next few questions.

The next question is is there sufficient fuel pool cooling system capacity to handle the heat loads from both pools. The answer to that question is yes right now. The answer to that question is no for the design loads. Right now, with the fuel pools cross-tied and the current heat loads, you could initiate fuel pool cooling on the adjacent unit and cool both pools.

In the other cases, you go down and, again, you look at the same questions for the RHR system; is the building accessible. Again, when I say reactor building accessible, in this case, this is the Unit 1 reactor building, assuming a Unit 1 accident.

You have to get into the Unit 1 side to use the

Unit 2 RHR fuel pool cooling system to cool that unit. The other questions remain the same, so I won't go through them.

If you're not able to restore fuel pool cooling using the adjacent unit's equipment, you end up, on the third page, with a boiling spent fuel pool.

Now we look at what happens once you have a boiling spent fuel pool, what you can do or what's the design. The first question is, again, is the reactor building accessible. This is the reactor building on the accident unit. Again, as we've said, with assumed Reg Guide 1.3 source terms, it will not be.

But if you can't access the building, the next question is is the fuel pool cooling system piping intact. The reason that question is asked is that PP&L stated that the RHR fuel pool cooling system's piping is separate Class 1 from the non-seismic piping. That's true only if that valve is closed. It's a manual valve inside the reactor building.

With this valve open, this non-seismic piping is connected directly to the seismic piping. So if this piping fails, the water will be draining right onto the floor in the reactor building and not going through the piping to RHR. I'm sorry. I had the wrong valve. It's this valve that has to be closed. This valve has to be open to use RHR. But this is the boundary valve between non-seismic and seismic piping.

So if that piping is no intact, you have to go in there and close that valve. Again, at this point, you'd want to open up the ESW valves to allow makeup to the boiling spent fuel pool. The other significance of this is even if you open up these valves and this piping is not intact with this valve open, it would just fill up here, overflow into the skimmer surge tank and out.

Once you open up these valves, the PP&L procedure calls for the level to be restored to the desired level, then you close the valves. So it's a batch cycle. You lose level, you open up the valves, restore the level, shut the valves, and repeat it.

The purpose for using batch mode is primarily to prevent the fuel pool from being overflowed with constant makeup and water going directly into the reactor building basement and other areas from overflow.

The next question down there is is the standby gas treatment system designed for the temperature and moisture effects from boiling. As Don covered earlier and we will address later, the answer to that question is no from a design standpoint. If it does survive, then the next question is are the standby gas treatment components environmentally qualified for the boiling conditions.

Again, the answer to that from a design standpoint is no. If the answer to either one of those questions is no, you end up with a loss of secondary containment, no filtration of activity as it goes to the environment.

MR. PREVATTE: By the way, just to let you know what the significance of that is, standby gas treatment has a hundred-to-one cleanup ratio. That means if standby gas treatment fails, the off-site doses are going to be a hundred times higher than they were before, with nothing else failed.

MR. LOCHBAUM: The next question is -- and I left out a word here, but the question is are the ECCS pump room coolers designed for latent heat. As Don discussed earlier, the answer to that question is no from a design standpoint. If they do survive those conditions, the next question is are the safety switchgear room coolers designed for latent heat. The answer to that question is no from a design standpoint.

The consequence of this failing is that essentially you lose power to safety-related equipment in the reactor building. If they do survive, the next question is is flooding controlled in the reactor building. As Don has alluded to and as PP&L has identified in their submittals, the answer to that question is no from a design standpoint.

If the answer to those questions is no, you lose

ECCS pumps. You lose containment cooling. The result of that is a meltdown of fuel in the reactor core.

If you go through the other path, then you can maintain the spent fuel pool level using the accident unit's ESW system.

The plot on the next page -- the next page covers trying to use the adjacent unit's ESW system to make up to the boiling spent fuel pool. It's very similar. The only difference is in the way that the adjacent unit's ESW system would be used.

The way PP&L has proposed it is that you'd have --this is the adjacent unit, this is the accident unit. You would open up these valves, which are in the adjacent unit's reactor building. You would fill up the spent fuel pool, overflow into the skimmer surge tank, overflow into the cask storage pit, overflow into the skimmer surge tank, and eventually make its way into the accident unit's spent fuel pool.

I guess one of the things we'd like to point out is that if the -- the reason you'd be using the adjacent unit's ESW system is if the accident unit's reactor building is inaccessible. If the accident unit's reactor building is inaccessible, you won't be able to close this valve. So when you get all this overflow, it will go onto the floor in the reactor building and will not make it into the spent fuel pool.

The rest of the consequences of this fault tree are the same as on the preceding page. So I won't go through those again.

With all that, if you get through those, on the last page, you have a boiling spent fuel pool. You vent without makeup, which leads to uncovery of the irradiated fuel in that spent fuel pool, severe damage of the irradiated fuel in that spent fuel pool, and you have the release of activity, significant amounts of reactivity outside of primary containment, after you've already lost secondary containment.

MR. PREVATTE: At this point in time, we'd like to discuss the positions that PP&L has taken and where we differ with their positions. Before we made our 10 CFR 100 report, PP&L took the position that the accident scenario described, as we described today, was not directly addressed in the FSAR, and it's not directly addressed.

The FSAR had been approved by the NRC. PP&L had an unwritten agreement with the NRC that spent fuel pool boiling concurrent with LOCA didn't have to be considered and, therefore, it was not a valid concern.

We had a lot of problems with that. First of all, it is our position that an event doesn't have to be directly addressed in the FSAR to be a part of the licensing basis.

If it mechanistically results from an event that is described in the licensing basis, it's a part of the licensing basis.

Second, the fact that the NRC approved it, even if it hadn't been a part of the original licensing basis, doesn't mean that it isn't a valid issue and needs to be addressed. Just because it was not caught the first time around doesn't mean you can ignore it. You have to address it.

The third point there, that they had an unwritten agreement with the NRC, that's ludicrous.

MR. THADANI: I don't know what that means.

MR. PREVATTE: I don't know what that means, either, but that was an argument that was made to us time and again while we were discussing this.

There's also another point that needs to be made here. The original FSAR did describe some of the effects of this boiling spent fuel pool. The original FSAR listed as the design temperature for the air entering the standby gas treatment system as 180 degrees. That number was taken from a Bechtel calc in 1979, which did account for a boiling spent fuel pool. That was a part of the original design of the equipment.

They subsequently changed all the design of the equipment. They said to themselves we're not going to have a boiling spent fuel pool; therefore, we can change the design temperature to 135 degrees, which is what you get without a boiling spent fuel pool.

