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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTCR SAFEGUARDS

SUBCOMMITTEE ON NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

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NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

The Grand Ballroom
The Hotel Syracuse
500 South Warren Street
Syracuse, New York

Thursday, February 21, 1985

The subcommittee reconvened, pursuant to recess, at 8:30 a.m., Chester Siess, Chairman of the Subcommittee, presiding.

ACRS MEMBERS PRESENT:

C. SIESS, Chairman
J. EBERSOLE

DESIGNATED FEDERAL EMPLOYEEE:

J. McKINLEY
J. SCHIFFGENS

NRC STAFF AND PRESENTERS PRESENT:

A. SCHWENDER
E. WEINKAM
J. LANE
A. ZALLNICK
R. ABBOTT
G. MOYER
M. COLOMB
C. TERRY
M. DURKA
D. PIKE

E. KLEIN
N. RADEMACHER
D. PRACHT
P. VOLZA
R. RAYMOND
J. PERKINS
E. LEACH
M. STOCKNOFF
M. KAMMERON

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PROCEEDINGS

MR. SIESS: The meeting will reconvene.

I understand that the applicant would like to start off by addressing a couple of questions that were raised yesterday that they didn't have all the answers to.

MR. ZALLNICK: Yes, sir. We have a response on Mr. Ebersole's question about the redundancy of the crane and also about the HPCS system.

We will start out with the crane. Mr. Klein, who was up yesterday, will talk about the crane.

MR. KLEIN: Good morning, gentlemen.

My name is Ed Klein. Yesterday we discussing the potential failure of what you might call a mechanical component of the reactor building polar crane. I couldn't remember the NUREG number which governs the design of this crane, and that number is NUREG 0554.

Also yesterday, I stated that we were in full compliance with this NUREG. We are not in full compliance with this NUREG. There are six technical issues which have been discussed in the FSAR. There are no open issues and none of these six issues address single component failure.

And now to discuss mechanical component failure. The redundant main hoist system consists of dual load pass through the hoist gear train, the reeving system and the hoist load block to prevent uncontrolled motion of the load upon failure of any single hoist component.

(Slide.)

This sketch is a planned view of the crane drum, gearing, hoist motor and brakes.

As you can see, the crane is provided with dual gear trains, dual hold brakes and each brake is designed to safely hold a load. The brakes are applied with loss of power.

The drum and the main girders are not redundant, and this design demonstrates that the load can safely be maintained with a single component failure.

MR. EBERSOLE: May I ask a sometimes little obscure question. The main motor has a certain ultimate torque rating. It is positioned in the limits of its travel or load by switches, position switches or torque switches or load switches.

If these switches fail to intercept the power to the motor circuit breaker and the motor goes to its ultimate torque rating at say its uppermost limit, does the motor have sufficient torque to commonly disrupt those two spur gears or do other strain damage to the machinery and drop the load?

MR. KLEIN: I am not sure how to answer your question. The motor has an overspeed control over it and the brakes will lock on overspeed.

MR. EBERSOLE: Right. But when it gets to the end of its travel, it is stopped by switches.

MR. KLEIN: There are dual limit switches on the up travel of that crane that will stop it from traveling and there is a limit switch on the lower.

MR. EBERSOLE: Now that raises the eternal question, do you know when the first limit switch fails in order for you to go repair it and keep redundance?

MR. KLEIN: It is a paddle switch. So I am assume you are going to know when you hit it.

MR. EBERSOLE: You mean by listening? You know what I mean. In any redundant system one must know when the first failure occurs either by output signals or by periodic verification that you have maintained redundancy.

I am only asking if the crane, however, has the ultimate potential to brake itself?

MR. KLEIN: Mile Allen, would like to come up and address specifically, please.

This is Mike Allen from Stone and Webster.

MR. ALLEN: No,it does not. To answer your question on limit switches, there are two limit switches, one set just right after the other, and we do periodically verify their operation.

MR. EBERSOLE: Right.

MR. ALLEN: To answer your question about the motor, if the motor stalls, it will not disrupt the drive train past the bull gear, and if the motor burns out its winding and loses a field due to excessive current and stalling, when those fuses go, the brakes are automatically applied.

MR. EBERSOLE: So the brakes will pick up anything that happens to the motor?

MR. ALLEN: Yes, sir.

MR. EBERSOLE: Thank you. That is fine. Thank you very much.

MR. SIESS: Did you have another item?

MR. ZALLNICK: The other question you had was on the HPCS, and I will ask Mr. Rademacher to address that question.

MR. RADEMACHER: Good morning. I guess, first, that I wanted to discuss a few items relative to HPCS, the high-pressure core spray system, some of the improvements that are designed, and then I will get into the answer to your question relative to the GE letter.

First, service water is constantly running. In our plant service water is service water is service water. It is emergency service water and, therefore, you have a greater assurance that it works. Whereas, if it was just HPCS service water, it would only be checked periodically at that time.

MR. EBERSOLE: I understand.

MR. RADEMACHER: When you add additional redundancy you increase the reliability of HPCS.

MR. EBERSOLE: But does the service constantly run through the jackets of the No. 3 diesel, or is it turned on by valving, or, for that matter, the 1, 2 and 3 diesels?

MR. RADEMACHER: I believe it is deadheaded.

MR. EBERSOLE: Deadheaded, okay. Thank you. That is all right. You needn't pursue that with me. It is deadheaded, and you said you don't have any marine growths.

MR. RADEMACHER: Right.

No. 2, when you add additional redundancy, you increase the reliability of HPCS. Our design provides for redundant service water pumps, six actually, with redundant backup diesel power in lieu of the single service water pump and diesel that you would have if you just had the HPCS system.

As mentioned during the plant tour, we do have the capability to cross-connect the HPCS diesel to the service water pump if it was necessary.

MR. EBERSOLE: That is an electric cross-connect?

MR. RADEMACHER: That is correct.

MR. EBERSOLE: I think the real matter of issue is why should that not be automatic if you have got time to do what you said you were going to do, which is rack a breaker out.

MR. RADEMACHER: Okay, let me go on.

MR. EBERSOLE: Okay.

MR. RADEMACHER: Relative to a station blackout, and that is the loss of all AC, and you assume there is no AC ---

MR. EBERSOLE: Except for the third diesel.

MR. RADEMACHER: Let me explain the way that I have understood it.

For example, on Limerick they assumed for a loss of offsite power in their PRA that all four diesels would be inoperative. And we have three diesels, so we have always assumed that at least internally and based upon our discussions that even if we had the other design we would not take credit for HPCS.

Further, in our study for station blackout, we are using, or will be using our RCIC to provide assurance that we can safely shut down.

MR. EBERSOLE: Is RCIC in any way dependent on AC power such as for environmental controls in the room?

MR. RADEMACHER: We are addressing that as part of our study to determine the heatup of the room and this kind of thing. There is AC power in there to provide room cooling, but I believe we can survive for a period of time without that cooling.

Further, this was not an arbitrary but a conscious decision that we made when we selected the service water system. We performed the failure modes and effects analysis on the service water system and this was reviewed by a detailed Niagara Mohawk design review at the time of its development. And it included our operations and engineering people for a detailed review.

Lastly, we discussed this matter with GE and we asked them to respond in writing. We received a letter on February 12th from GE, and the letter basically indicated that the design meets the intent of what the system requirements are.

MR. EBERSOLE: They then agreed to let it remain depending on the switchover of the pumps, which I understand is manual?

MR. RADEMACHER: Pardon me?

MR. EBERSOLE: You are going to pick up service water by manual transfers?

MR. RADEMACHER: No. There is no service water transfer. You have basically two check valves from either division, division one or division two. The water is always available, and if you have a failure in HPCS it won't affect the other division because there is a check valve, two check valves in there.

MR. EBERSOLE: Well, I guess I don't understand. I understand that when you lose AC power but you retain the high-pressure core spray diesel, invoking the fact that it is independent of the grid, that you don't have any service water, but you go and pull out one breaker and rack in another to get water, is that correct, to utilize the third train of feedwater?

MR. RADEMACHER: If we had a loss of all AC, that is right.

MR. EBERSOLE: So the real crux of it seemed to be have you got time to do that before the diesel overheats?

MR. ZALLNICK: We wouldn't do that before the diesel overheats, Mr. Ebersole. The station blackout procedures currently being evaluated based on our blackout analysis, immediately calls for using RCIC for that event.

MR. EBERSOLE: So you don't claim the third diesel on a station blackout at all?

MR. RADEMACHER: That is correct.

MR. EBERSOLE: Okay. You have the prerogative of trying to do so, but you don't.

MR. RADEMACHER: That is correct.

MR EBERSOLE: I thought you wanted to do it, or I thought GE wanted to do it.

MR. RADEMACHER: They may have done so on other projects, but not on ours.

MR. EBERSOLE: Okay. Thank you. I have the picture.

MR. RADEMACHER: There was just one point that I wanted to clarify, and Mr. Doug Pike, one our Assistant Managers whispered in my ear, so I will let him talk.

MR. PIKE: This is Doug Pike, Assistant Manager of Engineering. On the tour yesterday, Mr. Ebersole, we discussed this. I think we have no procedures in place and we really haven't looked at that as far as the actual capability to do it. It was just well, if that happened this could be a way of getting out of it.

We would have to take a very close look at that to make sure that it was even possible.

MR. EBERSOLE: I understand. Do you remember the old steam driven HPCI. Its thesis was that it was emergency feedwater pump as well as a small break mitigator, and it had of course a degree of independence from AC.

I think the philosophy of GE's putting in this diesel was it was an independent, non-connected to the grid design and in thesis at least a reproductive function of the old, original stream driven HPCI but driven by diesel electric power. That automatically inferred that it was not going to be dependent on ordinary AC circuitry but its own output, and you all have a slight bias to that, which is your design base.

MR. RADEMACHER: That is correct.

MR. EBERSOLE: Thank you.

MR. RADEMACHER: The second question that I had a response to related to the diesel generators as well, and this was on your question regarding steam created by injection on the manifold.

MR. EBERSOLE: By water spray on the running diesel.

MR. RADEMACHER: Correct. I wasn't in your group, so if you describe your scenario again..

MR. EBERSOLE: Let me tell you again what I overheard. In the case of a fire or, for that matter a synthetic energization of the spray system possible by comments events like earthquakes, you spray water on the diesel but it keeps on running. It aspirates air for combustion from the outdoors. So that doesn't bother it. And you keep the room open because you don't have CO2 in it and you just spray water.

I was then told that you spray the water on the hot exhaust system which created a steam environment, but subsequently I was told that the ventilation air throughput was maintained at high speed, and apparently I would argue with you that you would not a excessive steam environment because of the massive air throughput. Is that true?

MR. RADEMACHER: Okay. I guess -- let me ---

MR. EBERSOLE: The reason I was going to this point was I don't believe your equipment in the diesel generator room can stand a high humidity transient environment because of condensation on terminal blocks.

MR. RADEMACHER: Okay. Let me explain.

First off, I will address -- there are two questions the way I understand it. One is if a seismic initiated event caused water spray on the diesel. The way we are designed right now is that that is a pre-action system, and a pre-action system would ---

MR. EBERSOLE: It would take the links. I understand.

MR. RADEMACHER: So we would spray it from ---

MR. EBERSOLE: Right. I have got that.

MR. RADEMACHER: The second case is that we do use NEMA for enclosures in the diesel itself.k

MR. EBERSOLE: You need go no further.

MR. RADEMACHER: Okay.

MR. EBERSOLE: And you don't bore holes on them as some applicants have.

MR. RADEMACHER: I don't believe we do, no.

(Laughter.)

MR. RADEMACHER: Those are all the questions I had responses to.

I think during the presentation of fire protection we will discuss I think your last question relative to combustible controls.

MR. EBERSOLE: Thank you.

MR. ZALLNICK: We are ready to proceed, Dr. Siess.

MR. SIESS: Okay. Then I think we are back on the agenda with Item 13.

MR. ZALLNICK: The presenter for AC/DC Power Systems Reliability is Mr. Douglas Pike.

Mr. Pike has 17 years of BWR experience. He has been an operator at Unit 1 at Fitzpatrick. He is currently in engineering on Unit 2. He is the Assistant Project Engineering Manager.

(Slide.)

MR. PIKE: Good morning, gentlemen. My name is Doug Pike, Assistant Manager in the Project Engineering Department for Unit 2.

(Slide.)

I would like to start my presentation today with our offsite power supply system.

Our design does provide two independent 115 KV power sources for offsite feed into the station. It is ultimately tied to the grid system in the State of New

York. The grid system is tied via some 26 interties to other grid systems in New England, the Pennsylvania/Jersey/Maryland grid and grids in Canada.

(Slide.)

The origination of these 115 KV power sources is our Scriba Station which was specifically built for Unit 2. This station is located about 3,000 feet south of the plant, and we have a one-line diagram up there, a simplified diagram.

It is fed from five separate 345 KV transmissions, one each from each of the generating stations on the site and two feeds from our Volney Substation, which is a few miles south of this station and which is ultimately tied into the grid.

Any one of those feeds can power all station loads. That utilizes the breaker and a half scheme for reliability.

The two 115 KV feeds going into our station come off of the opposite diagonal ends of the station which provides about 400 foot of separation. The control power for those transformers and circuit breakers are fed from two separate and independent DC batteries located at the station.

(Slide.)

The slide on your left, the yellow lines show the routing of the 115 KV lines into the station. The lines are separately routed in. At their widest they are separated by about 500 feet and they obviously converge as they reach the switchyard. The lines are fault protected by primary and backup schemes fed from separate station batteries.

This is a one-line diagram of the the 115 KV switchyard. Our switchyard is segregated by motor operated disconnects and circuit switchers to maintain the independent separation of the offsite feeds.

Those disconnects and circuit switches are interlocked to prevent paralleling of the offsite sources and they are also fed from different station batteries for independence.

MR. EBERSOLE: I wonder if you could tell me why you don't parallel offsite specifically?

MR. PIKE: Simply to maintain their independence so that a common failure can't take both of them out.

MR. EBERSOLE: I see. Okay.

Could you explain the aux boiler?

MR. PIKE: We have electrically heated auxiliary boilers in the plant to provide auxiliary steam sources and that has a pretty high power demand. So we have a separate transformer and feed for that boiler.

MR. EBERSOLE: That is an immersion heated boiler?

MR. PIKE: It is an electric heating element type.

MR. EBERSOLE: Is it located against anything which would be affected by its explosion, its hypothetical explosion?

MR. PIKE: I believe it is located in the turbine building. It is not near any safety related equipment.

MR. EBERSOLE: In recent years it has come to be known that it is much better to keep emergency Class 1-A equipment not tied to the unit output but to the station grid.

MR. PIKE: I am going to come to that.

MR. EBERSOLE: Good. Okay.

(Slide.)

MR. PIKE: We have a picture here of our 115 KV switchyard. Our offsite source A feeds our reserve station service transformer and the auxiliary boiler transformer, and our offsite source B feeds the B reserve station service transformer.

(Slide.)

And I have got a little picture here of those transformers. The big transformer in about the middle of the picture, and I will use this little light gun, that one and then the one over there are the reserve station service transformers. They are separated by the house service transformer and separated by fire walls. So we maintain a separation on the transformers also.

MR. EBERSOLE: There recently has been a spectacular explosion of a main transformer that had fire walls around it. I think they were taller than that, and I believe they were fog protected. But in any case, the fire certainly didn't threaten to crawl or cross the barriers. What sort of a fire extinguishing system does that have?

MR. PIKE: There are a fixed deluge systems on those transformers.

MR. EBERSOLE: What established the height of those barriers, could you tell me? The reason I ask that is I remember the other barrier was about twice as high as the transformer.

MR. PIKE: I don't know personally what the criteria was.

MR. EBERSOLE: Are there any standards for that sort of thing?

MR. PIKE: I don't know.

MR. ZALLNICK: We will get an answer on that for you.

MR. EBERSOLE: Well, certainly the wind must just blow one way.

(Slide.)

MR. PIKE: Now I would like to go into our onsite power sources.

The slide on your left up there is a simplified one-line sketch of the distribution system within the plant.

Just briefly to orient you, we have got our 115 KV source A coming in on the left at the top, the 115 KV source B a little to the right up there, and then the aux boiler transformer on the far right. So those are our 115 KV sources coming in. They go through the transformers and feed the three big 13.8 KV buses, and there are alternate feeds down to the emergency buses, which I will cover later.

(Slide.)

As you can see from the diagram, our offsite source A feeds one of the 13.8 KV buses and it also feeds the auxiliary boiler transformer.

Offsite source B feeds the other 13.8 KV bus through the reserve station service transformer B.

MR. EBERSOLE: Pardon me. I just wanted to ask as you were in this place, that means to me, what I see there, that in fact you ride the station auxiliaries on the main generator output.

MR. PIKE: That is correct. During normal operation all station loads other than the three safety related power boards are fed off of the unit generator.

MR. EBERSOLE: Right. So that means you must in order to maintain a normal shutdown execute a transfer?

MR. PIKE: That is right. Normally on a normal startup or shutdown the operators will manually transfer. Should you lose quickly, you know, suddenly lost a unit generator, there is a fast transfer to offsite power.

MR. EBERSOLE: Well, that is the old style arrangement. It followed down from steam turbine coal burning days. So it didn't matter. But nowadays the common practice is to simply tie the shutdown auxiliaries to the incoming common station service. Most of the new plants are doing that. That is what I wanted to see this.

MR. PIKE: I was going to point out that those 13.8 KV buses are fed from the generator during normal operation.

Then, while it is not shown here, the 13.8 KV buses that distribute power throughout the plant to other 4160 volt switchgear, 600 volt load centers and 66 volt motor control centers.

(Slide.)