In 1984, that 180-degree temperature was still in the FSAR. In 1984, PP&L submitted an FSAR change request to the NRC to change that number from 180 degrees to 135 degrees. The reason they gave for the change was it was a typographical error. It was not a typographical error. That number of 180 degrees was the correct number.

PP&L's original position as we discussed this with them was that they did not have to consider Reg Guide 1.3 source terms. They also said that even if they did have to consider it, their operators could take the needed heroic actions to save the day and that their emergency management organization could handle any condition, even if it hadn't been analyzed.

We don't agree with that position, either. Since we've made our Part 21 report, PP&L's position has changed somewhat. They still claim that this accident scenario is not a part of their licensing basis and, therefore, they're not required to deal with it.

However, they finally concede that Reg Guide 1.3 is the required source term applicable for operator access and that airborne radiation must be considered. By the way, that's another point to make. Before we made our Part 21 report, PP&L contended they did not have to consider airborne radiation. Even though it would be there in an accident condition, they didn't have to consider it.

Their reason that they didn't have to consider it was it wasn't addressed in the original FSAR. That's true. The original FSAR did not describe airborne radiation. It was an obvious oversight. There is airborne radiation in a design basis LOCA. That made a big difference in whether or not operators would have access to the building.

Since we made our report, they've made numerous equipment, procedure, analysis and training changes and these are all for conditions that they maintained before the report were of minimal safety significance. It's hard for me to understand why these massive changes would have to be made for something that is of minimal safety significance.

These have only been made since we made our report. Even though they've made these changes, it still hasn't fixed the problem. It's definitely a big improvement over what it was before the report, but the problem is still there. PP&L is forced to admit, even with these changes, there are a number of shortcomings in the design with equipment failures for this accident scenario.

For instance, their core spray pump, they already admit that one of the core spray pumps fails and unspecified other safety-related equipment. They've taken the position that that's okay if the core spray pump fails because they have another core spray pump to back it up.

Well, that is not the way single failure works. Single failure says you may have a failure in spite of all your best design efforts to make it good, not because of a faulty design. You're not allowed to take a failure as a result of a design that's inadequate.

Also, they've admitted that the safety-related instrumentation that the operator would have to monitor in order to know what's going on in the fuel pool is inaccessible. And even with these changes in failures, they still contend that there is not now and never was any problem and they're asking the NRC to buy into this position. We feel that that is absolutely inappropriate.

There are some other points of contention that Dave will go over here.

MR. LOCHBAUM: There's no real order to these points of contention. They're not in order. A point of contention was that the fuel pool instrumentation -- PP&L's claim is that the fuel pool instrumentation is seismically qualified. The reality is that the instrumentation is only seismically mounted, not seismically qualified.

In addition, this instrumentation is neither qualified for the LOCA conditions nor the boiling spent fuel pool conditions. Let me show you what I'm talking about by this instrumentation.

It's now shown on this diagram, but the only instrument that they have in their environmental qualification files is the level instrument on each of these skimmer surge tanks to tell you what the level is. That's the only instrument that's seismically mounted. There's also level instrumentation in the spent fuel pools, temperature instrumentation in the spent fuel pools. It is not even seismically mounted or at least at the time of our report in November 1992.

All of this instrumentation, including the seismically mounted skimmer surge tank level instrumentation, is not designed for the post-LOCA radiation, humidity and temperature effects. It is not designed for the temperature and humidity resulting from blowing the spent fuel pool.

Another claim PP&L has made is that the fuel pool instrumentation is Class 1E qualified. The reality is that it can be powered from a 1E source, but the circuitry is not 1E powered and the instruments are not 1E qualified.

Specifically, there's a local panel, I believe, on the fuel pool cooling system elevation, that gets a signal from the instrumentation that I showed you earlier. This panel can be powered off of the diesel generators, Class 1E power.

So the temperature level instrument, level instrument, feed a local panel in the reactor building. That panel is Class 1E powered. Those instruments are not. When you have a loss of off-site power, you lose these instruments. That panel is still powered. So it sends a signal to the control room operator telling him he's got a problem. As we discussed earlier, there's not much that the operator can do about it, but he does know there's a problem.

MR. PREVATTE: But he doesn't know the specifics of the problem.

MR. LOCHBAUM: He doesn't know what the temperature or what the level is.

MR. PREVATTE: He gets an alarm window that says fuel pool cooling trouble. That's all he knows.

MR. LOCHBAUM: Just a common trouble alarm is all he gets. In 1984, 1988 and 1992, PP&L's Nuclear Safety Assurance Group has examined fuel pool cooling and recommended that this instrumentation be upgraded. I would point out that the Nuclear Safety Assurance Group was formed after Three Mile Island to be an independent agency to look at safety issues from outside the industry and from within the plant to make sure that the plant is operated safely.

Three times this plant said that instrumentation needed to be upgraded. As of November 1992, that instrumentation had not been upgraded.

Also, there were three separate events that were looked at here. In 1984, they were looking at an NRC correspondence on reactor cavity seal failure. They said Susquehanna's would be upgraded if the instrumentation was upgraded.

In 1988, they were looking at an event that happened at Susquehanna when they inadvertently drained the fuel pool to the condensate storage tank. In 1992, they were looking at just general refueling risk or the risk associated with shutdown. At that time, all three times, they recommended the instrumentation be upgraded.

My understanding is that looking at this issue, the group has again recommended that the instrumentation be upgraded.

Now it's your turn.

MR. PREVATTE: Another claim that PP&L has made is that the standby gas treatment system is qualified for boiling spent fuel pool conditions. The reality of this situation is if the equipment in the rooms is not qualified for the temperature resulting from the boiling spent fuel pool -- that is, you're going to have very hot air passing through the system and it will tend to heat up the room.

In that same room, you have other equipment that is vital to the operation of the standby gas treatment system. That equipment is not designed to handle the temperature that would be generated.

Also, before we made our Part 21 report, the HVAC system, the ductwork which carried all the air that's coming to the standby gas treatment system had fire dampers in them and these fire dampers were designed to close at 165 degrees. We understand that since we've made our report, they've done a modification on those to raise the actuation temperature.

It's obvious if you have a 165-degree damper and you have 180 degree air going through it, the damper is going to close and it's not going to function any longer.

Another claim that PP&L has made is that the reactor building HVAC recirculation system can be isolated to prevent boiling fuel pool heat and moisture from reaching the safety-related equipment in the reactor building. It's my understanding that that's their prime position right now for coping with this situation.

That is they will shut down the recirculation system, keep all the heat and moisture on the refueling floor, and, therefore, it won't get to the safety-related equipment in the reactor building.

MR. THADANI: I missed something there. Boiling fuel pool generates 185 degrees or 180 degrees? Is 180 for Susquehanna? I'm a little unsure. The handout says 185.