Some of the reliability features of our system.

The main and tie breaker control circuits in the station are fed from one of the station's DC batteries.

The feeder breaker control circuits are fed from a separate station battery. The two station battery feeds can be interchanged through manual switching.

Most buses in the plant are sectionalized so that either of those 13.8 KV buses can feed the lower distribution buses through switching. We do have seven uninterruptible power supplies that supply 120 volt AC power to such things as the central lighting, non-safety related instrument and control circuits and the main plank computer and the reactor protection system trip circuits.

Our UPS power supplies have a normal preferred AC source. However, on loss of that source there is an automatic transfer to a backup DC source fed from the station batteries. And in the event that that fails or the power supply needs maintenance, there is a bypass AC source.

(Slide.)

As far as the safety related AC power systems which are shown across the bottom of the left-hand side, again there are three independent divisions of safety related power, Class 1-E equipment, seismically and environmentally qualified and physically and electrically separated.

MR. EBERSOLE: However, one of those three is somewhat compromised by the need for the other one, right?

MR. PIKE: As we have discussed.

MR. EBERSOLE: Yes.

MR. PIKE: Each division has a dedicated 4160 volt bus. Unless you have a loss of offsite power, again offsite power source A normally feeds division one and three buses and offsite source B normally feeds division two. Through switching offsite source B can provide a backup feed to division three and also the auxiliary boiler transformer can provide a backup feed to division one or division two.

Again, each division has its own dedicated diesel generator that provides safety related power under a loss of offsite power or degraded voltage conditions.

MR. EBERSOLE: Could you tell me in sort of a nutshell, you know, one hears you have got seven AC supplies and a number of DC supplies, but one must always ask the question, yes, but in how many cases do they simply converge to one out of two even though you may have six or eight? The functional dependency may converge to one out of two in "X" cases like DC control. Is that true?

MR. PIKE: No. Again, the design is that the divisions will be kept independent and separate from each other. So division one power is separated from division two power and is separated from division three power.

MR. EBERSOLE: Does that include the DC supply for circuit breakers?

MR. PIKE: Yes, sir.

MR. EBERSOLE: So you have one out of three competence, except for this curious business about the water?

MR. PIKE: Yes, sir.

MR. EBERSOLE: Which puts you back in one out of two.

(Slide.)

MR. PIKE: Some more features of our safety related power system. Divisions 1, 2 and 3 buses again feed additional 660 volt and 120 volt distribution systems that aren't shown on the slide.

The division one and two load centers, incidentally, can be supplied through two redundant 100 percent capacity feeders for reliability.

We do have division one and two uninterruptible power supplies that provides power for critical instruments and control circuits with the same type of backup feeds as the non-safety related batteries.

Another feature we have is the division one and two buses can feed the non-safety related stub buses in the absence of a LOCA signal, and we have located on these stub buses equipment that we consider critical for reliable power generation to prevent any kind of equipment damage to non:safety related components, such things as instrument air compressors, dry well coolers, closed loop cooling pumps, control rod drive pumps and UPS power supplies.

MR. EBERSOLE: May I ask you this. In your 120 volt, these are fed by what, inverters?

MR. PIKE: Yes. They are Solid State power supplies.

MR. EBERSOLE: Off of DC?

MR. PIKE: Well, the preferred source is AC, and if that fails, there is an automatic switch to the DC.

MR. EBERSOLE: Right. So the preferred source is normal AC?

MR. PIKE: Well, depending on if you are in the divisional systems, it is divisional AC.

MR. EBERSOLE: And if it is lost, a transfer is made without any cyclic interruption I guess?

MR. PIKE: That is right.

MR. EBERSOLE: Thank you.

(Slide.)

Now I would like to get into our onsite DC power supplies briefly and take a look at our safety related power supplies.

Again, we have three divisions of DC power corresponding to the AC power divisions that are fully Class 1-A seismically and environmentally qualified and separated.

Each division has its own battery and two 100 percent redundant battery chargers that are on line and operating in parallel.

Each battery charger can supply all of the non-UPS loads and recharge a fully discharged battery within 24 hours, and each battery can supply the worst case DC load profiles for two hours with loss of the battery chargers.

MR. EBERSOLE: This battery charger brings up sort of a standard question. What is the ultimate terminal upper voltage that you can get with the batteries if I invoke failure of the control relays that hold it to the normal saturation equalization voltage? Can you burn out the connected DC loads?

MR. PIKE: When we have an equalizing charge on the batteries, we are running them at about 140 volts. Normal is about 125. So they are good up to 140 volts.

MR. EBERSOLE: Well, you can hold 140 volts with a regulator of some sort. If I invoke contacts on the regulator, what is the ultimate terminal voltage of the DC charger, 120 volt?

MR. PIKE: I guess I can't answer that.

MR. RADEMACHER: We will have George Moyer answer that quéstion.

MR. EBERSOLE: I am trying to look into whether you have a potential common mode burnout.

MR. MOYER: My name is George Moyer and I am a Station Shift Supervisor. We have 142 volt trip on the battery chargers which opens up the AC supply breaker to the charger.

MR. EBERSOLE: You have an overvoltage trip, right?

MR. MOYER: Right.

MR. EBERSOLE: Thank you.

(Slide.)

MR. PIKE: Our non-safety related DC power supplies.

We have four batteries and battery chargers that supply 24 volts DC to the neutron monitoring system, two batteries and battery chargers for the normal 125 volt DC station loads and we have a battery and a battery charger dedicated to feed the main plant computer.

Again, our battery chargers can feed all the non-UPS loads and recharge the batteries within 24 hours, and the batteries again can supply their load profiles for two hours with the loss of the chargers.

MR. EBERSOLE: The chargers I gather can charge a discharge battery while they are carrying the load?

MR. PIKE: That is correct.

That concludes my presentation on our power sources.

MR. EBERSOLE: Do you run equalization charges on the batteries with the DC loads remaining connected?

MR. PIKE: Yes.

MR. EBERSOLE: Thank you.

MR. SIESS: Anything else, Jesse?

MR. EBERSOLE: No.

MR. SIESS: Thank you, sir.

Our next item is No. 14, Systems Interactions.

MR. ZALLNICK: The presenter for systems interaction is Mr. Carl Terry.

Mr. Terry has 12 years nuclear experience on Unit 1 and Unit 2. He has worked in quality assurance and engineering and is currently the Manager of Nuclear Engineering.

(Slide.)

MR. TERRY: Good morning. I am Carl Terry.

This morning I would like to provide an overview of what we have done in the area of systems interaction.

(Slide.)

The Nine Mile Point, Unit 2 systems interaction has not been evaluated in a single formal study, but it is something that is considered in virtually all aspects of design.

It is strongly believed that implementation of fundamental and established principles of defense in depth used in the design of nuclear power plants is a primary method of precluding systems interactions problems. This would include inherent design features such as physical separation and functional independence of redundant safety systems, and these principals are considered in virtually all aspects of NMP 2 design.

The significant events which are looked at and included as part of the design base include protection against hazards such as pipe ruptures, missiles, seismic events, fires and flooding.

However, while no single systems interaction study has been performed at Nine Mile Point, Unit 2, numerous analyses have been completed and programs have been implemented which consider certain specific systems interaction concerns and provide further assurance that the overall area of systems interaction is properly evaluated.

By way of overview, systems interaction evaluations typically examine three generic types of interactions. These are functional interactions which involve interconnected systems, spacial interactions basically involving physical impacts of material or components and human interaction, including man-machine interfaces and information interpretation.

In my presentation today I would like to discuss the specific evaluation programs which have been implemented or are being implemented on the Nine Mile 2 project relating to each of the above three categories.

(Slide)

In t. area of functional interactions, examples of evaluations that have been performed at Nine Mile Point, Unit 2 are as follows.

A limited probabilistic risk assessment has been completed, which is based on a full-scale PRA performed at the Grand Gulf Station. The results of this evaluation are included in our environmental report, and I believe there is a limited discussion on this later.

A failure modes and effects analysis has been completed, and this is included in a separate two-volume report as part of our FSAR.

An evaluation is currently in progress relating to the evaluation of control systems failures due to loss of a supply bus. This analysis is being completed in response to Bulletin 7927 and involves joint effort involving our NSSS supplier, General Electric and Stone and Webster.

The methodology for completing this evaluation has been included in response to an NRC question, and I did note in a review of the SER that this methodology has been accepted by the staff.

A related evaluation which is also being performed is examining control systems failures associated with common power sources and common sensor failures. Again, this involves joint efforts between Stone and Webster and General Electric. This analysis methodology has also been provided in response to an NRC question.

Both of these evaluations are to be completed approximately mid-year and both of these evaluations are subject to review by the staff when completed.

Regarding fire protection, a fire hazards analysis has been completed and this is included in the FSAR.

Additionally, a safe shutdown analysis per the requirements of 10 CFR 50, Appendix R, have been completed. This is also included as an Appendix in the ouf FSAR.

The project has implemented and excluded equipment list system which is part of Sone and Webster's standard program. This system provides an excellent method to disseminate problems associated with particular devices and components to all appropriate equipment specifications.

The Nine Mile Point 2 project has imposed more stringent quality programs for procurement of non-safety related equipment. This is done through the use of quality assurance categories 2-A, 2-B and 3.

These programs provide a more thorough

evaluation of vendor quality programs as well as additional shop verification during the manufacture of the components and prior to delivery.

Finally, functional interactions are also considered through piping analyses where transients caused by non-safety related systems failures are evaluated on safety related systems.

An example of such an analysis would be evaluating the transient impacts on a service water system due to the loss of an offsite power event.

MR. SIESS: Excuse me. Did your failure modes of effects analyses extend to the non-safety related equipment?

MR. TERRY: Well, certainly the evaluations that we are looking at under Bulletin 7927 specifically examine both safety related and non-safety related equipment for controls systems and that kind of thing.

In terms of extending the failure modes effects analysis into the actual performance of safety related equipment, I believe it did not, but it does look at of course the impacts of non-safety related systems failures on safety related components in terms of initiating events and that kind of a thing.

MR. SIESS: I don't understand the distinction. If you look at the effect of failure of a non-safety related system on a safety related system, how is that different than your failure modes and effects analysis?

MR. TERRY: What I am saying is in terms of non-safety related systems and in establishing the reliability of systems we did not do a failure mode effects analysis ---

MR. SIESS: I am not talking about a reliability. I am talking about interactions, and one of the interactions that we see most frequently is a non-safety related system whose failure interacts with a system that is safety related.

MR. TERRY: That has been looked at to a degree. It is being looked at even further in these evaluations that we are doing in terms of control systems and that kind of a thing where you do have a definite interaction. What happens in terms of information to the operator and that kind of a thing are being evaluated right now.

MR. SIESS: One of the outstanding areas is seismic, that is equipment that is non-safety related and not qualified seismically, but in the event of a seismic event its failure could affect ---

MR. TERRY: Right. That is specifically looked at in terms of the design. I will be talking about that in a little bit in the next slide. But that is looked at. What I thought you were talking about is the system performance itself and the evaluation of a non-safety related system performance.

MR. SIESS: Going back a little bit you mentioned defense in depth has some built in interaction. The trouble with that is that defense in depth usually is limited to looking at the depth of safety related systems.

MR. TERRY: Exactly, and that is why these additional evaluations.

MR. SIESS: It is the non-safety related systems that frequently interact in a way that wasn't anticipated.

MR. TERRY: Yes, and that type of thing is being looked at both in the 201 program and the control systems.

MR. EBERSOLE: May I ask a couple of questions. I see the excluded equipment list and it brings to mind instantly the horrible case of Salem when they had a "Q" list that didn't include the most important things in the plant, the DD-50 breakers for the scram system.

Where is your included list as a point of beginning that you look at to see to what they might be susceptible, that is the critical equipment for shutdown?

MR. TERRY: Well, we have documented safety related equipment in the FSAR.

MR. EBERSOLE: It is all tabulated?

MR. TERRY: Yes.

MR. EBERSOLE: You all noticed immediately that I haven't read that massive volume, but it is documented, and that is supposed to be a hundred percent, right?

MR. TERRY: These things are something that are very dynamic and on Unit 1 and Unit 2 we have to have methods to maintain our "Q" list in an updated manner.

MR. EBERSOLE: Let me try another one. There is some statistical probability which is thought in the regulatory business to be a rather substantial contributor to core melt, which is total AC power loss.

MR. TERRY: Yes.

MR. EBERSOLE: With this curious third diesel certainly in part compensatory to that if you fix it, let me give you the scenario which complicates that which is a sort of systems interaction.

In the course of a turbine trip and a cascade of the offsite grid, which is very improbable, and I will be the first to agree with whatever number you come up with about the probability of a station blackout, but included in that is a somewhat disastrous interface, a stuck PORV.

If that happens, you bleed the steam off and I don't think your RCIC will work after a while. You will lose the capacity to put water in. And then you dearly would wish that you had that third diesel because that is the only way you are going to cool it. I think that is one of the most substantial contributors, the third diesel.

MR. TERRY: Well, a third diesel or a fourth diesel or a fifth diesel.

MR. EBERSOLE: I am not talking about a standard diesel. I am talking about one isolated from the network and a different design, by the way.

MR. TERRY: Excuse me?

MR. EBERSOLE: A different diesel. It is a smaller one.

MR. TERRY: Yes. But in terms of all of the safety related diesels, they are all independent from the network in that regard.

MR. EBERSOLE: Well, they have tie breakers.

MR. TERRY: Well, they all do.

MR. EBERSOLE: The third one does even, sure.

MR. TERRY: So what I am getting is in my mind I am not so sure as you really would add that much in reliability. As we talked about earlier, and I think it is significant, in looking at the systems that you need most of the time to maintain a plant in a safe operating condition, those are what we looked at in terms of trying to enhance the reliability. Nothing is impossible and anything can happen.

But I think in terms of what we have done in overall reliability, I feel comfortable at least that we have taken the right path. We have substantial reliability on our service water system and we do have two independent diesels to provide that in addition to the HPCS, which, yes, if you had a failure in terms of the other two diesels and a total blackout, you would lose the third also. But, again, you have to remember the other things that have been done to enhance reliability with that.

MR. SIESS: The question Mr. Ebersole is asking, and he shifted gears a little bit and we are now into a PRA type core melt sequence.

MR. TERRY: Yes. We will be talking about that.

MR. SIESS: We have to admit there are PRA core melt sequences.

MR. TERRY: Yes.

MR. SIESS: The issue then becomes a probability, and if the probability is not low enough what can we do to reduce it.

MR. TERRY: Yes.

MR. EBERSOLE: I think it might be argued do you really buy anything with those breaker ties to the third diesel.

MR. TERRY: Well, frankly, there is a complicated logic scheme that would go along with that in terms of we certainly couldn't feed the entire division one or division two bus from our HPCS diesel. So if you were to have the intertie, you would also have to restrict load to the service water pumps. You would have to limit it to one pump and it would be quite complicated. I really don't think in terms of overall reliability you would find that you would gain that much. That is an opinion, but I am pretty sure on that.

MR. EBERSOLE: Did you find that you gained a lot by putting those breaker ties to the other diesels on the third diesel bus by inviting cascade failure of the third diesel?

MR. TERRY: I don't know as we are inviting cascade failure.

MR. EBERSOLE: You do if you close the breakers without coordinating the unloading of the other buses.

MR. TERRY: Yes, but there is protective relaying for that.

MR. EBERSOLE: Did you do a PRA on that?

MR. TERRY: I can't address that.

MR. EBERSOLE: I mean it looks good on the surface until you remember you can cascade it to failure.

MR. SIESS: The only way you can evaluate what you add by adding diesels is through a PRA, and if you put three diesels, is that better than two, or is four better than three. And if you have done it, the immediate problem you get into is what assumptions you make about common mode failures.

MR. TERRY: I agree with that, that the more that is involved, the more difficult it becomes to ---

MR. SIESS: What what can you gain by diversity, and then if you get into the seismic PRA, you have got another problem. But you really can't answer these questions without a PRA and the assumptions that go along with it.

MR. ZALLNICK: I think Mr. Rademacher has a comment.

MR. RADEMACHER: We will be talking about PRA later, and we will be glad to address those kind of questions then.

(Slide.)

MR. TERRY: The next type of interaction I would like to review are spacial interactions and examples of evaluations which have been performed on the project relating to spacial interactions are as follows.

High-energy line break evaluations have been performed which assess damage due to pipe whip and spray impact. The results of these evaluations are included as part of the FSAR.

MR. EBERSOLE: May I ask you a question referring to kind of a dark place in this process, in this topic.

If you look back in it you will find a basis for your analysis is the hypothesis that certainly redundant equipment will close off high-energy line breaks rather than permit sustained discharge. That is one of the root theses.

Yet, I will pick two high-pressure lines. The reactor water cleanup is one, and what will be another, well, let's say the steam supplied through the RCIC. Now look deeply into the valve rationale, the design and the QA and reliability of the valves and tell me that you have found that in fact they are designed to cope with closure under dynamic loads of flowing steam or water. They are not subject to the outboard environmental impact in the event the hypothetical line break is near to them, another degradation event. They are periodically checked to see that whatever initial margin of force to close against these hydrodynamic loads has been maintained, if you ever established it in the first place.

Again, I am talking about PRA type things, and come with an answer about how much you believe in this more or less arbitrary hypothesis of the efficacy of simple redundancy against all of these impacts that I mentioned and then tell me it doesn't matter if this discharge is sustained because the environmental impact will be coped with by the qualification of the equipment. But I don't think you can do that.

I think your environmental qualification is based on the hypothesis of rapid closure.k

MR. TERRY: Doug, do you have anything to say in terms of the EQ program itself from what we do look at?

MR. EBERSOLE: It is based on closing.

MR. PIKE: I believe that when you establish the accident environments you assume that your isolation valves operate.

MR. EBERSOLE: Yes, I am sure of that.

MR. PIKE: However, if you are talking in general about the effects of fluid transients on active components, we do have a program that will identify those components that see these fluid transients and then we will address whatever needs to be done to show that they will perform their function.

MR. EBERSOLE: It gets back to a reliability under duress, and you say you have a program going at this time?

MR. PIKE: That is correct.

MR. EBERSOLE: Could you comment on what the status of it is now?

MR. PIKE: Well, I can give you some examples of things we have done. In fact, yesterday it was noted on the feedwater check valves what has been done to date on those.

Our containment purge valves have been shown to be able to close against dry well accident pressures. Main steam isolation valves, the ball type valve, there was an actual test performed on an eight-inch valve that actually showed that it closed under steam flow conditions while being seismically excited. So these are some examples of things that have been done in this area.

MR. EBERSOLE: It is interesting to note that the main steam isolation valve, since main steam discharges into the turbine hall and then to outer space, that it wouldn't hurt much critical anyway. So it is these discharges into the aux building that count in the context that that becomes regressive to sustaining equipment after the accident occurs. Do you follow me?

MR. PIKE: Yes. Obviously a line break in the secondary containment is more critical than one in the turbine building as far as equipment operation, critical equipment operation.

MR. EBERSOLE: Right.

MR. SIESS: Let me get something clarified. As far as high-energy line breaks are concerned on pipe whip and spray impact, that does not assume any valves close?

MR. TERRY: No, not at the time of the break.

MR. SIESS: On moderate-energy line breaks for exposure to spray I assume that doesn't assume any valves close.

MR. TERRY: That is correct.

MR. SIESS: What about flooding?

MR. TERRY: Well, each of the evaluations are different in terms of the line break, but certainly there are credits taken in certain cases for certain actions to be undertaken after certain periods of time, be it closing valves or other actions.

MR. SIESS: Your interaction analysis for flooding assumes that somewhere you will turn the water off?

MR. RADEMACHER: That is correct.

MR. TERRY: Yes.

MR. RADEMACHER: About 30 minutes after the event in most cases.

MR. EBERSOLE: Let me comment on the statistics I am going to hear in a bit about the PRA, which will include these nasty things called valves. Those statistics have been built on the basis of punching a signal at the valve and watching it go from red to green, essentially a bi-stable state at zero load, like a barn door swinging in the wind not with a load on it. It gives you a false confidence that the valves are reliable and that is what shows in the records.

I would be interested in how you alter that hypothetical reliability to one more near reality.

MR. SIESS: Let's save that for the PRA part.

MR. TERRY: We will give Norm time to think about it.

In terms of, as we stated just a minute ago, moderate-energy line breaks, we have evaluated equipment impacts due to exposure to spray and flooding, and this evaluation, the results of this evaluation are also included in the FSAR.

Control systems failures due to high-energy line break are also being examined to address concerns of I&E Information Notice 79.22, an NRC question that we have gotten in this regard.

Completion of this evaluation in response to the question are currently being scheduled for the spring of this year.

A separate report has been completed, which is referenced in response to an NRC question in which we have discussed to some degree yesterday and this morning in relation to heavy loads.

A more detailed discussion is planned relating to equipment qualification, but it should be noted here that spacial interactions are a prime consideration in this program.

Submittal of the actual results of the Nine Mile Point, Unit 2 qualification results is planned for 1985.

MR. EBERSOLE: You are on equipment qualification now, aren't you?

MR. SIESS: Yes.

MR. EBERSOLE: Let me comment on that. We have found some applicants who have used these NEMA type four boxes and then discovered to their consternation they couldn't stand external pressure and yet they were going to be in a pressurized environment like a dry well or a containment. And rather than get equipment which could sustain that external pressure, they simply bored holes in the, the holes I referred to earlier.

This produces, of course, invalidation of the NEMP type four characterization and leads to the picture that in a transient, which includes steam, high humidity and the initial cool condition of the terminal boards, inevitably you have a condensation function on terminal boards for which the face to ground clearance certainly with a little dirt in it looks like a shortcircuit or a strong leak on sometimes milliampere circuitry.

There is a neat balance in this business of whether you can tolerate the leakage current or even in the high voltage case certainly not the shortcircuit.

What has been your approach to this, the clean one being to tell me you seal these things and keep them sealed and you don't bore holes in them, or you don't use terminal boards at all and you have taped equivalents.

MR. TERRY: Well, I believe we use the junction boxes.

Doug, do you have information on that?

MR. PIKE: We don't use terminal boards inside the primary containment. Outside the primary containment most in the reactor building, if we find that that specific box is subjected to a steam environment, then we would use qualified splices rather than terminal boards.

MR. EBERSOLE: Right. But that steam environment would be predicated on the thesis that these valves would close?

MR. PIKE: That is correct.

MR. TERRY: An evaluation of internally and externally generated missiles has been completed, and this is also included in the FSAR.

Nine Mile Point, Unit 2 has been designed to meet the electrical separation requirements of Reg. Guide 175 and we talked about electrical separation a little bit earlier.

And, finally, implementation of the seismic category two or category one requirements of Regulatory Guide 1.29 specifically evaluates spacial interactions concerns relating to damage of safety related components during a seismic event by non-safety related equipment or components.

MR. SIESS: Leave that up just a minute. On your first slide you had a number of things that I think go beyond what is required by the standard review plan, the PRA, the FEMA, some of your comments about your QA for non-safety related items and so forth.

On this slide it seems to me that all of these items are things that are now required by the standard review plan. Am I correct?

MR. TERRY: I believe so, yes.

MR. SIESS: Now, as I recall, in some of the studies that research has had made on systems interactions, they were looking to see to what extent the current requirements of the standard review plan lead to avoidance of system interactions and they concluded that there were quite a few things that did. They weren't called systems interactions, but they worked in that direction, and that is the kinds of things we see on this list, right, and these are things that are not particularly unique to your design?

MR. TERRY: Certainly the performance of these evaluations is not particularly unique. I can't really say also that what we are doing is particularly unique. On a regular basis bulletins, information audits and circulars come out that address problems related to this area.

MR. SIESS: Some of the things you indicated on the previous slide are things that I don't recall having seen done on some of the previous applications that we reviewed.

We can come back to that. I didn't realize the PRA as required. It that what, an NTOL requirement, the PRA?

MR. RADEMACHER: Yes. There was a requirement for a near-term operating license to provide the environmental assessment of the effects of severe accidents.

MR. SIESS: We will come back to that. But the FEMA is not required, is it?

MR. RADEMACHER: Excuse me, the FEMA is required by Regulatory Guide 170, Rev. 3. So earlier plants were not required to do that.

MR. SIESS: What is the title of 170?

MR. RADEMACHER: Standard Content and Format for the FSAR.

MR. SIESS: Oh, okay. The FEMA is required in the standard review plan?

MR. ZALLNICK: Under the standard format and content, Reg. Guide 170, not the standard review plan.

MR. SIESS: That is the outline for the FSAR?

MR. ZALLNICK: Yes, sir.

MR. SIESS: What chapter?

MR. RADEMACHER: We have two separate books. It is not a chapter per se.

MR. SIESS: Okay. I didn't realize that. This is an overall FEMA?

MR. RADEMACHER: I believe there was also an addition, NSOA, which is normally performed for GE plants. That is a safety analysis performed on a system basis. The FEMA is a component level evaluation and it includes systems as well.

MR. SIESS: Thank you. I learned something. I may have to start reading FSAR's, if I could find enough time.

(Laughter.)

I think this one is 17 volumes; is that correct?

MR. ZALLNICK: Thirty-eight.

(Laughter.)

(Slide.)

MR. TERRY: The final and third area I wanted to talk about in terms of types of system interactions are human interactions, and human interactions are something that have been considered throughout the design of Nine Mile Point, Unit 2.

As indicated in previous presentations, we have had extensive involvement of our operating plant personnel in review of design layouts. I think you saw some of this that was done in terms of model reviews and other things during your plant tour yesterday.

MR. EBERSOLE: May I ask something about that?

MR. TERRY: Yes.

MR. EBERSOLE: Where is the human interaction as particular system designers decide they want to display information on their system in the control room and they stick up a bunch of enunciator windows and indicating lights and dozens of these people do that to produce an absolutely mind boggling flow of information to the operator who has been forgotten.

MR. TERRY: That is not true on Nine Mile 2.

MR. EBERSOLE: Tell me why it isn't true.

MR. TERRY: It is because our operating people have reviewed the enunciator layouts and what is going to be on there and where it is going to be located. They have been included in that. We did a specific review of that, what would be enunciated and what would not. Of course, there are multiple layers of enunciation. In other words, one light indicates various problems and ---

MR. EBERSOLE: But now let me compound it a little bit. A lot of that, in fact most of it, is non-seismic and non-whatever. It is intermixed on common cable trays and so forth. So it is subject, as I say, to fire malfunctions. Now tell me what fraction of this massive flow of information into the operator's brain can be identified to the exclusion of others so he can home in on a safe shutdown?

MR. SIESS: SPDS.

MR. EBERSOLE: SPDS is equally unqualified.

MR. RADEMACHER: As mentioned during the simulator tour, we have a set of parameters that meet Regulatory Guide 1.97 for safe shutdown, which is Class 1-E equipment and they are uniquely identified on the panel boards for operator identification.

MR. EBERSOLE: How are they uniquely identified? Can you tell me?

MR. RADEMACHER: I think they either have an orange or a red marker around them. I can't remember.

MR. EBERSOLE: There was no attempt to localize them in one place, was there, like you do ECCS?

MR. RADEMACHER: I think the indications are near the equipment that they serve, but I would have to confirm that with our operators.

MR. EBERSOLE: It has been interested to see the ECCS lumped in one place, like an airplane panel, but all these scattered circuit elements and indicators for the critical shutdown function are in fact scattered all over the place.

MR. RADEMACHER: I will let Doug Pike answer that question. I believe he can address it.

MR. PIKE: Generally the indicators are on the main bench boards with their systems. However, we do have an independent post-accident monitoring panel that has recorders on it fed from a redundant channel. So they are grouped on that panel.

MR. EBERSOLE: In the context in which you are speaking, what is an accident? Is it a fire?

MR. RADEMACHER: It is a loss-of-coolant accident.

MR. EBERSOLE: I know it. That is the problem. That is the only accident that we really in an organized way have approached. Yet, that is not going to be the accident, and this is the problem. Well, that comes for later generations.

MR. SIESS: I am sorry, are you asking whether that instrumentation will be there after a fire?

MR. EBERSOLE: Yes.

MR. SIESS: Will Reg.. Guide 1.97, instrumentation, be there after a fire?

MR. EBERSOLE: No.

MR. RADEMACHER: As Doug mentioned, I believe that if you had a fire in one panel and you were capable of remaining within the control room, you could go to the other division which has the same equipment on the other panel. For example, if you wanted to use shutdown Lipsy or shutdown coolant ---

MR. EBERSOLE: The divisional fire.

MR. RADEMACHER: Yes. And if you had a fire that wiped out the whole control room, you would go to the remote shutdown panel, which you saw ---

MR. EBERSOLE: That is probably the best organized panel you have got for shutdown.

MR. SIESS: Well, that is what is it for.

MR. TERRY: Okay. Additionally, the same people that have been involved in the review of the plant layout have been involved in the human factors control room design review, which is currently being conducted, and these same personnel, or not the same personnel at least in all cases, but our operating personnel have been involved in control room panel mockup reviews during the initial conceptual phases of the control room design.

I would add also that in terms of the human factors review, while the control room was being staged in San Jose, we did perform more or less an intermediate control room design review in order to identify any changes that might be necessary and implement those prior to delivery of the panels.

Overall it is felt that the systems interaction related evaluations I have just discussed provide additional assurance that systems interaction concerns are addressed.

Furthermore, while changes have resulted, the result of virtually all of the above programs or evaluations, it is not felt that the results of the evaluations are indicative of major deficiencies in the Nine Mile 2 design or in terms of implementing the established principles of defense in depth we talked about earlier.

(Slide.)

Numerous programs exist to assure that systems interaction concerns are properly implemented both in the design process and in the physical installation.

Design review testing and inspection programs provide assurance of implementation of systems interaction considerations. For example, in the design process multidiscipline review of design documents and independent design review are used to assure incorporation of systems interaction considerations in the design outputs, primarily specifications and drawings.

Furthermore, preoperational testing provides actual simulation of accident scenarios and specifically verifies many of the systems interaction interfaces.

Additionally thermal growth and vibration monitoring performed during startup testing further verifies the adequacy of the installation.

As part of the implementation of the seismic two over one program and the thermal growth verification program, actual walk-downs are performed to assure that physical requirements are met.

To ensure that designs are kept up to date ongoing reviews and evaluations are performed of current problems and concerns with are identified by the NRC and industry in documents such as NUREGs, bulletins, circulars, SOERs form INPO, et cetera.

Finally, I would like to mention the Stone and Webster engineering assurance program, including its technical audit program, which is currently being reviewed with the NRC and may be an acceptable alternative to an independent design verification program.

Niagara Mohawk engineering personnel also have been extensively involved in review of the design through a formal design review process which is proceduralized and was originally included as part of our PSAR.

All of the above provide assurance that systems interaction considerations are implemented in both design documents and physical installations.

MR. SIESS: You mentioned that some changes were required. Could you give any examples of say a change that was required by the design review and/or a change that was required as a result of a walk-down?

MR. TERRY: The walk-downs themselves have not been performed to a large degree. These walk-downs, some of them are starting now, but most of those are going to be involved when the plant physical design is completed. For example, the two over one walk-down is going to be done when that area is basically completed in terms of physical installation.

In terms of design review, though, just to give you an example, on the Niagara Mohawk design review we basically went through three phases of review. We had initial conceptual type reviews that were performed.

MR. SIESS: What I asked was not what you did, but any change, just an example of a change that resulted from this.

MR. TERRY: Let's see. There were hundreds of changes that resulted from the design review process.

MR. SIESS: I am talking about system interactions. That is the subject. A system interaction that was discovered in a design review and it required a change.

MR. TERRY: I am sure if I went through the list there would be a number of them, but ---

MR. SIESS: Well, that is all right. If you think of it later, let me know.

MR. TERRY: Doug, do you have any that come to mind right now?

MR. PIKE: I guess I can't think of anything that you would call a systems interaction. One of the big ones that sticks in my mind was the steam supply to the RHR heat exchanger for the isolation cooling mode of that system.

We felt, based on our operating experience at the Fitzpatrick plant that it was not adequately sloped and drained to preclude water hammer, if that system had ever been put into service.

As a result of that, we have made changes to that system to improve the ability to drain that system prior to putting it in service.

MR. SIESS: Well, I wouldn't call that a systems interaction.

MR. PIKE: No, I understand.

MR. EBERSOLE: Let me try one. I was admiring the Limerick design which anticipated failure of these valves I spoke about earlier, and it is so compartmentalized the steam lines and water lines, such that if a prolonged discharge occurred, true it would destroy the equipment in that compartment, which was a part of the destructive process anyway, but it would be confined in some chase, so to speak, and be discharged to atmosphere, much as it would be in the turbine hall. It is kind of a forward looking compartmentalization process. Now did you do that?

MR. TERRY: Well, yes, in terms of compartmentalization on ECCS equipment and things of that nature, yes. As a matter of fact, in the next presentation, we will be talking about that through the use of the auxiliary bays and that kind of a thing.

MR. EBERSOLE: Great.

MR. TERRY: So that definitely has been done. That was done a long time ago.

MR. EBERSOLE: Well, that was a rather reliable escape form the hypothetical valve failure.

MR. TERRY: Yes.

MR. SIESS: Any other questions?

MR. EBERSOLE: No.

MR. SIESS: Thank you.

I think we will try to schedule the break a little closer to the scheduled time. So we will go on with the next item.

MR. ZALLNICK: Mr. Terry will make the presentation on decay heat removal also.

(Slide.)

MR. TERRY: Today I would like to have a brief discussion relating to decay heat removal. I will be providing a brief summary of the systems involved in decay heat removal, but the concentration of the presentation will be on design enhancements which have been implemented at Nine Mile Point, Unit 2.

(Slide.)

Just by way of a very quick overview, the next slide delineates those systems which are involved in decay heat removal. These systems are reactor core isolation cooling, the residual heat removal system, which of course has multiple modes of operation, including suppression pool cooling, steam condensing, shutdown cooling, alternate shutdown cooling, low pressure coolant injection and containment spray.

MR. EBERSOLE: I have got a little problem with the caption. Only one of the systems up there gets heat out of the containment, the second one.

MR. TERRY: Yes.

MR. EBERSOLE: So it is really core decay heat removal.

MR. TERRY: Yes, but in order to get the heat out you have to transfer it from the vessel to the pool.

MR. EBERSOLE: All right. Do you have an equivalent slide on containment heat removal?

MR. TERRY: Containment heat removal?

MR. EBERSOLE: After you get in the suppression pool how are you going to get it out?

MR. TERRY: Well, the primary method that we have is pool cooling, suppression pool cooling.

MR. EBERSOLE: RSR, that is one system.

MR. TERRY: Yes.

MR. EBERSOLE: Go ahead.

MR. TERRY: In terms of safety grade systems.

MR. EBERSOLE: Yes, I understand.

MR. TERRY: The other three systems are high-pressure core spray, low-pressure core spray and automatic depressurization system.

(Slide.)

As can be seen, Nine Mile Point, Unit 2 has similar normal and emergency decay heat removal systems to other GE BWR/5s.

What I would next like to review are certain enhancements which have been implemented relating to reactor building design and equipment location which we believe contribute to improved overall maintainability and reliability of these systems.