MR. PREVATTE: I think the number is 180. I think that was a mistake.

MR. LOCHBAUM: It was a Bechtel calculation.

MR. PREVATTE: I think the Bechtel calculation said 180. I'm not sure. PP&L has claimed that they can isolate their recirculation system, thereby keeping the heat and the moisture up on the refueling floor.

Well, if they did that, that would be an unanalyzed condition. Let's discuss for a moment why you have a recirculation system in the first place. It serves several functions.

The first function is it mixes up the atmosphere such that any leakage out of the primary containment is diluted. Therefore, when it goes through the standby gas treatment system, it will minimize the off-site dose at any one point in time. By securing the recirculation system, you have the potential for getting concentrations of radioactive gas in the building which can be pulled into the standby gas treatment system and give you excessive off-site doses. That condition has not been analyzed.

The second reason you have a recirculation system is that all the equipment in the building, its temperature qualification is based on the air being mixed pretty thoroughly and, therefore, being at a relatively uniform temperature. If you don't have the recirculation system going, the air becomes stagnant and in various rooms where safety-related equipment is located, it hasn't been analyzed for that condition.

The third reason is because with the original design, the Zone 3 area acted as a big heat sink. If you look at the relative volumes, the Zone 1 volume is about 1.5 million cubic feet in the reactor building. The Zone 3 volume, which covers both units, is about 2.7 million cubic feet. That 2.7 million cubic feet, in an accident condition, under the original design, without a boiling spent fuel pool, would tend to mix with -- when the recirculation system is running, would mix with the air that was in the rest of the building and give you an overall temperature significantly lower than you would get if the system is not running.

If the system is not running, then all the heat that's in Zone 1 stays in Zone 1. So it's likely that temperatures are going to be significantly higher even without the effects of blowing spent fuel pool. None of these conditions have been analyzed.

MR. LOCHBAUM: Related to that issue is that PP&L has indicated that the boil-off in the boiling spent fuel pool can be contained in Zone 3. Some of the condensate from that boil-off will go into the Unit 1 reactor building sump and approximately half of it will go into the Unit 2 reactor building sump.

Zone 3, by the time the isolation is shut down or if it is not shut down, it's going to have some accident source terms in Zone 3. This condensate is going to capture or transport some of these source terms into the Unit 2 and Unit 1 reactor building sumps. Both units are going to see high radiation levels during the course of the accident. All three zones are going to have to be cross-connected.

PP&L has relied heavily on answering the adjacent unit to perform equipment operations. They have not explained or have not identified how the condensate that ends up in the reactor building sumps on the non-accident unit will be handled.

MR. PREVATTE: By the way, one other point that seems to have been lost in the conversations to this point, Zone 3 is not isolated from the primary containment. The single largest penetration of the primary containment is in Zone 3, and that is the drywell head. It also has several penetrations.

So any leakage from those penetrations will tend to contaminate Zone 3, even if you don't have the recirculation system running, and that, in turn, will contaminate any condensate which would drain into the non-accident unit and potentially render it inaccessible.

MR. LOCHBAUM: Another PP&L claim was that the emergency procedures and training at the time of the 10 CFR 21 report adequately addressed spent fuel pool boiling events. The changes they have made since our report are just enhancements to this procedure.

In fact, procedures at the time of our 10 CFR 21 report caused a loss of spent fuel pool cooling. The current changes address the boiling spent fuel pool case for really the first time in any consequence.

Prior to that time, the procedures were basically looking at making up due to a seismic event. We've had a loss of piping integrity. You can go into the reactor building at will and do anything you want to. They did not have to contend -- the procedures did not contend with high radiation effects and use of other equipment in the adjacent unit.

Getting back to the procedure, in 1988, in response to high EQ problems or EQ problems in the reactor building, certain components could not be qualified for the high temperatures they were experiencing for a LOCA in which you did not have a loss of off-site power.

That resulted in the fact that in the case where you have a LOCA and not loss of off-site power, you'll have non-Class 1E equipment operating that wasn't considered in the original calculation, original heat load calc, and also used in the EQ problems.

In order to get around this hot temperature problem, PP&L initiated, in 1988, a change to their procedures that caused non-Class 1E power to the reactor building to be shed manually approximately 24 hours after a LOCA.

That introduced a loss of spent fuel pool cooling because, as we stated earlier, the fuel pool cooling system pumps are not 1E power. They won't be shut down when this load shed occurs.

If, in fact, PP&L believes that the only licensing basis event in spent fuel pool cooling for Susquehanna is a seismic event, then this procedure change in 1988 introduced another failure mode for that event and increased the probability to one with this event. Otherwise, it increased the probability of the existing event to one from whatever it was before. In either case, you end up with a loss of spent fuel pool cooling.

MR. PREVATTE: Another claim that PP&L has made is that the Reg Guide 1.3 source terms that are required to be considered constitutes a severe accident, that it's outside their design and licensing basis. They've used the term "severe accident" throughout their submittals describing what we have considered to be a design basis accident.

We consider that to be a blatant misuse of that term. Heretofore, until PP&L started using it that way, that term has only been applied to those accident events that were outside the licensing basis; that is, events such as ATWS.

This is not an event like ATWS. This is in their licensing basis. It is not a "severe accident." We believe that that usage of that term is extremely misleading.

MR. THADANI: But you would agree -- getting away from the design basis accident considerations and the non-mechanistic aspects of it, you would agree, though, that in order to best understand what the source terms would be and so on, it would be the right thing to do to use severe accident source terms, don't you?

MR. PREVATTE: No.

MR. THADANI: Getting away from design basis accident considerations.

MR. PREVATTE: The job I was assigned at PP&L was to look at things from a design standpoint. Therefore, I have to go to the underlying requirements established by the NRC. That's the job I did.

MR. THADANI: I understand that. I was just reacting to this statement when they talk about severe accidents. Again, going back to what I said earlier, one of the things I'd like to make sure I understand is what one has to do in the context of design basis accident, Reg Guide 1.3 source term that goes with that and the conditions that would exist for that case.

Parallel to that, I want to make sure that we have a clear understanding of what would happen if you, in fact, had a big accident. What would be the real source term? What time would be available for humans to take appropriate actions? And 24 hours later, even if you manually shed certain loads in terms of spent fuel pool cooling and so on.

What I guess I'm trying to make sure we understand is during those time periods and using those realistic scenarios, what would be the source term, what would be the threat to humans who have to perform these actions.

MR. PREVATTE: All I can say with regard to realistic versus nonrealistic is we have an experience level of one reactor plant accident in this country and that experience level produced a core meltdown, core damage significantly worse than what's required by Reg Guide 1.3.