Another item I would just like to mention here is that Niagara Mohawk's specified that GE provide 1.15 service factor motors for use on RHR and LPCS pumps.

MR. EBERSOLE: If you hadn't done that, what would have gotten?

MR. TERRY: 1.0 service factor motors.

MR. EBERSOLE: You mean they don't put as much in that as they do a washing machine motor?

MR. TERRY: I can't answer that, but all I can tell you is that we specifically specified that for Nine Mile 2.

MR. EBERSOLE: That is a standard number for utility apparatus?

MR. TERRY: Yes, in terms of what Niagara Mohawk would normally buy, that is true, but in terms of the NSSS supply, that is not a problem.

MR. EBERSOLE: You are giving me a bad thought that they skin down these critical motors down to a 1.0 as a standard practice. Is that true?

MR. TERRY: The standard design is a 1.0 service factor motor.

MR. EBERSOLE: That is very interesting. You can mark that, Jeff.

(Slide.)

MR. TERRY: The containment design at Nine Mile Point, Unit 2 represents what we believe is an enhancement in a traditional Mark II containment design. It is further representative of Niagara Mohawk's design philosophy to provide additional space for operability and maintainability. The results of this philosophy are reflected in Nine Mile Point, Unit 1 and other Niagara Mohawk generating stations which were designed by Niagara Mohawk.

The reactor building at Nine Mile Point, Unit 2 has been enlarged by the addition of North and South auxiliary bays. These auxiliary bays extend from elevation 175, or the reactor mat, to final grade elevation, 261.

The addition of these auxiliary bays relieves congestion that is typical inside most facilities. It is also felt that the auxiliary bays enhance reliability of the RHR and ECCS equipment by permitting distinct isolation compartments.

MR. SIESS: Does that first bullet mean that the Mark II containment at Nine Mile Point, Unit 2 is larger than those at the other plants?

MR. TERRY: Yes.

MR. SIESS: Larger in which direction, the diameter?

MR. TERRY: The primary containment is about two feet larger in diameter. I will be covering that, but that is what it is.

MR. EBERSOLE: May I ask sort of a fundamental question. How many trains of RHR have you gotten in the context of motors and exchangers?

MR. TERRY: There are two heat exchangers and three motors.

MR. EBERSOLE: Now remember the original old doughnut design had four trains, but it took all three or four in the initial stages of operation to get the heat out, but then you could regress down to one later. This permitted the thesis that you didn't need maintainability and that in the long term you would have at least one left.

You know, the single failure criteria was based on the notion that you were really talking about a point in time very short, like a scram. It didn't include the notion that you had to keep running for three months.

So one then begins to invoke failures in time and is it adequate to have a single failure and then ride on one pump for three months.

That bring up the notion are you going to repair after contamination due to an accident and what is your logic here? Do you compartmentalize and drain to permit subsequent repair in case you have initial failures at the beginning point of an accident?

MR. TERRY: Well, certainly these are areas that we could get into in terms of the pumps themselves and the auxiliary bays.

MR. EBERSOLE: If they had been handling contaminated coolant, could you scour them out and go in and fix them?

MR. TERRY: I can't really address exactly what can be done there .

MR. EBERSOLE: It gets particularly interesting

MR. EBERSOLE: It gets particularly interesting when you have only two trains.

MR. TERRY: Yes.

MR. EBERSOLE: And here it is somewhat in between.

MR. TERRY: Yes.

MR. EBERSOLE: Have you given any thought to how long you are going to last for three months on three trains?

MR. RADEMACHER: I have an answer to your question.

MR. EBERSOLE: All right.

MR. RADEMACHER: Yes, we have. In our EQ program basically we qualify the equipment long enough so that the doses after a cleanout of the RHR system we could go into the auxiliary bays and repair that equipment and put that in service and go to the other aux bay and repair that one.

MR. EBERSOLE: Okay, fine. Thank you.

(Slide.)

MR. TERRY: The slide on my left provides a schematic view of the floor plan for the reactor building at elevation 175. I think it was fortunate yesterday that you had a chance to look at the auxiliary bays in the model so you can appreciate more in elevation view just what they look like.

Note that the equipment included in the north auxiliary bays are the LPCS pump and RHR pump and heat exchanger bay. In the south auxiliary bay are the RHR pump and heat exchanger, loop B, and RHR pump, loop C.

Finally, within the confines of what would be the normal bounds of a typical Mark II reactor building are the HPCS pump and motor and the RCIC turbine and pump.

I believe that a view of these slides clearly shows the advantage of the auxiliary bays in terms of allowing additional space for equipment and additional capability in terms of containability and operability.

(Slide)

The Nine Mile 2 design has been further enhanced to prevent loss of NPSH due to decay heat removal due to lowering of suppression pool level.

Flood troughs are included which segregates suction line leakage into watertight compartment houses. You can see, and I will have an elevation view in a minute, but you can see here in plan view where the flood troughs are located.

(Slide.)

The next slide on my left is an elevation view of the flood trough installation. As can be seen, leakage form a suction line is collected into a sump and control room enunciation is provided if flow to these sumps exceeds

10 GPM.

What this system allows for is maximum credible leak detection within two minutes of such leakage occurring. It should be noted that this maximum credible leak is calculated based upon NRC mechanical branch technical position 31 for moderate energy systems.

To bring this totally into perspective, assuming leak isolation, takes one and a half hours between the time of enunciation, associated investigation of the problem as well as isolation. This maximum result in water loss represents only seven inches of suppression pool level.

(Slide.)

I would also like to mention certain enhancements that have been made to the suppression pool and primary containment at Nine Mile Point, Unit 2. As I mentioned earlier, the primary containment diameter has been increased by approximately two feet which provides for an enlarged primary containment volume which aids in reducing congestion and increasing the total available suppression pool water inventory.

Additionally, the entire suppression pool is lined with stainless steel which both aids in improving cleanliness of the water therein as well as precluding long-term degradation of the pool itself.

This basically concludes my presentation on the enhancements at Nine Mile 2 associated with decay heat removal.

MR. EBERSOLE: May I ask, in view of the fact that the suppression pool heats up rather fast relevant to the concrete that contains it and it is lined with a skin of stainless steel, how do you handle the relative movement of the stainless steel skin which expands and the concrete which doesn't? You know, there is a rather striking thermal grade.

MR. TERRY: Well, first off, it is a metal lined pool and the pool itself is actually a clad stainless. It is a carbon steel with about a 1/8th inch clad stainless in that regard. So that is typically really of other plants.

MR. EBERSOLE: Oh, I didn't know.

MR. TERRY: Yes. The floor plates themselves are stainless, but the liner going up the pool is actually a stainless clad material.

MR. EBERSOLE: And you have accounted for the most severe thermal gradiant on the most rapid heatup.

MR. TERRY: Yes.

MR. KLEIN: That is controlled by the spacing of our studs that hold that to it. The closer they are, the more suppression they can take when you get ---

MR. EBERSOLE: Do you get a little buckling?

MR. KLEIN: The spacing of the studs will control that so that you don't get buckling.

MR. EBERSOLE: Tell me, what normally cools the dry well and the main pump seals? There used to be a system called RBCCW. What does it now?

MR. TERRY: The dry well cooling is normally part of the reactor building closed loop cooling system. The unit coolers inside containment can't be fed with service water.

MR. EBERSOLE: But they are normally on a treated water circuit?

MR. TERRY: Yes, it would be treated water. The reactor building closed loop cooling is a normal feed.

MR. EBERSOLE: So you have in essence the equivalent of a component cooling on PWRs for reactor building cooling, dry well cooling? It is a closed treated loop?

MR. PRACHT: The reactor building closed loop system is nothing more or less than demineralized water. There is no treatment to it in that respect. We found through Nine Mile 1 operation that it has been very successful not to have to do any actual treatment of the water. So it is just a closed loop in and out of the dry well with the unit cooler.

MR. EBERSOLE: What is the basic reason it is a closed loop than a standard cooling loop using service water?

MR. PRACHT: Cleanliness. Long-term fouling is effectively eliminated as far as that internal loop. Any fouling that would occur can be picked up rather quickly in the main heat exchanger outside. It is easy to maintain and it gives us a lot better reliability.

MR. EBERSOLE: Why don't you have to say the same thing about the diesel plant?

MR. PRACHT: I am sorry?

MR. EBERSOLE: Why don't you have to say the same thing about the diesel plant?

MR. PRACHT: Well, in a sense the diesel plant is the same in that you have a jacket water cooler in which you also have an internal loop. The internal is a closed loop, but the direct exchange to the ultimate heat sink is service water.

MR. EBERSOLE: Right. Thank you. I get the picture.

MR. SIESS: Anything else, Jesse?

MR. EBERSOLE: No.

MR. SIESS: Thank you.

The next item has to do with the containment and the staff experts on that are not here. So I think what I would like to do is go on to Item 17 and following that we will have a break.

MR. EBERSOLE: Chet, I forgot one thing.

MR. SIESS: Go ahead, Jesse.

MR. EBERSOLE: We were on decay heat removal. At this point I had a notation here to bring up the topic which I would like to have you talk about that I referred to earlier whose concept was envisioned some 15-odd years ago, and we recently found Limerick is going to sort of patch together this system and the ABWR and perhaps even GESSAR 2 will use it.

It becomes interesting, according to how you designed it, in virtually any kind of decay heat removal malfunction, and it is called in its final stage a UPPS system, ultimate plant protection. I don't know what Limerick is going to call it, but it is a patched up version of the formalized design which GE is developing for ABWR.

Basically it is no more than opening the pressure vessel, the SRV, some fraction of the total number by gas or whatever, providing an independent probably engine driven source of low pressure water to keep the fuel covered, and it permits the steaming of the vessel to the dry side of the dry well, allows ultimate heatup of the suppression pool and transfers steam to the back side where it is passed to atmosphere prior to core damage. Thus, it is a preventive system. It is not a mitigating. It mitigates accidents, but it doesn't mitigate core damage. And it is so simple that you can easily qualify it for virtually any kind of a particular objective you want, whether it be fire, seismic or sabotage or whatever.

I would just like to know to what degree you are coupled to that effort in the context of reviewing Limerick, Grand Gulf, et cetera. There are many plants that are looking at this because of its fantastic simplicity and apparently the consequential reliability that it might have.

MR. ZALLNICK: Mr. Carl Terry, the Manger of Nuclear Engineering will respond to that.

MR. TERRY: Where we are on that, we are familiar with basically what is done in the UPPS system. What we have done up to this point is we have examined what it would take, first off, to vent the containment. Right now if we were to do that, there are some modifications that would need to be done in terms of containment purge and being able to actuate those AOVs under a loss of power. That could be done.

The other thing that is being looked at right now, as we indicated previously, we are doing a station blackout analysis. One of the things that GE is looking in that evaluation are the capabilities of our fire pumps to be able to provide water to the vessel utilizing this type of a system.

Additionally, we are taking a look at what kind of cross-connections could be made in order to tie the fire system into say an RCIC injection line or some other injection.

MR. EBERSOLE: So you have an active effort to pursue that?

MR. TERRY: Oh, yes. Like I say, we will be getting information in terms of what needs to be done and what can be done on that system probably about mid year.

MR. EBERSOLE: One of the major advantages of it is of course it is highly comprehensible in comparison to a decay heat removal train which is dependent on a daisy chain of 25-odd elements. And certainly almost anybody can understand how this can work, possibly including the public, which I am confident do not understand how this thing is done now.

MR. SIESS: Since this type of system is not required by any of the NRC's regulations at this time, are you looking at this from the point of view of protecting your $5 billion investment, or from the point of view of protecting the health and safety of the public or both?

MR. TERRY: Well, really we are looking at it more from the perspective that Dr. Ebersole indicated, which is to be well aware of what is being done in this area and also what it would take to implement such a thing at the plant.

Frankly, in terms of evaluating it for implementation, we would have to review with the Commission the fact that it is one of our scenarios and we would be venting the containment. You know, you are relying upon the filter effect through the suppression pool to clean things up and that would have to be reviewed.

Additionally, there would be a need for analyses in terms of just when do you start to vent in terms of actuating the system and under what conditions.

So we are really not to the point where we are looking at this kind of thing in terms of emergency operating procedures and that kind of situation. But we are at least going to be to the point where we are aware of what it would take to implement a system and it is something that we will be evaluating in terms of actual implementation after commercial operation of the unit. It is not something we are looking at trying to implement prior to commercial operation.

MR. EBERSOLE: One final thing. The usual death knell for this thing is the staff's defense of what they already have by the route of cost risk benefit analyses, a somewhat hypothetical analytical process which can be easily made to swing either way.

In your deliberations what kind of mix of PRA and judgmental effort do you contemplate doing to make your final decision on this?

MR. TERRY: Well, I think that the primary area really the degree of reliability that you have in being able to feed AC power to your RHR pumps. That is really the critical factor.

Frankly, from Niagara Mohawk's perspective and our overall system standpoint, we feel we do have a highly reliable AC power system. This is to a large degree a judgmental kind of thing, although there is a lot of information in terms of the true probabilities of the total loss of AC power. But that would weigh heavily. Right now from our perspective we really look at that as a very, very low probability event.

MR. EBERSOLE: Thank you.

MR. SIESS: Okay. Are you ready to go on with No. 17?

MR. ZALLNICK: Yes, sir. The topic of instrumentation for detecting inadequate core cooling will be presented by Mr. Doug Pike who was previously introduced. He is the Assistant Project Engineering Manager.

(Slide.)

MR. PIKE: I am Doug Pike, the Assistant Manager of Project Engineering. I will talk about instrumentation to detect inadequate core cooling.

(Slide.)

Basically as a result of Three Mile Island the Commission required that licensees shall have instrumentation that provides unambiguous, easy to interpret indication of inadequate core cooling.

(Slide.)

We are a member of the BWR owners group and we have been participating and following the activities that they have been working in this area with NRC by the way.

As a result of the NRC concerns, two studies through the BWR owners group were performed. One was an evaluation of present level instrumentation in BWR's. The other was an evaluation of inadequate core cooling and the need for additional ICC instrumentation.

The basic conclusions of those studies were, first of all, in a BWR water level is a conclusive indication of inadequate core cooling. They did find some problems with existing water level systems that were plant specific.

They made some recommendations for improving existing systems and procedures, by the way. A PRA as also performed on a generic plant model to put some of the problems and potential improvements into perspective.

Some of the conclusions of that PRA. First of all, they found a water level measurement contributes about eight percent of overall plant probability of core melt, enhancing of the operator's recognition of level measurement failures and improvement in level measurement reliability is equally as effective in reducing risk as is adding new ICC devices and, last of all, that the reduction in risk is so small that additional ICC devices are not required.

MR. EBERSOLE: We I guess asked a few questions yesterday, but maybe you can clarify the picture. In the presence of the worst hypothetical dynamic event in the containment, which I guess is the large LOCA, can you comment on the hypothetical -- well, I shouldn't say hypothetical, but the probable real damage that will be done to water level instrumentation and to the amount of residual equipment that is left to give you redundancy to do what you are supposed to do?

MR. PIKE: To clarify what we discussed yesterday, again we do look at high-energy line breaks in the primary containment.

MR. EBERSOLE: The large LOCA is one.

MR. PIKE: Yes. As far as damage from jet impingement from those breaks, we identify the potential targets and then we take a look and see if we can mitigate that specific break scenario with those targets damaged, and we also look at a single failure in the redundant division.

MR. EBERSOLE: That second single failure is not consequential but it is random, right?

MR. PIKE: Yes.k

MR. EBERSOLE: Otherwise, it would be guaranteed to occur when the accident did.

MR. PIKE: And then if we can mitigate that accident with those conditions, then that target may not be protected. Otherwise, it would be protected from that ---

MR. EBERSOLE: Well, did you realize it in this design in the presence of this violent LOCA?

MR. PIKE: Again, I am not familiar with the exact things that were looked at. However, you define what is causing the LOCA, which line break and so on and so forth.

MR. EBERSOLE: It implies against that 180 degree separation logic that on either side there is redundancy.

MR. PIKE: That is correct.

MR. EBERSOLE: Is there?

MR. PIKE: Yes.

MR. EBERSOLE: Good. Thank you.

Sorry, I meant on both sides and that means four because one side is torn away.

MR. PIKE: If one side is torn away, there is sufficient redundancy on the other side, yes.

MR. EBERSOLE: Good. Thank you.

(Slide.)

MR. PIKE: Some features of our Unit 2 level measurement systems. We have pretty much the BWR 5 plant specific design. We measure level via differential pressure of the water in the reactor vessel compared against a reference standard. We measure the differential using a Rosemount pressure transmitter which transmits that signal back to the control room to various indicators, recorders and trip units.

It has been shown that this Rosemount analogue transmitter and trip system is highly reliable, it is testable at power, it minimizes spurious operations and minimizes instrument drift.

Incidentally, at Unit 1 we originally had the old mechanical pressure switch system. That was changed out, as Mr. Stuart indicated yesterday, and they have conclusively shown that that resulted in a significant reduction in instrument drift and spurious operation, and this has been shown at other plants also.

Our system has redundancy and diversity built in, namely, things like the 180 degree separation of the sensing lines inside the containment. We have also separation as far as similar functions, in other words, the reactor scram for low water level is on completely different sensing lines than the ATWS water level scrams.

We just mentioned our jet impingement study inside the containment.

We also took a look at the worse case reference leg failure coupled with an additional single instrument failure in the redundant system and found that for the worse case the core still remains covered without operator action.

MR. EBERSOLE: May I ask you this. I am almost sure that the full committee will raise a question about one of the common topics now, which is overfill of the steam generator in the context of what does it do to the main steam system. I don't think the boilers have the problem because they run on level anyway. Well, so do the PWR's for that matter. But you have a multiplicity of level controls.

MR. PIKE: Yes, sir.

MR. EBERSOLE: Are these safety grade?

MR. PIKE: The high-level trips for your high-pressure core spray and RCIC are full safety grade.

MR. EBERSOLE: What about main feedwater?

MR. PIKE: Feedwater and turbine trip are not safety grade. However, we have specifically bought highly reliable non-safety grade equipment. We have a logic there that it is a two out of three logic, and we also have tech spec requirements on those instruments for surveillance testing.

MR. SIESS: How does the highly reliable non-safety grade equipment differ from safety grade equipment physically?

MR. PIKE: It probably doesn't. It is a matter of the kind of QA programs that are applied.

MR. SIESS: Thank you.

MR. EBERSOLE: Probably not seismic.

MR. SIESS: It is probably the same piece of equipment.

MR. PIKE: Pardon me?

MR. SIESS: I would suspect it is the same piece of equipment without the paper.

MR. PIKE: I can't say that it definitely is. I am not sure, but it would be the same or equivalent, yes.

MR. EBERSOLE: Well, there is some number in your PRA which is not zero that you might overfill and stack it up right to the turbine stop valves. Are you prepared to do that without knocking the steam lines down?

MR. PIKE: I can't answer that question. I am not sure whether the steam lines have been analyzed for that event or not.

MR. ZALLNICK: Mr. Rademacher can address that, sir.

MR. RADEMACHER: We have analyzed that case and the main steam lines can handle the water.

MR. EBERSOLE: With the normal hangers?

MR. RADEMACHER: Yes, sir.

MR. EBERSOLE: Thank you.

(Slide.)

MR. PIKE: This just gives you a quick indication of what kind of level indication we have in our control room. We have 10 separate indications of reactor water level. As you can see, the fuel zone and wide range are safety grade and all the others are backed up with uninterruptible power supplies.

(Slide.)

Instrument ranges. We have five separate ranges that provide level indication from below the core to above the reactor head flange. All of those ranges are referenced to a common zero, a common water level reference, and the safety related fuel zone and wide range indication are fed to our SPDS displays for trending and invalid data indication.

MR. EBERSOLE: Could you tell me what level trips the high-pressure core spray? What trips it to go into operation?

MR. PIKE: High-pressure core spray?

MR. EBERSOLE: Yes.

MR. PIKE: Low water level or high dry well pressure.

MR. EBERSOLE: Either?

MR. PIKE: Either, yes.

MR. EBERSOLE: Thank you.

(Slide.)

MR. PIKE: Some of the problems that have been identified in the owners group studies with systems. One concern is uniform heating in the dry well and its effect on the sensing lines in the dry well. That is not a problem at Unit 2 because our sensing and reference legs for the narrow, wide range and fuel zone instruments all have the same vertical drop inside the containment. So uniform heating of those legs does not result in any net change in indicated water level.

The other main concern is conditions where you could get flashing and boil off of the reference leg sensing line. This generally occurs when the vessel saturation temperature drops below the saturation temperature of the water in the sensing lines.

It has been shown for pipe breaks inside containment that protective action occurs before these level trips are adversely affected for these scenarios.

For long-term effects our emergency operating procedures provide for the following. They instruct the operator in avoiding the situation, it instructs the operator in how to recognize and respond to the situation if it occurs as far as actions to be taken on either increasing dry well temperatures when vessel saturation temperature equals dry well temperature and for total loss of level indication.

We also have a safety related dry well temperature monitoring system which will alert the operator to increasing dry well temperatures, and he can use this system coupled with the reactor pressure indication to determine when flashing and boiling off conditions are probable.

MR. EBERSOLE: Let me ask a question about the instrumentation that sends a signal to such things as the low-pressure valves between high and low pressure systems or valves which open to supply water to the core from high-pressure sprays or RCIC in this case. Are there any instrument line breaks which will synthesize a low pressure signal and be one of two redundant signals which then tells the low to high-pressure valving to do what they are supposed to do and find that they are up against an impossible torque load and they trip out in common?

Are you with me?

MR. PIKE: Yes, I understand your question, and I am trying to recall the logic involved with the high and low pressure interlocks.

MR. EBERSOLE: It is kind of a one-track start to a common mode failure.

MR. PIKE: We are looking at reactor pressure here and not level.

MR. EBERSOLE: Yes.

MR. PIKE: Basically our reactor pressure systems are the same as the level systems as far as the number of sensors, the logic involved, the way they are routed and so forth.

MR. EBERSOLE: Here the hypothesis is though that you synthesize a signal which locks the valves on a tripped out mode and then subsequently you get a low level because you really did have a leak in the primary vessel.

MR. PIKE: I guess I can't answer that.

MR. EBERSOLE: Later on maybe when we get together we ---

MR. PIKE: I will have to look into that.

MR. RADEMACHER: Are you questioning that the valves will open up at high pressure.

MR. EBERSOLE: That they tried to open, but they are unable to because they don't have the ability and thus they torque out and trip and then you will need them a half an hour later.

MR. ZALLNICK: But you are saying this is the result of an instrument line rupture ---

MR. EBERSOLE: Yes, right.

MR. ZALLNICK: --- and that you are getting a low water level because of an instrument line rupture?

MR. EBERSOLE: I get a one out of two invalid low-pressure opening signal.

MR. RADEMACHER: If I might address that at this time then now that I understand the question.

MR. EBERSOLE: Sure.

MR. RADEMACHER: We originally had I think a 750 psid transmitter that tripped these valves. Subsequently we have committed to the NRC to add an additional 100-pound permissive. Therefore, if a single instrument line were to break, we could still have the redundant permissive, and if that was not initiated, then the valve would not try to open up under that condition.

MR. EBERSOLE: Are you telling me they have independent impulse lines for each division?

MR. RADEMACHER: I believe that is the case. I am going to have to check on that. It was a special hundred pound trip that we had that goes directly to the reactor.

MR. EBERSOLE: I wish you would because sometimes you find that the IEEE divisional logic was never applied to the impulse lines.

MR. RADEMACHER: Okay. I think that went right back to the beginning and it has a separate line, but we will check on that.

MR. PIKE: That concludes my presentation on inadequate core cooling.

MR. SIESS: Is there anything that you have described for the ICC instrumentation on Nine Mile Point, Unit 2 that is significantly different than that for other BWR 5's?

MR. PIKE: No.

MR. SIESS: Thank you.

Let's see, did I say we would have a break?

MR. EBERSOLE: Sooner or later.

MR. SIESS: Well, we are only a few minutes late. it is now 10:20 and we will reconvene at 10:35 and take up the next item, and I assume the containment item will still be deferred.

(Recess taken.)

MR. SIESS: I would like to continue with Item 18, ATWS.

MR. ZALLNICK: Yes, sir. The presenter for ATWS is Mr. Norman Rademacher. Mr. Rademacher has 10 years of nuclear BWR experience with Niagara Mohawk. He is the nuclear design coordinator. He has worked on engineering also at Unit 1 and Unit 2.

(Slide.)

MR. RADEMACHER: Good morning.

My name is Norm Rademacher. I am the Nuclear Design Coordinator for Nine Mile Point, Unit 2.

(Slide.)

Back in June the NRC issued 10 CFR 50.62 which requires mitigation of ATWS for Nine Mile Point, Unit 2 and it required alternate rod insertion, recirculation pump trip and automatic standby liquid control.

It was published in the Federal Register. In the comments it also addressed reactor trip system reliability assurance and challenges to safety systems.

(Slide.)

Nine Mile 2 is installing an alternate rod insertion subsystem, a recirculation pump trip and automatic standby liquid control, and we are in conformance with 10 CFR 50.62.

We are currently preparing our submittal to go into the NRC to describe our designs.

(Slide.)

Anticipated transients without scram rules, which were published as part of the Federal Register, addressed many aspects of the design, and this slide basically summarizes what the rules required.

As I have noted here, we are in conformance, or in fact in some cases we are even a little bit better off. For example, in redundancy it is not required, but we are redundant and most of our equipment is safety related.

(Slide.)

The rule did not require seismic qualification, but ours is seismic.

As far as quality assurance, there was a recent publication by the staff that indicated that the ATWS equipment need not be quality assurance category one, and basically all of our equipment is category one or separated from category one.

We also have a safety related power supply which is even more than what is required by the rule.

Another point that the ACRS wanted to ---

MR. SIESS: Let me interrupt a moment in talking about the QA level. They didn't require this to be QA'ed at category one, right?

MR. RADEMACHER: That is correct. They published some rules and some guidance that said ---

MR. SIESS: Now I have been reading a lot of the responses from various licensees about the QA level, and some people have seemed to think that it is a multiple level QA system and they are not too happy with it, and they would either like to be category one or just good industrial quality.

Now you chose to make some of these things category one, right?

MR. RADEMACHER: This is one of the cases where Niagara Mohawk made a decision on its own prior to any rule coming out. Back maybe two or three years ago we decided on the Nine Mile 2 project to install ATWS 3A for Nine Mile 2. That decision was made by Chuck Mangan and the previous vice president. And at that time, since we were not aware of what the criteria was that would be acceptable, we basically bought QA CAT 1 equipment.

MR. SIESS: The reason I asked is that some of the objections were to multiple QA levels. They thought this was a very complex thing and very difficult, but you already have a multiple QA level. It was indicated on the previous slide I think three other categories.

MR. RADEMACHER: That is correct. Well, it is really CAT 1 and CAT 2, which has two subsets and then CAT 3, yes.

MR. SIESS: Now you found this workable and useful?

MR. RADEMACHER: We have been implementing that quality assurance criteria since I think when we started construction in 1975. The purpose of the ---

MR. SIESS: Now anything but category one in on your own, isn't it?

MR. RADEMACHER: The staff didn't require it, no.

MR. SIESS: So this is an internal matter, the 2A and 2B and 3?

MR. RADEMACHER: That is correct. Basically our upper management decided that they wanted when this plant was built to end up with a reliable non-safety related plant. So we imposed certain quality measures on our non-safety related equipment, and that was a management decision going back from the start of construction. Our quality assurance program for construction incorporates category 2 and category 3.

And that multiple level goes all the way down through your QC and your vendors?

MR. RADEMACHER: We have existing programs for that, yes.

MR. SIESS: And your vendors and your contractors all work with a multiple QA level?

MR. RADEMACHER: Yes.

MR. SIESS: Has that given you any trouble at all of people know which QA level they are working at?

MR. RADEMACHER: Well, I am not a quality insurance inspector, so I couldn't answer that question, although we could bring up our quality assurance people.

MR. SIESS: Well, if time permits later on I might. It is sort of a peripheral issue. We have been looking at QA in another aspect and this is I think the first time we have seen that approach and I was just wondering how it worked. But maybe we will ask you to come to our QA subcommittee sometime and tell us how it works.

MR. RADEMACHER: By the way, I just wanted to point out also that I am not sure where that stems from, that comment. But in any case, since Branch Technical Position 951 has been issued, applicants have been required to implement a fire protection quality assurance program, for example, on Nine Mile 1 and we are using the same program on Nine Mile 2 having a fire protection program as well.

So that, as far as I know, has worked out quite nice. We have certain standards and requirements for our fire protection equipment. So that has also been implemented. So to address your question, I don't see that that has caused any problem.

MR. SIESS: Well, actually, coming back to ATWS, the staff has published something describing the QA levels for ATWS. It is quite a document. I don't remember what it is published as.

MR. RADEMACHER: I believe it was published in the Federal Register.

MR. SIESS: I think it was in the Federal

Register, and a number of people have had difficulty understanding it. Now I assume that doesn't give you any problem.

MR. RADEMACHER: Well, we are basically quality assurance CAT 1.

MR. SIESS: Well, that is one advantage I guess.

MR. EBERSOLE: May I ask about the last two topics up there. Testable at power. Now the ATWS mitigation involves pump trip and automatic boron injection, right?

MR. RADEMACHER: Yes.

MR. EBERSOLE: How do you test these at power?

MR. RADEMACHER: The logic circuitry is solid state. So we can test the logic circuitry but not all elements are testable. Obviously if you trip the recirc pumps and you go all the way to the end result, you are going to scram the plant or you are going to initiate automatic standby liquid control into the vessel. But the logic is solid state which is similar to the Rosemount pressure transmitters and that kind of thing.

MR. EBERSOLE: But they never include the final element?

MR. RADEMACHER: That is correct. Those kinds of tests, the remaining portions of the tests can be checked like at refueling outages and that kind of thing.

MR. EBERSOLE: Well, certainly when you want to

shut down you can test pump trip, but what about boron
injection?

MR. RADEMACHER: Well, boron injection I believe we check it -- I think the tech specs require it once a month, but it goes to the test tank and not the ---

MR. EBERSOLE: I mean when do you test or replace the injector, or what are they, they are squibs, aren't they?

MR. RADEMACHER: Yes.

MR. EBERSOLE: And what do you do about them?

MR. RADEMACHER: According to the tech specs at the fueling outage we fire them not in place, but we ---

MR. EBERSOLE: You take them out and fire them.

MR. RADEMACHER: Yes.

MR. EBERSOLE: Now early on when this automatic boron injection was proposed, there was a tremendous hue and cry about how many million dollars it was going to cost you all through inadvertent injection, and I am quite sure it was heavily awarded in the direction of a higher frequency that wouldn't really occur.

What did you come up with in a PRA context about the frequency of inadvertent injection?

MR. RADEMACHER: I have a backup slide that I can show.

MR. EBERSOLE: Will that come out separately later?

MR. RADEMACHER: It is a backup slide, and I can throw it up if you would like.

MR. EBERSOLE: You must have concluded that it was not too frequent, and what was the real estimated cost of cleanup? Did you put in any bigger demineralizing systems for the hypothetical inadvertent actuation?

MR. ZALLNICK: I think Mr. Terry can address part of that question while Mr. Rademacher is getting ready.

MR. PRACHT: With reference to the cleanup system, Nine Mile 2 is double what GE normally would supply for this vessel. We did that early on due to operational considerations and of course this is a fallout, if you will, in this case.

(Slide.)

MR. RADEMACHER: This is the slide I was talking about. A reliability analysis for ATWS 3A was performed. This was performed by GE and I believe it is a needy document that was prepared for us. It is the inadvertent of RRCS. That is the initiating logic for ATWS.

Signals resulting from random mode sensor and logic failures is less than 1 times 10 to the minus 4th per year, and inadvertent feedwater runback signals is 1 times 10 to the minus 4th.

MR. EBERSOLE: How does that compare with the original estimate when they were fighting against doing anything about this, do you know? I suspect it is two orders of magnitude different.

MR. RADEMACHER: I am not sure what the original estimate was. What GE did for the design was basically to make it redundant 2 out of 2 logic.

MR. EBERSOLE: You mean coincident?

MR. RADEMACHER: Yes. So, therefore, they improved both the reliability of non-injections, so to speak.

MR. EBERSOLE: It is interesting to see how you can do what you want to do if you want to do it.

MR. SIESS: You are leaving out the uncertainties.

Have you made any estimate of how long it would take you to clean up if you had an inadvertent injection?

MR. RADEMACHER: Yes, we have.

MR. SIESS: How long?

MR. RADEMACHER: We estimate with both trains of the cleanup, which Don Pracht I think mentioned that we have extra capacity, that it would take on the order of one to two days to clean up. That would get us down to about 50 parts per million I believe is the number. We would be able to return to power, but we probably would experience fuel problems until we got it all out.

MR. EBERSOLE: I wonder if you could run back a couple of slides to where you were talking about the boron injection in the standby liquid coolant system, the 86 GPM.

(Slide.)

There it is. Let me ask you at this point about one of the problems of that system. As you know, it is a limited supply and everybody is exuberant when this is going to keep the core covered, and the SRV's are open or may have stuck because of a variety of causes. So the inspector of washout is in your face.

It would take a skillful and knowledgeable operator action to preclude washout, and if you do wash out, there is only one other back door, which we heard about last week which is superdepressurization which runs the core down to seven percent just operating in froth cooling and the reactivity is so low at that time that you hold it at seven percent which is probably compatible for a long time with your absorptive capacity and your heat removal from the reactor to the pool, but would eventually heat the suppression pool.

Tell me what you do to prevent washout. It seems to be almost entirely administrative.

MR. RADEMACHER: Well, first of all, I am not sure that what you are describing is the same as our design. I think you are looking at the 2A design where they lower the water level. Is that what you are describing?

MR. EBERSOLE: Well, it doesn't matter. I am looking at the aspect of on the one hand wanting to get water to the core, but with it the possibility that you drive water out the SRVs and flush out the small bit of boron that you have forever into a big pool so that it is so diluted that it is no longer effective.

MR. RADEMACHER: Oh, okay, I see what you are saying. You are asking me whether we have boron replenishment?

MR. EBERSOLE: Well, that is one course, or just tell me why you don't wash out what you have got.

MR. RADEMACHER: There is a possibility that we could wash out. So we normally keep approximately 1500 pounds or so of boron available to replenish it into the standby liquid control tank.

MR. EBERSOLE: So you have a second shot?

MR. RADEMACHER: There might be the need for additional injection, yes.

MR. EBERSOLE: Everybody is keenly aware of washout then. Somebody says the frequency is so low that it will be three generations before we see it have to be done, if ever. It is a long memory interval.

MR. RADEMACHER: Well, basically what we do is we have that available so that we could use it if we need it.

MR. SIESS: How much boron did you say?

MR. RADEMACHER: I think it is 1500 pounds per charge. We normally keep two additional charges.

MR. SIESS: That is two additional charges?

MR. RADEMACHER: Yes.

MR. SIESS: Okay.

MR. EBERSOLE: That is on the thesis that you don't wash it out the second time.