So if you want to talk about realistic versus nonrealistic, I think what we are talking about is realistic. It's been proven by experience.

MR. THADANI: We could go on about what the source term was in the containment, but that's okay.

MR. PREVATTE: The point is that you can get severe core damage. It is not an unrealistic thing.

MR. THADANI: I agree with you that you can. In fact, when you estimate probabilities of those things happening of one-in-a-million, I don't question one-in-a-million chance of core damage.

MR. PREVATTE: One-in-a-million is basically what we design for for a LOCA. That is a starting point for designing.

MR. THADANI: We have studies that we do. I hope you don't misunderstand what I'm saying. Certainly there is no question of -- it's not a question of zero risk. But the idea and the concept is to make sure risk is low enough and there you have to take into account probabilities and consequences, just as you kind of touched upon early on.

And there are -- at probabilities of one-in-a-million,. I think there is a chance of a severe accident. I think in today's technology, you have to recognize there are certain limits to what one can do and I would never say that there was zero risk.

MR. PREVATTE: We don't believe there is zero risk. We believe the risk is very high. As you pointed out, you must consider probability. We say probability is the same as a LOCA, but the consequences are much worse.

MR. LOCHBAUM: At the time of our Part 21 report, all the documentation available within PP&L was that the reactor building was inaccessible following a design basis LOCA due to high radiation. In a report submitted to the NRC earlier this year, PP&L reported the results of a time-motion study to determine whether the ESW valves on the accident unit could be operated to provide makeup to the boiling spent fuel pool.

PP&L indicates that the results of this are such that the ESW valves may be or can be operated within the five rem exposure limit. The problem we have with this study, and this is without having reviewed the study, just reviewed what they reported to the NRC, is that it appeared as though the time-motion study indicated an exposure of 4.2 rem for a single trip to operate these valves.

As I discussed earlier, the procedure calls for these valves to be operated in a batch mode. You open them, you restore the level, you close them, and you repeat that as often as necessary to maintain the desired level of spent fuel pool. If each single trip to these valves results in a 4.2 rem exposure to an individual, to an operator and probably to an HP tech who will be accompanying him, then that operator can only make a single trip to the valves without exceeding the five rem dose.

So you'll cycle through a number of operators, depending upon how many trips you have to make to those valves.

In addition, it wasn't clear whether the operator goes there, opens the valve, leaves and returns later to close it when the level is restored or if he stays there until the level is restored. It's also not clear since all this instrumentation is not available to the control room operator, the procedures at the time of our 21 report said the operator was supposed to go up there and physically look at the level in the pool and he was supposed to call the guy down on this level to tell him when to close the valves.

I don't know how they currently do it. That would result in much higher radiation levels than what we're reporting.

MR. PREVATTE: There's also one other point to make about this and that is these aren't going to be easy trips these guys make. They're going to be dressed out in the most elaborate anti-Cs one can imagine in order to survive the radiation conditions.

The temperature conditions in the reactor building at best are going to be about 135 degrees as they do this, not to mention besides the radiation conditions. There's potentially not going to be light in the building, either, except for emergency lighting.

So we're not talking about something that the guy is going to casually stroll down and do. It's going to be an extremely tough thing to do.

MR. LOCHBAUM: There's also a related issue. PP&L has also indicated that any -- they submitted a report to the NRC earlier this year that states that any time, under any condition, they can go into the adjacent unit -- assuming an accident on this unit, they can go into the adjacent unit and open these valves and get water back to the accident unit through the cascading forward system.

As we pointed out, if this building is inaccessible, you cannot close this valve. If this valve cannot be closed, this non-seismic piping has to be intact in order for it not to just drain the skimmer surge tank out onto the floor and prevent water from reaching the accident unit spent fuel pool.

Again, this piping is not seismically qualified. It's not designed to handle the hydrodynamic loads that result from the LOCA itself. It cannot be assumed to be there.

MR. PREVATTE: Another claim that PP&L has made is that the flood --

MR. THADANI: Can I ask you on that one, because I think that is -- you make, I think, an important point. If you were to calculate these loads, would you expect that piping to fail, in your judgment?

MR. LOCHBAUM: I'm not really a civil engineer. I can only report what the PP&L civil engineers have indicated. That is there will be stress beyond design on the hangers and you'll get the zipper effect of piping going.

MR. PREVATTE: Another claim that PP&L has made is that the flood water from the boiling pool can be sent to rad waste. That's how they would handle it. Well, the reality of that situation is the sump pumps that would be required to send it to rad waste, they also aren't 1E powered, they're seismically designed, they're not environmentally designed.

So there's a very high potential that those pumps would not be available. Also, operators wouldn't have access, again, due to radiation levels. Also, the rad waste systems themselves aren't 1E powered or seismically qualified. So even if they send the water over there, they may not be able to process it.

Additionally, the rad waste systems aren't designed to handle the accident water volumes or the radiological content that they would have. They are not designed to handle accident water.

Just the mere fact of sending water over to rad waste would be a breach of secondary containment. That's exactly what you don't want to do in a LOCA event. And even if they got all the water over to rad waste and even if they processed it all, just like they say, and everything went fine, then they've got to do something with it, and we're not talking a small volume of water here. We're talking as much as 5 million gallons of water.

If you recall, TMI just recently got rid of the water that they had been storing for years since the TMI accident. The reason was it was laden with tritium. This same accident water would be here in this case. They don't have any facilities whatsoever for storing this kind of water on-site and no way to get rid of it.

So from a practical point of view, they can't get rid of the water that's in the reactor building.

MR. CALVO: Can you put that picture back again?

MR. LOCHBAUM: Sure.

MR. CALVO: That pipe in there, that it breaks. How big is that pipe?

MR. LOCHBAUM: It is a ten-inch pipe. I believe it's a ten-inch header down to here and then splits into six or eight-inch piping, I believe.

MR. CALVO: No matter what you do, if that pipe breaks, you're going to have all these problems. The RHR will not work. The emergency service water system will not work.

MR. LOCHBAUM: If you close this valve, you isolate it.

MR. CALVO: Your piping up above it is not also Category 1. It also may go, too. Suppose you cannot close the valve.

MR. LOCHBAUM: That's true.

MR. CALVO: If you break that pipe, nothing you can do is going to help. So it's all key on that pipe.

MR. LOCHBAUM: That's correct.

MR. CALVO: Because of that and you put water in the sump, then you -- see, no matter -- if that pipe breaks, the RHR will not help you at all. The emergency service water system won't help you at all. So all you will do is you put the water in and dump it someplace else and then you've got to worry about how to get all that water.