MR. SIESS: That gives them three times.

MR. EBERSOLE: That gives them three times, right. It is interesting to note, you know, that the boiler, one of its fascinating features is if you completely depressurize it, it will in fact reverse itself to I am told seven percent, which is not incompatible with the heat removal rate from RHR and other cooling systems, although it will ultimately heat the reactor up.

Do you have any instructions to your operators or do they know that?

MR. RADEMACHER: Yes. Right now, as a matter of fact, we haven't updated our emergency operating procedures and they are still back on the old 2A design which is ---

MR. ZALLNICK: Norm, I think Mr. Colomb can answer that question.

MR. COLOMB: My name is Mike Colomb, and I am a

Station Shift Supervisor. Our emergency operating procedures at this time direct the operator not to flood the vessel to a point where it will overflow through a relief valve if in fact we have injected boron, and we do use the seven or eight percent power as a level requirement to maintain power load until the boron is injected and then raise the level.

MR. EBERSOLE: Do you go to full depressurization to lower pressure to get a high void content?

MR. COLOMB: What we do is lower level to approximately top of active fuel ---

MR. EBERSOLE: I didn't say level. I said pressure.

MR. COLOMB: Yes, we would.

MR. EBERSOLE: What pressure to you descend to?

MR. COLOMB: The pressure is dependent on other parameters. I would have to look at that. I could get you an answer to that.

MR. EBERSOLE: Well, certainly, you would be willing to sacrifice the RCIC, wouldn't you?

MR. COLOMB: Yes.

MR. EBERSOLE: It would be interesting to hear how you really stand off from ATWS using low depressurization if you could tell us sometime.

MR. RADEMACHER: We will get back to you on that.

MR. SIESS: What do your emergency operating procedures say about determining the cause of the failure to scram and attempting to remedy it?

MR. COLOMB: That is a parallel path. We take steps to find the cause and correct it while we are addressing the ATWS situation, addressing the vessel itself.

MR. SIESS: Of the causes you have thought about, which one takes the longest to correct?

MR. COLOMB: Probably the hydraulic problem and the scram discharge volume. The scram has to be cleared and has to be vented and drained.

MR. SIESS: How long does that take?

MR. COLOMB: That would depend on how severe the problem was.

MR. SIESS: Okay.

MR. EBERSOLE: Finally, I wanted to call your attention to an ancient proposition or kind of a root logic. Based on this superstitious fear of losing a little water with radiation in it, the notion of course grew into the GE design that they close the dump volume before they got the rods in. You can judge for yourself whether it is important to have a little controlled discharge of radioactive water to a controlled tank or getting the rods home. I think I know which I would rather have.

In order to overcome what was eventually realized as a failure in logic due to the fact that a single failure of the event or dump valve, if it locked open, you would have fluid going to the containment. The Hatch incident, which I hope you have studied intensively, an almost automatic reaction took place to put redundant vent drain valves in. That doubles the responsibility of those three levels switches on which is the safety of your whole plant.

I was thinking of looking at your model when I say this and all those complex details. It all focuses down to three level switches in the scram dump volume, which if they don't work, you are in trouble.

I wonder if you look in sort of perspective at the effect of adding the redundance switches which doubles the probability of failure in the direction of not opening. Are you with me? And therefore it doubles the duty and the reliability context of the level switches and is in the wrong direction. It guarantees to a higher degree non-leakage, but it guarantees to a higher degree the possibility of ATWS unless the level switches always work. Are you with me?

MR. RADEMACHER: I think I am a little confused.

MR. EBERSOLE: I am saying you have doubled the probability that you will not either vent or drain because you put these two valves in series. The motivation behind that was to enhance non-leakage. It was detrimental to scram. It had to be because either of these valves on either vent or drain now will preclude dumping the dump volume. That forces your level instrumentation to have to have to do its thing. Are you with me?

MR. RADEMACHER: I think I understand now.

MR. EBERSOLE: Have you put this on a PRA and come out with a net answer in a radiation hazard context as to whether it was a good thing to do that?

MR. RADEMACHER: No, we haven't.

MR. EBERSOLE: I think it might be an interesting exercise.

(Slide.)

MR. RADEMACHER: One of the other topics that you asked us to discuss here today is relative to scram reductions.

Basically because of our operating experience on Nine Mile 1 we have had some pretty good experiences. So we wanted to relate them today.

We have improved the equipment for system design to reduce scrams and enhanced, as we discussed earlier, some enhancements in our quality programs for non-safety related equipment which we already discussed.

(Slide.)

Relative to the materials upgrade, we have used low carbon stainless steel to mitigate intergranular stress corrosion cracking in this area. We have done an analysis for what is called the fast scram hydrodynamic loads and the HCU pilot scram solenoid valves have refurbished. There was a recent bulletin I believe out on the type of rubber goods in those that needed to be upgraded.

(Slide.)

Improvements on the component and system design level to reduce scram.

For feedwater we have three 50 percent capacity strings. Feedwater I guess is probably one of the major contributors to scrams. We have one pump in standby. I don't know if Doug mentioned it or not, Doug Pike, on the AC design, we have the capability to provide power for the feedwater pumps from either a non-safety A bus or a non-safety B bus kind of idea.

MR. EBERSOLE: What is the configuration of your main feedwater pumps? Are they all electric driven?

MR. RADEMACHER: Yes, they are.

MR. EBERSOLE: You didn't go for the standard approach to turbines. Could you discuss why you didn't do that?

MR. RADEMACHER: Don Pracht could probably discuss that. He is our lead mechanical.

MR. EBERSOLE: I see more often than not the turbine driven jobs with all the problems that go with them, except economy.

MR. PRACHT: Niagara made a conscious decision in this area back in late '71 or early '72 regarding which pumping scheme we should utilize. Substantial consideration was given to maintenance and physical configuration of getting the turbines within the plant in order to exhaust correctly to the condenser. The bottom line was that we chose to go with the pre-electric driven feed pumps figuring that those were the most reliable design that we could come up with.

MR. EBERSOLE: Well, does the third one manage to get on line before you trip off without runback?

MR. PRACHT: No. The third pump will not come alive until manually brought on. In other words, as Norm has indicated there, the recirc runback will occur and a single pump can handle 68 percent.

MR. EBERSOLE: So you go back to recirc runback and you don't have to scram it?

MR. PRACHT: That is correct. We have run a computer simulation of the plant in order to guarantee to the best of our ability that this will occur.

MR. EBERSOLE: But Nine Mile 1 has turbine pumps, doesn't it?

MR. PRACHT: No.

MR. EBERSOLE: I had forgotten. I didn't know.

MR. PRACHT: Nine Mile 1 has a single shaft driven pump plus two small electric driven pumps.

MR. EBERSOLE: You are getting a mighty good trip rate on that one. Thank you.

MR. RADEMACHER: The other enhancements, the use of a Rosemount analogue trip system and recirc. runback on partial loss of feedwater.

(Slide.)

A prime contributor to unnecessary scrams is inadvertent actuation by personnel during surveillance testing. We have carefully evaluated our surveillance test procedures on Unit 1. We have experienced operating people there and we have not experienced a scram caused by surveillance tests since 1974 on Nine Mile 1.

On Nine Mile 1 the overall experience is 5.4 scrams per year, and if you look at that in groups, the industry average is 6.4 and the Nine Mile 1 average between '72 and '84 was 3.25 and most recently it is down to 1.4 scrams per year.

MR. SIESS: Now I assume you don't really expect Unit 2 to be at the 1.4 level the first year or so. But with your physical enhancements that you mentioned, you feel that you have the potential in Unit 2 to get down below the Unit 1 level?

MR. RADEMACHER: I think there are two factors that maybe bear on this subject. One is that first topic there on carefully preparing your surveillance procedures and the second one is we have an experienced operating staff. I think those two key facts have resulted in the third one which is that we haven't had scrams. I am not saying that an immature plant can do the same as a mature plant, but I think certainly those values are obtainable on Unit 2.

MR. SIESS: And you think that the mechanical enhancements you had on the previous slide can't do much to help ---

MR. RADEMACHER: --- to help us. Yes, sir.

MR. SIESS: Most of it is in the surveillance and human factor things?

MR. RADEMACHER: I believe that is the case.

MR. EBERSOLE: Now NRC normally hands out fines. Don't you think they ought to hand out a reward when you get down to one per year?

MR. SIESS: It has its own rewards.

(Laughter.)

MR. EBERSOLE: Which is a lot more than NRC would give.

MR. SIESS: Yes. They are built in.

(Laughter.)

MR. RADEMACHER: Okay. That basically concludes my presentation on ATWS and scram reduction.

MR. SIESS: Now your ATWS fix is basically standard, except that you do have the CAT 1 QA on it, which may or may not be an improvement.

MR. RADEMACHER: I am not familiar with other designs. So I can't say what they have done.

MR. SIESS: Well, some of them haven't done anything yet. The category one QA was certainly an administrative improvement from your point of view, but whether it makes the system any more reliable, you are not really prepared to say I assume.

Thank you.

MR. RADEMACHER: Thank you.

MR. SIESS: Has the staff got the containment people here?

MR. WEINKAM: Yes, sir.

MR. SIESS: Okay. Let's see. Let's then take up the containment item which is Item 16.

MR. ZALLNICK: Could you give us about a minute to reset the slide projector back to the containment.

MR. SIESS: Okay, sure. Go ahead.

MR. ZALLNICK: While that is being done I guess I can introduce the presenter.

The presenter for Mark II is Mr. Ed Klein who has been presented before. Mr. Klein is the Assistant Engineering Manager for Unit 2.

(Slide.)

MR. KLEIN: Good morning, gentlemen. My portion of the presentation is Mark II hydrodynamic unique features.

(Slide.)

The first unique feature I would like to address is the self-supporting dry well floor. The dry well floor is integral with the pedestal and it has a fixed moment there at the primary containment wall, that is that is that the liner goes through. However the tension and compression are transferred by Nelson studs.

This design allows us to eliminate a supporting column right here next to the primary containment. Our original design had that supporting column and this left us a possible bypass leakage there. We had a special inflatable seal at that junction.

On top of this dry well floor now we have a 3/16ths steel plate that is welded to every penetration in the dry well floor and is welded to the primary containment liner which almost doesn't allow any leakage with that welded liner plate in there.

Our second unique feature is that we do not have any downcomer bracing. We started out our design assuming we may need it and we had had the embedments put in, but applying all the specific load requirements, we determined that the downcomer could take that design and that certainly eliminating that bracing that is in there provides more space for the water.

Once you start to design that bracing, you design it for the horizontal load and then you have got to design it for the vertical pool swell loads and they become quite cumbersome and take up quite a bit of space. So we think that that is unique and it serves the purpose quite well.

MR. EBERSOLE: Could I comment on that particular feature. It has always been interesting to notice that the Germans must not make as good steel as the Americans because they double jacket those downcomers. They are double pipes. We must believe in our single wall and they don't because they are anxious to avoid suppression bypass and so they go to that trouble.

So I look at the absence of these swing braces at the bottom and I get around to the general question of bypass by virtue of breaking off one of those cantilevered pipes and ask you what happens?

MR. KLEIN: You are going over the hypothesis that I broke the pipe off?

MR. EBERSOLE: Yes. Brookhaven said you don't have long to go because you overpressure the containment without suppression. You get a hot layer on top of the suppression pool and there is no longer any condensation.

MR. KLEIN: I would like to point out that our downcomers, the seams on those downcomers were all X-rayed completely to determine their integrity. If you look at the pressure, the pressure stress on that thing is almost nill.

MR. EBERSOLE: It isn't that. It is the vibratory potential at low frequencies. Has that been gone through? You know, the low frequency chugging.

MR. KLEIN: The chugging, correct. We used all the criteria in the NUREG to design those downcomers and laboriously looked at that condition to make certain that we could take that loading.

MR. EBERSOLE: Did you look at the full spectrum of potential lateral loads due to any quality of distribution at the bottom of tne pipe?

MR. KLEIN: If you are talking about the load combination for the faulted condition on those downcomers, each downcomer has the seismic load, which is inertia and sloshing. It has a load from a T-quencher, which assumes a failure of an SRV during an SSE event, and it also has the LOCA loading which is condensation oscillation form an adjacent downcomer.

So that is an extremely conservative load combination.

MR. EBERSOLE: Well, what I am just trying to do is make it absolutely certain you don't have suppression bypass.

MR. KLEIN: I realize that is what you are leading to.

MR. EBERSOLE: And you are satisfied that you have got margins of stress like factors of safety of what?

MR. KLEIN: For the faulted condition -- I think I would like to ask Mark Durka of Stone and Webster to address the amount of safety factor on that faulted condition.

MR. DURKA: Just to go back a little, we have looked at the full spectrum that Ed has mentioned. We have been through seismic inertia sloshing. We have taken an SRV single failure concurrent with a full LOCA. We look at the chugging loads, which are both inertia and submerged structural loads from adjacent and direct chug and we look at the condensation oscillation loads. Now those are applied concurrently, except for seal and chugging which cannot occur at the same time.

We have evaluated in accordance with ASME NC. We have also done a fatigue evaluation on these downcomers and we have a cumulative usage factor that is less than .1.

In the faulted load case, the ASME allowable is 39 KSI. Now our resultant peak stress at the junction to the floor is 30.

Bear in mind that these downcomers are stainless. So that they do have what I would consider some reserve margin with regard to strain.

To put this in some sort of physical perspective, we are talking about an eight or nine-inch deflection at the top of the downcomer.

MR. EBERSOLE: They are stainless?

MR. DURKA: Yes, they are.

MR. EBERSOLE: Is that unusual? They were carbon steel, weren't they?

MR. DURKA: Well, this goes back to the discussion we have had previous, that the entire suppression pool is all stainless for water quality. At one point they were carbon. Now if you compare a carbon steel SA-106 Grade B, you are looking at a minimum yield of 35 KSI with an ultimate of 60.

MR. EBERSOLE: Yes, but I don't have the intergranular stress corrosion potential in carbon that I do here.

MR. DURKA: Well, in this particular case you are not sitting at an elevated temperature or have a stratified problem. Here you are very low in terms of pressure. But, as I was going to say, you have an ultimate capacity of this material of 75 KSI.

So from looking at say a braced carbon steel downcomer versus this, you are picking up an ultimate capacity from 60 to 75. In terms of the capability on a fatigue evaluation, I think stainless would give you far superior results. In addition, the loads are applied along a singular axis. They are concurrent in time in SSRSS and now we take and apply them all in the same direction and we also take the time factor of chugging. So that when you have a chug every two seconds, you know, we looked at the decay of that and we also used the recommended damping values of one percent because this is a single supported cantilever.

MR. EBERSOLE: Well, you have to ride on analysis, don't you? There is no way to test these pipes at all, or do you blank them off and pressurize?

MR. DURKA: Well, they were hydroed to begin with.

MR. EBERSOLE: What did you do, blank them off and hydro?

MR. DURKA: They were hydroed at the shop before they were installed.

MR. EBERSOLE: Oh, just as individual elements they were hydroed and then installed.

MR. DURKA: Yes, that is correct.

MR. SIESS: It seems like you have devoted quite an effort to that analysis. Is that because there is some perceived disadvantages of applying bracing at the bottom of these things?

MR. KLEIN: Yes. As I explained, sir, when you design those things, to take the horizontal load you have one diameter and then you have to look at the upward pool swell load and that strikes that and they you have to make it bigger to obtain that load and it is kind of chasing your tail a little bit.

If you ever went down in one of the other Mark II's that I have been in, there isn't an awful lot of space between those braces down there.

MR. SIESS: You end up with more steel than water.

MR. KLEIN: That is correct.

MR. EBERSOLE: However, you nailed the downcomers from the SRV's to the floor, didn't you, apparently?

MR. KLEIN: Yes. They are on a sliding joint down there, right.

MR. EBERSOLE: Well, let's drop the subject that you get bypass through a metallurgical failure and go to the main vent vacuum breakers.

As I recall, those things in some of these containments ---

MR. SIESS: Have you got a slide of the vacuum breakers?

(Slide.)

MR. EBERSOLE: If they were subject to oscillatory loads at low frequency which could have knocked the valves open and then they stuck and you could you get a bypass that way?

MR. KLEIN: That is if they were mounted on the downcomer itself, correct. That is my next presentation.

MR. EBERSOLE: Okay.

MR. KLEIN: The vacuum breaker location is our third unique feature. Our original design location was below the dry well floor and they were mounted on the four full-length downcomers. This would have caused a serious fatigue condition from the chugging phenomena.

Our construction schedule allowed us to relocate the four dual vacuum breakers in the dry well, above the dry well floor.

(Slide.)

The vacuum breaker disks were designed to meet ASME design criteria. The design basis condition is for the LOCA pool swell venting the air pressure. However, they are really designed to provide a vacuum when you have a possible vacuum buildup in the dry well. But they have to sustain the pool swell which opens them quite violently, and that is why those enhancements are required.

Basically those enhancements are that we added that bumper stop there, the dark model, and we thickened that hinge plate and upgraded its material and impact distribution devices were added. They are right in there. They are shown right there and they are shown by the dotted line over there. And that eccentric shaft in that area right there had a material upgrade. That is approximately 20 radians per second for that condition, and that is a one time only LOCA event.

MR. EBERSOLE: These could not be practically tested or were they? This is purely analytical that they will work, right?

MR. KLEIN: I will ask Mr. Durka to address that one again.

MR. DURKA: They were tested, these particular valves, at the LaSalle Station. These are GPU vacuum breakers and they have the same basic design. They tested an unmodified one and they did have some deformation and they were able to conclude that that was acceptable for their bypass leakage calculations.

They subsequently went and modified a vacuum breaker exactly like this and it was tested with an opening velocity of 18 radians per second and it sustained no damage.

We have taken an analysis which takes that opening velocity of 18 radians and it takes it up to 20, which is our opening. They bounded our test in closing. So they were tested.

MR. EBERSOLE: Let me conclude by asking all of you if you have read and agree or disagree with the Brookhaven report about what happens if you have a big bypass with a LOCA?

MR. KLEIN: I haven't read it myself.

MR. EBERSOLE: Any comments from the applicant about the Brookhaven study? Why don't you read it before we see you in Washington and see what you think about it. It is somewhat depressing.

That is all I have.

MR. ZALLNICK: I would like to have Mr. Rademacher provide the status of the open items in containment.

MR. SIESS: Okay, fine.

MR. RADEMACHER: There were several open items in this area.

One was steam bypass. That one basically entailed two items. The first item entailed initiation of containment sprays and Nine Mile 2 has a manual system for initiation. The staff asked for some additional information as to the adequacy of this and we have provided ---

MR. SIESS: That staff has indicated that is confirmatory now.

MR. RADEMACHER: Okay. Then the other aspect was relative to testing of the vacuum breakers at each refueling outage and we are preparing a response to go in to the staff on that issue.

Basically our feeling is that ---

MR. SIESS: I am sorry. I am having trouble relating these to the staff's items. Do you know their appropriate numbers in the SER? They weren't that well described in the SER. The first was No. 5. That is okay.

MR. RADEMACHER: I believe there were two parts to that No. 5, and this is still the same topic of steam bypass. There is Part A and Part B.

The second one was that we needed to provide some information, as I said, on the post-operational low-pressure leakage test, and we plan to address that in March and submit our report on that.

The next issue related to secondary containment bypass leakage. We have had a discussion with the staff relative to those leakages that could occur that would bypass secondary containment from the primary containment.

Our analysis included main steam and the post-accident sampling system and the main steam drains. They asked us to address some additional potential leakages. We are preparing our response and I believe that will go in the next amendment of the FSAR.

MR. SIESS: Now this is item 6 you are talking about?

MR. RADEMACHER: Yes.

MR. SIESS: The low leak rate for the MSIV's, and you are submitting something on that, right?

MR. RADEMACHER: Relative to the low leakage rate we have provided the staff with some information on that relative to the testing that was done at Leipstot. The MSIV's are the same at Leipstot as ours and we have provided some information on the low leakage rate for that to the staff.1

MR. SIESS: And the other item was the water seals?

MR. RADEMACHER: Water seals?

MR. SIESS: Yes.

MR. RADEMACHER: Yes.

MR. SIESS: I am reading from the staff's slides. Would it help if we had the staff explain? I am having difficulty correlating your responses to the staff's.

MR. RADEMACHER: That was my first discussion under secondary containment leakage. We owe them some information on the water seals and leakage paths from the primary to the atmosphere.

I indicated that we are preparing a response on that to go into the Commission in the next FSAR amendment.

MR. SIESS: Okay.

MR. EBERSOLE: The secondary containment here, I can't quite identify the perimeter of it. Does it include the refueling floor?

MR. RADEMACHER: Yes, it does.

MR. EBERSOLE: And that is sheet metal walls?

MR. RADEMACHER: Above the crane level, that is correct.

MR. SIESS: Now your MSIV's are clearly different. Is the other item having to do with water seals something unique to Nine Mile Point, Unit 2?

MR. RADEMACHER: I believe that is unique.

MR. SIESS: It is a balance of plant item?

MR. RADEMACHER: Yes, it is.

MR. EBERSOLE: Is your secondary containment destructible by tornadic winds?

MR. RADEMACHER: No. We are designed for tornadic winds.

MR. SIESS: And it has sheet metal walls?

MR. RADEMACHER: That is correct.

MR. EBERSOLE: It has got roof drains from inlets at the top?

MR. RADEMACHER: That is correct.

MR. EBERSOLE: When they fall down in an earthquake do they bother the walls?

MR. RADEMACHER: They don't fall down during earthquakes.

(Laughter.)

MR. EBERSOLE: Okay.

MR. KLEIN: The sheet metal above the crane rail is designed to either stay on . come off, either way.

MR. SIESS: In an earthquake or a wind?

MR. KLEIN: During a wind.

MR. SIESS: Okay.

MR. EBERSOLE: Well, doesn't that breach secondary containment?

MR. RADEMACHER: Yes. During tornado conditions we don't postulate a LOCA plus a tornado at the same time.

MR. EBERSOLE: That is what I thought you were going to say, but you didn't.

MR. RADEMACHER: I guess I need to clarify my statement because the rest of the plant is designed for tornado conditions.

MR. EBERSOLE: Right, but not the tin walls.

MR. SIESS: It is like all other BWR's.

MR. RADEMACHER: Mark II, that is correct.

MR. SIESS: Or Mark I's.

MR. RADEMACHER: Or Mark I's, yes.

MR. EBERSOLE: Do you maintain that subatmospheric after an accident?

MR. SIESS: Jesse, let's get through the outstanding issues before we come back to general questions.

MR. EBERSOLE: All right.

MR. SIESS: Okay. No. 7 on the staff's list ---

MR. RADEMACHER: --- is containment isolation.

The NRC asked for some additional information relative to an exemption request for the recirculation pump seal purge line, and we are currently preparing that to be sent in to the NRC.

MR. SIESS: Exemption of what sort, testing?

MR. RADEMACHER: No. This was for a containment designed to meet general design criterion on containment isolation.

On this line we have three check valves and the staff does not give us credit for the outside isolation valve as a check valve according to the current regulations. So we have to request an exemption.

MR. SIESS: Which of the three GDC's is this? The staff can answer that if they want.

MR. LANE: GDC-55.

MR. SIESS: 55, and 55 applies to systems that are connected to systems, closed systems outside of containment?

MR. WEINKAM: This is John Lane from the staff.

MR. LANE: GDC-55 is systems connected to the reactor coolant pressure boundary.

MR. SIESS: Okay. And that requires one valve inside and one valve outside?

MR. LANE: Yes, right.

MR. SIESS: And one of those can be a check valve?

MR. LANE: The inboard valve can be a check valve and the outboard valve cannot be a simple check valve.

MR. SIESS: And they have what, three check valves?

MR. LANE: Right. They have one check valve inside and two check valves outside.

MR. SIESS: Is this something you are likely to give an exemption to?

MR. LANE: I would say that it is probable in this case.

MR. SIESS: I have participated in the review of a number of SEP plants, most of which were not in compliance with GDC-55, 56 or 57 most of the time, and almost all of them got exceptions on the basis that the PRA showed it didn't make any difference, that the reliability of the valve was the governing factor, and I was just wondering, if that fed back into the staff in effecting some of their decisions?

MR. LANE: Well, it has to an extent. We try to factor in the fact that the plant is already built and we do factor in that type of a consideration. We don't go as far as accepting PRA as an alternate acceptable basis. We still require an exemption and a justification for the exemption on a non PRA type of grounds.

MR. SIESS: Thank you. I will come back to you on some of the others.

Okay, Issue No. 8 was on containment leak testing.

MR. RADEMACHER: Yes. This was five items basically. The staff will require ballast and system lines listed above to be Type C tested with air and leakage results added to the Type A test.

We submitted information to the staff that indicates that we would comply. Relative to the control rod drive system, they asked us to vent it but not drain for the Type A test, and we still owe a response to the staff on that issue.

MR. SIESS: But they still will be Type C tested and added to the Type A?

MR. RADEMACHER: These valves I believe need not be Type C tested if we drain and vent them.

MR. LANE: That is correct.

MR. SIESS: That is correct?

MR. LANE: Yes.

MR. SIESS: Okay.

MR. RADEMACHER: The next issue related to hydraulic control lines for their recirculation system flow control valve. The staff asked that these lines be drained and vented and Niagara Mohawk plans to provide an exemption request for that because of the hydraulic fluid in those systems.

The next item was the transfer ---

MR. SIESS: Just a minute. I want to understand this leak testing. On the recirc flow control valve, they are to be drained and vented during the Type A?

MR. LANE: No. The applicant has proposed not to do any venting or draining or Type C testing.

MR. SIESS: So what is the basis then that these will never leak? Do they leak into a closed system?

MR. LANE: Yes, it is a closed system inside containment, but that does not allow a deviation from the Type C testing or Type A testing.

MR. SIESS: I mean the object of the leak test is to find out whether the containment leaks.

MR. LANE: Yes.

MR. SIESS: And presumably whether it leaks to the atmosphere.

MR. LANE: Right.

MR. SIESS: Either directly or through some other failure path. If these are not to be included in either Type A or Type C, then they must not be considered a potential source of leakage.

MR. LANE: Right, and that is where we have a potential disagreement with the applicant on that, that we have not been shown yet that we can exclude them as a potential source of leakage.

We give credit for a closed system inside containment in regards to isolation. We consider the closed system inside containment as a isolation barrier. But as far as the leak testing goes, it looks like from our review of the system so far that we can't conclude, unless pending some additional information from the applicant that changes our mind, we can't conclude that there is not going to be any leakage through this system.

MR. SIESS: But if it is a closed system inside containment, what will be the source of leakage outside containment?

MR. LANE: It is a non-essential system that may be disabled post-accident.

MR. SIESS: But it must somehow go outside.

MR. LANE: It does go outside. It is closed inside and goes outside.

MR. SIESS: Are the ball valves exempt from Type C, but drained for Type A? I didn't get the details of that.

MR. RADEMACHER: Which ball valves, the Tip system?

MR. SIESS: Yes.

MR. RADEMACHER: Relative to the Tip system, we have provided to the staff an exemption request for these valves. This is a traversing incore probe system, and it is basically a closed system that is used to calibrate the LPRMs. It doesn't contain a fluid and it is normally closed.d

MR. SIESS: Now would leakage through those valves be detected in a Type C test?

MR. RADEMACHER: It is a closed system from containment.

MR. SIESS: I am trying to make a test to determine whether containment will leak when it is pressurized to some level, and in a Type A test I pump it up and measure the leakage.

MR. RADEMACHER: That is correct. If there was leakage at that time through this valve and through the Tip system, it would be picked up by a Type A test.

MR. SIESS: Now in the Type A test some valves in some way excluded and their leakage is determined separately and added in, but these would be included in the test?

MR. RADEMACHER: For Type A, yes.

MR. SIESS: Then what is the issue?

MR. RADEMACHER: They would like us to Type C test them. The NRC would like us to Type C test these valves.

MR. SIESS: Now the Type C is a more frequent test?

MR. LANE: No, it is not more frequent, but it is separate from the Type A test and it gives a little bit more reliability. As opposed to the overall containment structure, Type B and C is airlock and valve tests and it has a lower acceptance criteria of .6S L sub A versus the overall containment test, which is 1.1 L sub A.

So the issues here is whether or not they should be individually Type C tested in addition to the Type A test, which is the normal procedure.

I guess my confusion arises from knowing what is being proposed for revisions to Appendix A and trying to compare that with what we have been doing and trying to compare all of it with the containment integrity following a severe accident, and the three things are completely incompatible.

MR. RADEMACHER: Would you like me to continue? There is one last item relative to the leak test, and that related to the NRC's request for information that we provide both the total time and mass point method for the Type A test, and I believe we have submitted information to address that.

MR. SIESS: Are you familiar with the current activity and the proposed new Reg. Guide that has been kicked around for about two or three years and the proposed new standards having to do with determining the leak rate Type A tests?

MR. RADEMACHER: I am not personally, but I am aware of it and our Appendix J type people are aware of it. I believe it is an ANSI Standard 56.8 or something like that.

MR. SIESS: Okay. Would the staff sort of summarize where they stand on these open item, 5, 6, 7 and 8 and where they think they might be in a couple of months? Five I think you had already indicated you had received information from the applicant and it was now confirmatory.

MR. LANE: Yes. Issue 5 is the steam bypass. That is true. Now we just got Amendment 17 a few days, and I looked through that. The issue had to do with manual operator action of the sprays, and the analysis provided to date was assuming or was justifying the fact that the applicant had 30 minutes before the needed to have the sprays activated.

Now the actual Amendment 17, I think there might be a typo in that, and it says that the applicant has 20 minutes to activate the sprays. Did you notice that, Norm?

MR. RADEMACHER: That should have been 30 minutes, John.

MR. LANE: Okay.

MR. RADEMACHER: We will go back and fix that.

MR. LANE: That is what I thought. So we will take care of that open item.

MR. SIESS: Okay. Now Item 6 on the secondary containment bypass leakage.

MR. LANE: The secondary containment bypass leakage, we are awaiting some information from them. They have, I would say it is a little bit of a unique proposal in that they are taking some credit for the containment pressure response and using an equipment profile envelope to justify the use of their water loop seals on some equipment drain lines and RWCU lines.

We want to take a close look at that. That is unique and it looks like they end up with a relatively small excess capacity there in their loop seals. So we are doing to try and take a close look at that, and it is hard to tell where we will be on that. But once we can get some information and sit down and talk with them, I think we will be able to make some progress.

MR. SIESS: If the loop seals aren't effective, do we get large leakage, or do we get leakage in the Part 100 dose calculation range?

MR. LANE: Well, because it is bypass leakage, it wouldn't be processed at all. My understanding is it would be probably significant from the Part 100 type of consideration.

MR. SIESS: Part 100?

MR. LANE: Yes.

MR. SIESS: Have you any idea how significant it would be in terms of severe accident analysis?

MR. LANE: From that point of view, I think it probably would be low.

MR. SIESS: Okay. So we have got two different sets of thinking going around.

MR. LANE: Yes.

MR. SIESS: One is the old approach, which is the regulations, and the other is the severe accident issue of containment integrity, which is sort of a different ball park.

And on the GDC-55, you said you were working on that.

MR. LANE: The exemption request for containment isolation, that I think is going to be petty straightforward.

The last item, containment leak testing, the ball valve exemption, we have indicated previously that we have not granted that exemption for similar Mark II's. So that one is a little bit undecided at this point. It looks like we are going to have to see something new on that one that we haven't seen so far. That is the Type C exemption for the ball valve.

MR. SIESS: I assume it is possible to do a Type C leak test on the Tip ball valve?

MR. LANE: It is, right. The other Mark II's are doing it.

MR. SIESS: Other plants do it.

MR. LANE: Yes.

MR. SIESS: Okay. Are there any other comments you would like to make on this?

MR. WEINKAM: Yes, sir. Back on Item 6, as I understand it, we are still waiting for some information on preoperational testing of the main steam isolation valves, which I assume will occur later in the test program.

MR. RADEMACHER: That is correct.

MR. WEINKAM: But the staff has factored in the six standard cubic foot per hour and the LOCA doses as the value, and that is why it is an open issue.

MR. SIESS: Okay.

MR. ZALLNICK: Dr. Siess, we would also like Mr. Abbott to make a comment on the Tip test and on the Tip system.

MR. ABBOTT: Your question was is the ball valve itself testable. The design of Nine Mile 2, in order to test that ball valve, we will have to go in the dry well and disconnect Tip tubing from the Tip indexers, and then install test connections with valving in order to do that Type C test on those valves.

MR. SIESS: Is that different from any other BWR?

MR. ABBOTT: I do not know that.

MR. SIESS: I wouldn't think your Tip design was any -- anybody here from GE?

(No response.)

MR. ZALLNICK: We will try and get an answer on that from GE and get with you on that.

MR. SIESS: Okay.

Jesse, you had some other questions on containment I think.

MR. EBERSOLE: No, I am done.

MR. SIESS: Okay. We can pick that up later.

MR. LANE: Could I make a few comments about that?

MR. SIESS: Yes, sure.

MR. LANE: Not about the Tip system, but . out other things. We have just looked through Amendment 17, as I mentioned before, and we have some other areas that we had talked abcut and are essentially resolved.

One of the other ones regarding containment leak isolation was the installation of two barriers on testing and venting lines when they occur before the outmost isolation valve.

We have reached agreement on that with the applicant, although in the last amendment there were a couple of lines that did not have the isolation on that. So I think we are heading in the right direction on that one and it might have just slipped though and not been picked up.

The other thing on the containment analysis itself, the pressure analysis, Amendment 17 indicated that the original analysis was based on 124 to 127 downcomers, and the revised number is 121 because some of them are blanked off for use for RHR SRV lines or relief discharge lines.

Our experience has been that there might be a slight increase in containment accident, P sub A, on the order of a couple of pounds, and it might also affect the pool dynamic loads. So we may be talking with the applicant, or we will be talking with them rather about putting a handle on how much the containment accident pressure increases. If it is on the order of a couple of pounds, it will affect the leak testing because we use the P sub A value of 40 as the guide for the leak testing, and that would have to be increased if the accident pressure went up, and they have a design capacity of 45. So they most likely won't have any problem with design capacity. it is just a question of affecting the leak test.

MR. SIESS: Okay. Thank you.

That concludes the item on containment and that brings us to fire protection.

MR. ZALLNICK: Yes, sir. The presenter for fire protection is Mr. Norm Rademacher, who has previously presented.

(Slide.)

MR. RADEMACHER: Good morning.

My name is Norm Rademacher, Nuclear Design Coordinator. With me today I have Bob Raymond who is our Supervisor of Fire Protection Nuclear, who is available to answer questions as well.

Bob has an SRO and has a multitude of years of experience in our nuclear program and before that with the company.

(Slide.)

In Chapter 9 of the FSAR we provide a description of the fire protection program and discuss our defense in depth approach.

The responsibility of the fire protection program rests with the General Superintendent of Nuclear Generation who is on site and he delegates that through his supervisory chain.

We meet the requirements for the personnel qualifications that the staff has outlined in the Branch Technical Position 951 and we have a working ongoing program which is in existence at Unit 1 which will be utilized at Unit 2.