So the key to the whole thing is that pipe. If it doesn't break, then all this will go away.

MR. PREVATTE: It's not even key on that. Even if the pipe remains intact, you still have to deal with all this water, because you're putting water into the fuel pool. Even if this is intact, you're putting water into the fuel pool to make up for the boiling. The water is evaporating out as it boils.

That water that evaporates out condenses and that condensate runs down into the basement of the building. So even if you don't break this pipe, you've got a basement full of water.

MR. CALVO: That's right, because if I lose the service water system, I cannot cool. That's correct. But if that pipe doesn't break and there's some kind of other way I can put back the fuel coolant pumps back in service, then I'm all right.

MR. LOCHBAUM: Yes, but these are non-1E powered, not single failure.

MR. CALVO: That's correct, but those are very small pumps. I guess the pump to fix the problem -- if that pipe doesn't break, then I can find out if I have enough capacity in the diesel generators to make those pumps work.

MR. LOCHBAUM: These pumps are fairly small, but in order to cool these pumps, the service water system has to operate. Those are fairly large pumps.

MR. CALVO: Isn't the service water system connected to the on-site power system for the diesels?

MR. LOCHBAUM: No.

MR. CALVO: They're not?

MR. LOCHBAUM: The RHR service water system pumps are.

MR. PREVATTE: I believe these may be. The service water system, I think they may be, but that's not a normal lineup. Even if they are, even if they can be, they're very large pumps and the question then becomes do the diesels have sufficient capacity to handle all the other accident loads they have to handle plus service water, and that's very questionable.

MR. CALVO: That's correct. But during the initial part of the LOCA, it's the demand for loads. As the

LOCA moves along, then the load demand is not as big as it was before and you may have enough capacity in there to cool the service water pumps into the diesel.

MR. PREVATTE: That's correct.

MR. CALVO: The question is whether the capability is there to cool the fuel cooling pumps. You may not have it right now. Those supposedly are small pumps, right?

MR. LOCHBAUM: Correct.

MR. PREVATTE: If they survive the accident, if they survive the environment and the hydrodynamic loads.

MR. CALVO: I agree.

MR. THADANI: In your calculations, how long was it to the fuel pool boiling?

MR. LOCHBAUM: For the design loads, 19.8 hours.

MR. THADANI: So approximately a little less than a day.

MR. PREVATTE: Understand that, also. That's another very important point. This calculated time to boil is approximately 19 hours. The time until you go beyond the design condition for the fuel pool, though, is virtually immediately. The fuel pool design temperature is 125 degrees.

Any temperature over that 125 degrees is an unanalyzed condition. When you start getting up around 150 to 160 degrees, you're going to start getting a lot of heat coming off that fuel pool, a lot of vapor. It won't be in a boiling condition yet, but just because you haven't reached boiling yet doesn't mean that it's not giving you negative effects.

You start getting negative effects as soon as the temperature starts to rise. As you approach boiling, they get worse and worse. So you don't have to actually be at boiling to have negative --

MR. THADANI: When you say negative effects, I'm trying to follow what that means.

MR. PREVATTE: What that means is you're getting heat from the fuel pool. You're also getting vapor from the fuel pool. You don't have to have boiling to be getting --

MR. THADANI: I understand. But it's multi-million cubic feet volume. I guess I was just trying to understand. The real loads and the issue, it seems to me, would be once you start to boil and you start -- you probably have more stratification issues, condensation issues, I think probably you have to be much more detailed in your assessment of what is really going on and what components you have to protect.

MR. PREVATTE: There's no question that at the boiling condition there's a tremendous amount of heat and moisture to contend with. I don't know this for a fact, but it's my understanding that PP&L has assessed the effects of these less than boiling conditions and it's my understanding they have an evaluation that shows they are detrimental.

I haven't done the analysis myself. There's one other thing. Another claim that PP&L has made with regard to this flooding is that they have conceded that one of the core spray pumps will fail as a result of flooding. This is the point we made before.

They have contended that that's okay. Well, it's not okay. With only one pump failed, then any single failure in the core spray system will render the core spray system entirely out of service. That is unacceptable.

That failure has been based on their calculated volumes, which were extremely non-conservative. If you look at the calculated volumes of water that you are likely to get for this condition, where the operator goes down and opens the valve to the full open position, the watertight door on their second core spray pump room will also fail.

So under the design basis accident conditions that we would expect to see, they will get failure of both core spray pumps.

MR. LOCHBAUM: Getting back to the rates that PP&L is reporting as far as makeup rates and flooding rates and also time-to-boil rates, in a lot of cases, if not the majority of cases, they're reporting the conditions as of today, which are not design conditions. The pools are not filled to capacity. I have not seen too many, if any reports of what happens when the pools are filled, design conditions.

One of PP&L's early contentions which we disputed, which has been withdrawn somewhat in our later submittals, was that RHR could be used in the fuel pool cooling assist mode to prevent boiling spent fuel pool from ever occurring.

As I kind of went through earlier with the fault trees, the facts dispute this conclusion or this claim in that the critical valves are inaccessible after a design basis LOCA in the accident unit due to radiation levels. PP&L's analysis indicated that RHR pumps do not have sufficient net positive suction head to operate. They confirmed these analyses with pre-op testing which showed that the skimmer surge tank was pumped dry before the system could be aligned for fuel pool cooling.

In addition, these pre-op tests were done with relatively low temperatures, water temperatures. At the temperatures that you would encounter post-LOCA of 125 degrees or greater, it would even more doubtful you'd have adequate net positive suction head.

In addition, the spray pond at the time of our report did not have sufficient capacity for the fuel pool cooling heat load without requiring both loops to be operable. So it was not single failure-proof, as it was required to be.

It was required to be single failure-proof by 10 CFR 50, Appendix A, General Design Criteria 44, which is the cooling water system spec which requires all the heat from safety-related structures to be removed with safety-related equipment that's 1E powered and all the other pedigree. They do not meet GDC-44 at the time or, best I can tell, as of today.

As I said, a single failure cannot be tolerated. Not all the involved piping is seismically qualified. That valve has to be closed to isolate the non-seismic piping. In addition, as we alluded to earlier, the manual valves that are required to be operated to align fuel pool cooling assist mode were removed from the PP&L in-service inspection program in 1987 and 1988.

Following an engineering evaluation by PP&L, it determined that the RHR fuel pool cooling assist mode had no safety function. The ESW system would be used alone following an accident for the boiling spent fuel pool case.

MR. PREVATTE: Another claim that PP&L has made is with regard to the loss of off-site power. They maintain that it could not last any longer than 24 hours. We dispute that claim based on regulations and experience.

Reg Guide 1.137, for instance, requires that emergency diesel generators have a minimum of seven days fuel supply on hand. One could logically infer from that that the NRC expected that a loss of off-site power would last longer than 24 hours.

Recent experience with Turkey Point also points out that they can last longer than 24 hours. They had a six-and-a-half day loss of off-site power there. At a number of other plants around the country, they have had loss of off-site power that lasted longer than 24 hours.

There are three that we have identified credible, long-term loss of off-site power mechanisms. There would be natural events, such as the Turkey Point event, and at Susquehanna, a similar natural event could be like an earthquake.

Operator error, an example of that would be Plant Vogtle, where they had a significant loss of off-site power as a result of operator error. Also, at Susquehanna, they had loss of off-site power there, it didn't last 24 hours, but as a result of operator error.

Third is sabotage. If anybody wanted to sabotage off-site power, the power transmission lines, they could put Susquehanna -- take out their off-site power and that could not be restored in much longer than 24 hours.

MR. LOCHBAUM: This occurred at Palo Verde a few years ago.

MR. CALVO: Keep in mind that in the question of postulating a LOCA, sabotage has too many things to happen at one time.

MR. PREVATTE: No, no. We're not postulating concurrent. We're saying the LOOP would be caused by sabotage. That is that would be one of the mechanisms for it.

MR. CALVO: All I'm saying is you're postulating a LOCA concurrent with off-site power and then the recovery to be followed by the sabotage. You're talking about a lot of things have to happen.

MR. PREVATTE: I understand. With regard to a LOOP, we're not saying that the LOOP is necessary in order for this accident to occur. We're just saying that a LOOP is one of the things that can cause a loss of fuel pool cooling.

There are also some unacknowledged violations here which we would like to point out. With all of the things that have been described to that PP&L admits to or doesn't admit to with regard to failures of safety-related equipment, no 10 CFR 50.73 or 50.73 reports have ever been submitted, to the best of my knowledge.

Now, the Code of Federal Regulations requires that if you have safety-related equipment such as this that's non-functional, you must report it within one hour. It's been a lot longer than one hour since it was recognized that this equipment was not functional.

MR. LOCHBAUM: The fuel pool instrumentation, as we discussed earlier, is not environmentally qualified for the post-LOCA conditions or the conditions resulting from boiling spent fuel pool. This is a violation of 10 CFR 50.49 under environmental qualification, Regulatory Guide 1.97 on instrumentation, 10 CFR 50, Appendix A, General Design Criteria 63 on fuel storage systems, and Regulatory Guide 1.13, also on fuel storage systems.

The fuel pool instrumentation, as I indicated earlier, is one level transmitter that's seismically mounted to monitor the skimmer surge tank level. It's a single instrument. So single failure takes out even that instrument.

MR. PREVATTE: As we discussed before, there's quite a bit of safety-related equipment that is cooled by coolers which aren't designed for latent heat cooling. I won't go through the details on that again. Just suffice it to say that the fact that this equipment is not designed for the environmental conditions is, again, a violation of 10 CFR 50.49, 10 CFR 50, Appendix, Criterion 3 on design control, and it includes core spray pump rooms, RHR pump rooms, HPSI pump rooms, RCIC pump rooms, and the emergency switchgear and load center rooms.

By the way, that last one, the emergency switchgear and load center rooms, if that cooling goes out, it takes out virtually everything in the plant, because everything is powered from those load centers and switchgear.

Also, as we discussed before, the safety-related ductwork is not designed for this condensation. This, again, is a violation of 10 CFR 50, Appendix B, Criterion 3 on design control.

MR. LOCHBAUM: In addition, PP&L is relying on some draft analysis in their response to the NRC which has not yet been technically reviewed and approved. Again, that's another violation of 10 CFR 50, Appendix B, Criterion 3.

I guess there was some earlier -- I got the impression that the Commission wanted us to better document our assessment of where PP&L is. I think the burden is on the utility in this case to show that they can meet these conditions, and that has not been done.

I'd like to take a minute or maybe two to go over my colleague's experience and education. Don Prevatte holds a Bachelor of Science degree in mechanical engineering. He's an officer in an engineering consulting firm. The significance of that is that's why we were obligated to make a 10 CFR 21 report, as we'll cover a little bit later.

Don has over 22 years of commercial nuclear power experience, involving design, startup and management. 13 years of that experience is with BWR designs or BWR plants. Nine years of that experience is at PP&L or with PP&L. Four years of that experience has been performing NRC or NRC-type inspections at more than 25 nuclear plants in this country.

Two-and-a-half years Don worked as a design engineer on the Susquehanna Power Uprate Project and he also has worked two years as an architect engineer discipline manager.

MR. PREVATTE: Dave has a similar background. The reason we're telling you all this is to let you know, so that hopefully you will realize that we're not coming out of left field on this stuff. It's not like we started working on these things yesterday and don't know what we're talking about. We both have significant experience, I believe, in boiling water reactors and in the industry in general.

Dave's background is he has a Bachelor of Science degree in nuclear engineering. He has over 14 years commercial nuclear power experience in startup, operations, licensing and engineering; seven years as a reactor engineer. He has three years experience as a boiling water reactor shift technical advisor.

He also has experience with all three BWR containment types, including the type that's in question here at Susquehanna. He has one year experience as a reactor engineering supervisor and STA supervisor, one year as a licensing engineer, two years as a design engineer on the Susquehanna Power Uprate Project, and one year as a design engineer on design basis document project and handling of design basis document open items.

MR. LOCHBAUM: I'd like to, at this point, go over a brief history of the concerns which culminated in the Part 21 report in November of 1992. Don Prevatte, as part of his responsibilities for the Susquehanna Power Uprate Project, prepared an updated reactor building heat load calculation. The reactor building heat load calculation looks at heat loads from piping, equipment, lighting, post-LOCA environment, and all of the other key load sources.

This calculation looked at them in three time periods -- one hour, one hour to 24 hours, and long-term periods for the LOCA and also the LOCA/LOOP case. Don took the original calculation that was done and updated it to account for the higher power level in Power Uprate.

After he prepared his calculation, I was assigned to perform a technical review of this reactor building heat load calculation. In doing the technical review, Don and I got together and discovered and reported to PP&L management in March of 1992 that both the original and the updated calculation that had been prepared non-conservatively neglected any mode of spent fuel pool cooling and also that the latent heat load from the boiling spent fuel pools that would result.

At that time, Don updated the reactor building heat load calculation for Power Uprate to assume that the fuel pool cooling system, the normal spent fuel pool cooling system was in operation for the non-LOOP case. If you had non-1E power, we would assume the system would be there.

Don also assumed that the RHR fuel pool cooling assist mode would be utilized for the loss of off-site power case following a LOCA. The research we then conducted to support these assumptions or confirm these assumptions led us to believe that the boiling spent fuel pool will result or could result from a design basis LOCA.

At that point, we felt that the calculation was not bounding since it did not address the boiling spent fuel pool case. We would not sign the calculation because it was not bounding.

PP&L asked us to initiate an engineering discrepancy report, EDRG-20020, to document our concerns with the loss of fuel pool cooling event. We initiated the EDR, amended the calculation to refer to the resolution of the EDR for that issue, and signed off the calculation.

In signing this calculation, that allowed the PP&L Power Uprate submittal to be submitted in June of 1992.

That submittal probably could not have gone in to the NRC at that date without this calculation.

MR. CALVO: When you first brought up the hydrodynamic loading --

MR. LOCHBAUM: We did not. PP&L brought that up on September 1, 1992 when they -- the Manager of Nuclear Engineering had an independent person who had not been involved with any part of this conduct a review of these concerns. He identified the fact that they were not designed for hydrodynamic loading and the LOCA event itself caused the fuel pool cooling system to fail. That was in September of 1992.

The significance we'll cover a little bit later in the separate meeting. That was the history of the concern. At that point, we signed off the calculation in good faith, thinking the PP&L engineering discrepancy report process would evaluate these concerns. When it became apparent -- or after repeated failures of this process to adequately address our concerns, we took it to PP&L management, senior management.

After PP&L management failed to properly determine operability and reportability on these concerns, we escalated it further up the PP&L management. On November 17, 1992, PP&L submitted an inaccurate, incomplete and misleading report under 10 CFR 50.9, essentially a voluntary

LER, saying that they had these problems and were taking steps to measure them, and, at the same time, failing to properly report the concerns under 10 CFR 50.72 and 50.73.

When it became apparent that PP&L was not going to take any further action and submit a complete and accurate report to the NRC, by law, under 10 CFR 21, we submitted a report of a substantial safety hazard at the Susquehanna facility. This report was dated November 27, 1992.

MR. PREVATTE: In the next section, we're going to talk a little bit about why we made the report. I guess for most people, it might be obvious, but we want to go over it anyway. We saw this as an extremely high risk accident. We also saw it had very serious generic implications; that is, a third of the plants in the country are this way.

By the way, let's stay on that point for just a minute. PP&L has made many modifications to their design, their procedures and their training since we've made our Part 21 report. Other utilities have not. So even though things are much better at PP&L than they were before our report, the other utilities in the country, except for Washington Public Power Supply System, have not officially recognized the concerns that we brought up here and potentially have made no changes whatever to their design, their procedures or the way they operate the plant.

Therefore, we consider this to be a very serious reason. We are concerned that the other plants find out about this and take appropriate measures.

We also felt that it was an ethical responsibility. Even if the law didn't require it, which it does under Part 21, we saw this as a high risk accident. We did not feel that we could, in good conscience, let it go unrecognized.

We also felt that PP&L was not properly handling the reporting of this. That's something we'll talk about in another meeting later. But we wanted to make sure that was brought to the attention of the NRC.

What's in it for us? Well, so far, it's been ridicule, loss of contracts or potential contracts, embarrassment and strain on personal relationships with coworkers, family, neighbors. I think some people implied at times that we had some ulterior motives for doing this. If we had, I wish somebody would tell me what they are, because so far I haven't been able to find any good side to this.

We've expended untold personal time, effort, and money in a very discouraging one-sided battle, trying to bring this to the proper conclusion. Quite frankly, doing something like this for people like Dave and I, in the consulting business, is equivalent to career suicide. You get labeled as a troublemaker, non-team player, and alligator. An alligator is a term that's used within the NRC, I think, sometimes to describe one who makes allegations.

Once you get labeled as an alligator, if you're in the consulting business, it's very difficult to shed that label. We went into this with our eyes open, though, knowing the seriousness of it and the potential consequences to us.

Next, Dave is going to touch on a few reasons why we know that we're right in this.

MR. LOCHBAUM: Among the reasons we know we're right is that in April or May of this year, the Washington Public Power Supply System reported the same problems at their Washington Nuclear Plant Number 2, Unit 2, and reported those under 10 CFR 50.72 and 50.73 in April and May of 1993.

In March of 1993, General Electric was aware of the Part 21 report, evaluated it, issued a warning letter to all domestic BWRs alerting them to this concern. They were not able to say if it was applied or not because GE is not responsible for the design of the fuel pool cooling systems. That's a responsibility shared between the architect engineer for the individual plant and the utility. But they thought it was a serious enough concern to make plants aware of it.

I'd like to point out that this is not unprecedented for GE to do this and it's by no means a routine occurrence, either. Also, what gave us some confidence that we were right was that four independent PP&L engineering evaluations concurred with our technical assessments. In fact, they provided -- like the September 1, 1992 report, they identified the hydrodynamic loading problem. That was not us. That was PP&L that identified that problem.

The other thing that gave us confidence was that we spent two years prior to identifying these problems researching the systems involved, the specific systems at Susquehanna. We were doing this for the Power Uprate Project, where we evaluate whether these systems could handle power uprate conditions. We had to look at them from the design standpoint and see if they could handle operating in power uprate conditions, where the operated today and what the effects of power uprate were.

After discovery of the concerns, we focused our research on the concerns, trying to -- when we were looking at those assumptions we made in the calculation, trying to justify those assumptions. We were unable to find anything to support those assumptions.

PP&L also researched these concerns. That kicked off these four independent -- PP&L engineering evaluations.

PP&L was able to resolve two of the original nine concerns we had, but was unable to resolve the remaining seven. To date, we have not seen any resolved on the remaining seven.

Since the time of our report in November 1992, PP&L has made and has committed to make extensive modifications to plant hardware, procedures, training and calculations to cope with the problem they identified or they concluded had minimal safety significance.

The point we'd like to make here is that with their process, when they identify something as minimal safety significance, they're very unlikely to take anything -- to do anything about it, because things that have moderate or high safety significance take precedent.

The reason this minimal safety significance item has been worked, we think, is because we made the Part 21 report. To support that contention, if you recall, in 1984, 1988 and 1992, their own Nuclear Safety Assurance Group recommended the instrumentation be upgraded. Those three times, they said minimal safety significance, no action was taken.

So, again, when we saw minimal safety significance, based on what we saw at PP&L, we concluded that no action would be taken.

Another thing that led us to believe --

MR. PREVATTE: Excuse me, if I may interrupt for a second. By the way, per PP&L's procedures for handling EDRs, minimal, when it's defined as minimal safety significance, per their procedures, means it is a paperwork glitch, a minor documentation problem, not a real hardware problem.

MR. LOCHBAUM: Another thing that leads us to believe that we're right was that the Massachusetts's Assistant Attorney General has had an independent consultant evaluate our Part 21 concerns, specifically for Pilgrim, but this independent consultant determined that the concerns were valid and confirmed what we had already thought.

At this point, we'd like to briefly go over what we think should be done now and what we want the NRC to do. Don is going to start this off for us.

MR. PREVATTE: We're asking that the NRC perform a thorough review of the current Susquehanna condition versus the regulatory requirements and versus common engineering sense to ensure that Susquehanna is safe.

Now, I realize that you've been doing that before, but I would suggest that in doing that, a safety system functional inspection type inspection be made, where you verify it yourself as opposed to taking PP&L's inputs at face value.

We would also like to see the NRC determine the generic impacts of this discovery on other BWRs and potentially PWRs and independent on-site fuel storage facilities. As we said, there have been numerous changes made at Susquehanna as a result of our discoveries. Most of the other plants have not initiated any changes and they probably need to.

MR. LOCHBAUM: In addition, we'd also like the NRC to determine the Susquehanna condition at the time of our Part 21 report versus regulatory requirements. There's a big difference between where they are today and where they will be in the future and where they were at the time of our report.

Our contention is that PP&L should have reported this, not us. We shouldn't have had to have gone through this. We'd like you to confirm that.

We'd also like you to investigate PP&L's actual response to our discovery before we made the Part 21 report. We'll have to go into the details of that at a later date. We'd also like the NRC to investigate the counter-to-nuclear safety management attitudes at PP&L. We'll have to go into detail on that at a later date, also.

We'd like the NRC to bring this issue to closure as soon as practical. On a personal note, this has taken up a lot of time. I have another job and it's -- I just don't have time to be involved in this, but I have to.

Lastly, we'd like the NRC not to take our positions or anything we've said at face value, but, on the other side of the coin, we don't want you to take what PP&L has said at face value, either.

We feel confident that if you look at the facts and these facts are examined impartially, our conclusions will be supported.

To go over those conclusions, first is that a design basis LOCA results in, directly results in a boiling spent fuel pool. This is not a separate issue. It's not a combination of two separate events. It's one event that results directly in a boiling spent fuel pool.

Also, our conclusion is that there is an inability to provide makeup to the boiling spent fuel pool and this results in meltdown of irradiated fuel outside of primary containment. Also, that a boiling spent fuel pool results in failure of safety-related equipment in the reactor building and also the loss of secondary containment for that building.

Loss of safety-related equipment in the building results in the meltdown of the fuel in the reactor core, failure of the reactor vessel, potential failure of the reactor vessel, and failure of primary containment.

All of these consequences above result in radiological consequences to the public thousands of times greater than is presently allowed by 10 CFR 100. Also, I think you will conclude that PP&L did not and still is not satisfying their legal and ethical responsibilities.

MR. PREVATTE: That's the basis of our formal report.

MR. THADANI: Are there any questions?

[No response.]

MR. THADANI: I want to thank you very much, indeed, because it's very evident that you have spent a great deal of time doing a lot of analysis and probing into a number of areas. I'm very pleased that we've had this discussion.

It's always better, I think, to have face-to-face communication and I definitely appreciate also your having put this information in a fairly systematic fashion. I think this way it's easier to follow each issue in some fashion, making sure you look at all the information.

I'm sure you've spent a great deal of your time doing these studies and we do appreciate that. I assure you that we will take this information, look at it very carefully, as we will the information we receive from -- we've been actually looking at the information from PP&L.

You have identified, at least in my mind, some issues, new issues that at least I wasn't aware of and we're going to take a look at those carefully, also.

You've made a lot of recommendations in terms of how we should proceed. Most of them I basically agree with you on that that's what we ought to really do. The other thing is we would like to keep you informed at some intervals as to how things are moving.

If necessary, maybe we'd ask you to meet with us again if there's a need to clarify things.

MR. PREVATTE: We'd be happy to meet with you at any time.

MR. THADANI: We would move as quickly as is practical on this issue.

MR. LOCHBAUM: Do you have a timeframe that you're shooting at or a schedule?

MR. THADANI: I had honestly hoped that we would have come to closure by now on this issue, but it was critical that we hear you and take time to assess what you're telling us. I don't want to give you a date because I'd rather we do this job right and go at it. As soon as we have better information, we'll be happy to tell you about it.

I thank you also for trying to put it in the safety perspective. I will share with you I think it's not quite as significant risk as you think, but I think we better look at it carefully. I'm not drawing that conclusion, but that's the perception I have at this time. We will go through systematically.

Dick?

MR. CLARK: I think Mr. Thadani expressed the opinion of all of us. We'd really like to thank Don and Dave for taking the time and extending the effort to come in and meet with us today. As he said, I know we're going to carefully consider what you presented.

I'll send both of you a copy of the transcript when it's available.

Before I close this meeting, I'd like to ask if any of the observers or other attendees have any comments or statements that they wanted to make. Identify yourself.

MR. GUNTER: Paul Gunter, Nuclear Information Resource Service. I wonder if the two engineers would provide us with their conclusions of how this would apply to single-unit Mark I reactors. I know your experience could lend -- I understand PP&L had said that the makeup systems, for example, of the second unit would lend itself to a solution.

I'm wondering what your comment is on a single-unit Mark I.

MR. LOCHBAUM: Single-unit Mark I. I'd say it's a complex combination of the age of the plant, the regulations that were in effect at the time the plant was licensed, and who the architect engineer was. There are a number of factors that can effect the outcome.

I guess from my own personal perspective, I don't know of any Mark I plant BWR, Mark II plant BWR that's designed to avoid all these problems. As far as the single unit, you're less vulnerable with a single unit because you have equipment from the adjacent unit that may be able to be used.

So just from a relative risk standpoint, a multi-unit site is better off than a single-unit site.

MR. FLEISHMAN: A multi-unit site is better than a single-unit site.

MR. LOCHBAUM: Right, because there's equipment that can be used on the non-accident unit, potentially.

MR. PREVATTE: Not necessarily, but possibly.

MR. CLARK: Were there any other comments or statements?

[No response.]

MR. CLARK: If not, again, I would like to thank Don and Dave for meeting with us today. With that, this meeting is hereby closed.

[Whereupon, at 3:07 p.m., the meeting was concluded.]

REPORTER'S CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission
in the matter of:

NAME OF PROCEEDING: Susquehanna Spent Fuel

DOCKET NUMBER:

PLACE OF PROCEEDING: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Ian Rothrock

Official Reporter
Ann Riley & Associates, Ltd.

[illegible]