(Slide.)

Our fire brigade for Nine Mile Point, Unit 2 is on site now. They are in the process of training and learning the Unit 2 systems. We have an existing quality assurance program for fire protection which we are using for Nine Mile 2 and it is described in FSAR documents.

(Slide.)

We have performed the fire hazards analysis. The fire hazards analysis also incorporated the capability of safe shutdown analysis and associated circuits which includes spurious operation.

(Slide.)

As I indicated during the site tour in my presentation I would show you some of the separation in the control building, this kind of puts it in perspective. It didn't show up in the model and you might be able to see a little bit clearer.

Basically this is a view of the control building. Looking north on the top floor you have the control room. The next floor down is the relay room and then the switchgear rooms remote shutdown panels and then we go into our tunnnels which provide separation around the plant.

As you can see, the chases, we have division on the left side and division 2 on the right side. Our yellow and green are main divisional power separations.

From there we go through the tunnels as was shown to you in the control room. On the sketch through the tunnels I couldn't get it all on the slide. They go up around on the north side of the reactor building. So you have to just kind of show the green around that top part, and the yellow comes in from the south.

(Slide.)

Basically what we have done is split the reactor building in half with a north/south center line and this is a cross-sectional view. You have your two aux bays, your north aux bay and your south aux bay with the no-man's land in between, so to speak, and that is basically the separation. This separation allows us to basically meet the 10 CFR 50 Appendix R requirements for safe shutdown.

We did indicate that we were going to discuss transient combustion. So I will have Bob Raymond come up and talk about that.

MR. SIESS: Before you leave this, I think this arrangement will be of particular interest to the full committee, the separation, and I would suggest in lieu of the slide you had that you couldn't get everything on.

MR. RADEMACHER: We will fix it.

MR. SIESS: I liked the one you had during the site visit that started with the diesel buildings.

MR. EBERSOLE: May I ask this. This separation, the yellow and the green implies a compartmentalized approach to it. But isn't it a fact of life that eventually if I look at the cables I converge to Reg. Guide 175, degrees of localization, even though at the extremity you did this?

MR. RADEMACHER: Basically, as I mentioned during the site tour, the cases where we have non-meeting 175 or we have provided an analysis or whatever you want to say, it occurs basically for the most part in the control room and the relay room.

MR. EBERSOLE: I know, right.

MR. RADEMACHER: And for those cases then we go to the remote shutdown panel.

MR. EBERSOLE: Really what I am saying is this looks very good. But then on the other hand, in the darkness, the normal functioning of this to the control room eventually converges down to the speading room from the control room to Reg. Guide 175, fire suppression.

MR. RADEMACHER: That is correct.

MR. EBERSOLE: So the massiveness of this separation concept is somewhat abrogated by the ultimate confluence, except for the backup control.

MR. RADEMACHER: That is correct.

MR. EBERSOLE: Okay. And that backup control, could I say that it has a degree of tertiary compartmentalization similar to this?

MR. RADEMACHER: Yes, as you remember, we keep the rooms separate.

Bob Raymond is going to talk about transient combustibles control.

(Slide.)

MR. RAYMOND: Good morning. I am Boy Raymond.

You were asking about combustible controls. We have administrative procedures in effect at Nine Mile 1. Those procedures will be used at Nine Mile 2 also. In other words, we are developing them right now and we will put them over in effect at Nine Mile 2.

Combustible materials right now at Nine Mile 1 have designated storage areas. Any combustibles that are not stored in those areas, like for an outage or something like that, must have a permit written by the fire department. We then notify the fire chief and he then does extra surveillance on these areas so that we know if something happens in those areas.

MR. EBERSOLE: But it is administrative. I guess it is the degree of discipline that is interesting.

MR. RAYMOND: That is correct.

MR. EBERSOLE: I guess the old expression used to be that -- well, I won't use it. But does everybody get the word?

MR. RAYMOND: I would say yes.

MR. EBERSOLE: Like the janitor.

MR. SIESS: Well, you have been operating Nine Mile Point Unit 1 for how many years now?

MR. RAYMOND: Since 1969.

MR. SIESS: That is 15 years plus. What has been your experience in terms of fires from imported transient combustibles? I assume you have had similar administrative procedures in effect at Nine Mile 1 for a good part of that time anyway.

MR. RAYMOND: That is true. From day one we have had four fires at Nine Mile 1. We had one in the shaft pump that was oil that had soaked down in the insulation. We had one during an outage where a welder's cable in the dry well shorted out on the steel grading. We had another one where the welder's contractor's cables going through a penetration shorted out, and we had one in a wastebasket in a copy room. And those have been the four fires in the life of the plant.

MR. SIESS: That wasn't computer paper, was it?

(Laughter.)

MR. RAYMOND: No. Somebody threw something in the waste basket.

MR. SIESS: That is probably the biggest source of combustibles in a plant nowadays.

(Laughter.)

MR. EBERSOLE: You said you only had four fires.

MR. RAYMOND: That is right.

MR. EBERSOLE: Could you describe some interesting breaches of this administrative control where you control ---

MR. RAYMOND: Combustibles?

MR. EBERSOLE: Yes. My favorite amount of combustible is five gallons of acetone.

MR. RAYMOND: Right now we allow the operators to have only two gallons of lubricating oil anywhere in the plant and it has to be in a safety can.

MR. EBERSOLE: Lub oil.

MR. RAYMOND: Lub oil. In other words, if they have to add oil to an air compressor or something like that or to a bearing, they are allowed a two gallon can standing there in a safety can. Anything more than that has go into a combustible storage cabinet.

MR. EBERSOLE: What about acetylene tanks and dropping and busting the the neck off?

MR. RAYMOND: Pardon?

MR. EBERSOLE: What about acetylene tanks?

MR. RAYMOND: They are on carts. They are chained down and at night when the maintenance men are through working, they must go back to a designated storage area. They cannot be left at the job site. They must take them back at night.

MR. EBERSOLE: Any propane or acetone?

MR. RAYMOND: We have propane in the backyard that we use on one or two trailers and that is outside the protected area. Acetone we only allow them to take one gallon at a time in a container when they have to clean parts for the reactor.

MR. EBERSOLE: Are you going to put hydrogen in this primary system and, if so, how are you going to ---

MR. RAYMOND: We have hydrogen that goes in the generator, but that is all.

MR. EBERSOLE: Do you contemplate putting
hydrogen in the primary loop?

MR. ZALLNICK: We were going to discuss that
during the IGSCC presentation.

MR. EBERSOLE: Oh, that will be later. Okay.

MR. SIESS: Do you have any more questions?

MR. EBERSOLE: I have none.

MR. SIESS: Thank you, sir.

Gentlemen, it is now 11:53. We are now at an
item were you are going to change the slides, is that
right?

MR. ZALLNICK: I believe we are already changed
on the slides. We can either go into the next presentation
or we can respond to some questions that were asked earlier
this morning that we have answers for.

MR. SIESS: Okay. Let's do the latter for a few
minutes and then we will break for lunch.

MR. ZALLNICK: Okay. Mr. Rademacher, if you have
the responses on those questions from this morning.

MR. RADEMACHER: There were several questions
that you have asked, and I guess they go basically over a
vast variety of questions and answers.

Relative to the height of the wall of the
transformer in the yard, we don't believe that there is any
specific criteria, either NFPA or whatever that determines

what the height of that transformer is based upon our checking.

However, I don't know if Doug pointed it out or not during his presentation, and maybe that is why it didn't come out clear, was the fact that the reserve station transformers are approximately 50 feet spread apart with an aux boiler in between. So that there is separation provided.

The next question was provide some examples of system interaction changes. One of the design reviews for the aux boiler which was described earlier had the relief valve for the aux boiler going to the roof, and there is a potential, although unlikely, that the aux boiler could become contaminated and therefore release through the relief valve directly to the environment without being monitored.

Our design review of that system identified this concern and the aux boiler reliefs were rerouted within the building to a drain sump I believe so that it would go through a monitored release.

MR. EBERSOLE: Let me ask you a general question. I saw a hugh CO-2 tank, and I recall an incident years ago where I asked some engineer did he include the possibility that that would explode, and he says NRC vessels never explode, and I pointed out to him that was only the main vessel.

What is your rationale about vessel failures such as CO2 tanks?

MR. RADEMACHER: We have analyzed what happens to the CO2 in our control room habitability study. Is that what your question is?

MR. EBERSOLE: Well, just the physical impact of explosion. Do you account for that in a CO2 tank?

MR. RADEMACHER: I believe as part of our review we did look at it, and I believe the axis of the CO2 tank is such that it would hit the safety related structures.

MR. EBERSOLE: Would it become a projectile?

MR. RADEMACHER: It possibly could, but it would go I believe down the axis passageway.

MR. EBERSOLE: What about the gaseous pressurization of the spaces into which it would feed?

MR. RADEMACHER: Well, if it went through the wall out into the yard it wouldn't.

MR. EBERSOLE: If it just went into the room, are any of them so closed that it will be a structural wreck? You know, one of our plants recently blew its doors off in a test of the CO2 system, and it didn't have to have a tank failure.

MR. RADEMACHER: I don't have an answer for that question right now.

MR. ZALLNICK: We will check that one.

MR. RADEMACHER: The next answer that I had was relative to instrument line failures and a closure of high pressure, low pressure interfaces.

George Moyer, our Station Shift Supervisor will respond to that question.

MR. MOYER: The design of our instrument is such that it takes a hundred pound DP difference via the pump discharge over the reactor pressure in order to open the valve coincident with a LOCA signal. If the instrument was broke inside the dry well, you could not get that hundred pound until the pump started. The valve is designed to open against a 750 pound DP. Therefore, you would not overtorque the valve.

MR. EBERSOLE: What about the suction valves in the reactor cooling mode and the interlocks that prevent coupling of the low to high pressure systems and their degree of independence? I want to keep the 1100 pounds out of the 400 pound system.

MR. MOYER: Those valves isolate at 128 pound DP.

MR. EBERSOLE: Can they physically open against a differential which would be destructive of the low pressure side?

MR. MOYER: They are not capable of opening up against the differential of the reactor pressure.

MR. EBERSOLE: You mean they don't have enough torque?

MR. MOYER: They don't have enough torque. The breaker would trip.

MR. EBERSOLE: I am talking about the opening mode now.

MR. MOYER: Right.

MR. EBERSOLE: Was that deliberate?

MR. MOYER: It may have been. I don't know.

MR. EBERSOLE: Of course, they have interlocks to prevent energizing them anyway, right?

MR. MOYER: Right.

MR. EBERSOLE: Thank you.

MR. RADEMACHER: The next question that you had related the quality assurance Categories 1, 2 and 3 and have we had any problems implementing CAT 1 versus CAT 2 versus CAT 3 and Carl Terry, our Manager of Engineering, will respond to that.

MR. TERRY: What I wanted to mention on that is in terms of the programs themselves the QA programs are the same in terms of the procedures and that kind of thing regardless of whether it is QA Category 1, 2 or 3. So the inspectors don't have to shift gears in terms of documentation and that kind of thing.

Secondly, in terms of the application of the requirements, what is developed for each inspection is an inspection plan. There are unique inspection plans for each equipment specification and a set of requirements that are imposed on a vendor for that and a separate set of inspection plans relating to the shop inspections that I talked about.

And then, also, there is a separate documentation check list that goes along with that. Additionally, in terms of the field inspection, similar procedures are applied, and inspection planning for Category 2 and Category 3 inspections are separate inspection plans from the safety related inspections. Additionally, in terms of the drawings, the drawings identify non-safety related and safety related.

So in terms of what the inspector has to do in terms of form work and paperwork and that kind of a thing, it is all the same. In terms of the inspections themselves, there are unique inspection plans which describe exactly what attributes are to be inspected and what criteria are to be applied for those inspections.

And, frankly, the reason we apply the same program is to avoid confusion and we think it has been successful.

MR. SIESS: Thank you.

MR. EBERSOLE: If I can return for a moment to the RHR. There has been a rash of LER's come in that begin to indicate that maybe there is not enough supervision to prevent pumps from starting with an adequate NPSA such as with closed valves ahead of them. This would be unfortunate for the RHR pumps as a case in point because that is the way you get the heat out of the containment.

What is your general engineering rationale for either providing NPSH trip of the pumps or waiting for the signal to go to the operator and let him trip it?

In general, a big, important pump has NPSH protection. It will stop and it will trip out and smaller ones don't. How did you approach this problem?

MR. RADEMACHER: I believe, for example, on HPCS or RCIC there is a suction pressure switch that transfers, for example, low water from the condensate storage tank to the suppression pool.

MR. EBERSOLE: It goes to another source?

MR. RADEMACHER: Yes.

MR. EBERSOLE: And doesn't trip the pump?

MR. RADEMACHER: No.

MR. EBERSOLE: And it can ride through the transient?

MR. RADEMACHER: Yes.

MR. EBERSOLE: What about the other important ones?

MR. RADEMACHER: I believe on the RHR system, I believe that does have a pressure alarm, but I don't believe it is a trip. We all have to check to confirm that.

MR. EBERSOLE: If it is the alarm, then that goes with a question, will you get a proper action after the alarm before damage occurs?

MR. RADEMACHER: George Moyer, our Station Supervisor has an answer to that question.

MR. MOYER: We have suction interlocks such that if the suction lineup is not correct, the pumps won't start.

MR. EBERSOLE: Oh, but if I have already started them and then I close the valve?

MR. MOYER: It will trip the pump.

MR. EBERSOLE: So you have automatic trip.

MR. MOYER: It puts the trip signal into the pump.

MR. EBERSOLE: That is conservative. Many plants don't have that.

MR. EBERSOLE: On the other hand, there will be people who argue that is another way to make the pump quit.

(Laughter.)

MR. SIESS: Anything else?

MR. EBERSOLE: No.

MR. SIESS: Okay.

MR. RADEMACHER: There were a couple more items here that we owe you.

MR. SIESS: You have got a couple more. Go ahead.

MR. RADEMACHER: Relative to the Unit 1 feedwater check valve analysis that you requested, we have not performed a Unit 1 feedwater check valve slam shut analysis.

MR. EBERSOLE: Do you intend to do that, or does the staff have any generic action in that aspect?

MR. RADEMACHER: I don't believe the staff has any generic action in that, but I would let them answer.

MR. EBERSOLE: Does the staff have any comment?

MR. RADEMACHER: This has to do with the slamming shut of a check valve?

MR. EBERSOLE: Yes.

MR. WEINKAM: Again, since I am not the project manager normally, usually I know that that is something which the mechanical engineering branch looks at. I will have to check through the SER and see if there is anything on that.

MR. TERRY: This was in reference to Unit 1. I am sorry.

MR. EBERSOLE: They are looking at Unit 2 here, and we don't know what is going on in Unit 1 or generically I believe.

MR. SCHWENCER: No, we don't at this meeting. I think we could be prepared for that later.

MR. SIESS: Check with the mechanical branch and they could comment at the full committee meeting.

MR. EBERSOLE: It is a long standing issue, you know, like 15 years.

MR. TERRY: I can provide a little bit of input on that. In regards to Unit 1, we have not performed an analysis of that. However, one of the thing that we will do is take a look at the results of the Unit 2 analysis and determine whether there is any applicability on Unit 1. So we will at least be doing that.

MR. EBERSOLE: Earlier on we talked about spacial interactions, and you might have a plant interaction if that should occur, you know, continuous discharge from the boiler into space.

However, that would not damage shutdown equipment, would it?

MR. TERRY: That is correct.

MR. SIESS: Mr. Rademacher, another item?

MR. RADEMACHER: Yes. The last item was just a matter of clarification. When we indicated that the MSV line can handle water, that is the MSIV's closed, and the MSIV's do get a trip signal in high water. So I just wanted to clarify that.

MR. SIESS: That concludes your updates?

MR. RADEMACHER: Yes, sir, at the present time.

MR. SIESS: Okay.

MR. TERRY: One more. There is one statement that I would like to correct that I made during the presentation. I did indicate that the dry well coolers could be fed from service water. That was not correct.

MR. SIESS: We will recess one hour for lunch, and that says we will reconvene at 1:07, and I mean it unless I am late for lunch.

(Laughter.)

(Whereupon, at 12:07 p.m., the subcommittee recessed, to reconvene at 1:07 p.m., the say day.)

AFTERNOON RECESS

(1:10 p.m.)

MR. SIESS: The meeting will reconvene.

Are you all ready to go?

MR. ZALLNICK: Yes, sir, we are.

MR. SIESS: Okay. The next item then is the control room. We have four sub-items on that.

MR. ZALLNICK: Yes, sir. The presenter for the control room is Mr. Douglas Pike, who is the System Project Engineering Manager.

(Slide.)

MR. PIKE: Good afternoon, gentlemen. This is Doug Pike again.

(Slide.)

I would like to start off this afternoon with our control room design review.

Nine Mile Point is conducting a control room design review in accordance with the guidance provided in NUREG 0700 and 0737, Supplement 1.

(Slide.)

A little history of what Niagara Mohawk has done in this area. As has been alluded to earlier, we have had extensive participation in the system designs in this plant, both our engineering and operating people.

Back in the time period of 1976 and '77 we held some PGCC design reviews, and one of the main reasons we did that was to determine what kind of controls, indications and alarms we wanted to see in our control room.

That was followed up in 1977 by an actual panel mockup review where we had wooden mockups of the bench boards made, and with utilizing stick-on controls we went through there and arranged the controls in the configuration that we would prefer to see them in.

Following that review, we went through another series of final design reviews when our design was a little more finalized in 1978 and '79, which was basically a repeat of the earlier PGCC reviews.

Again, the people involved in these reviews were experienced engineering and operating people. Our operating people were both familiar with the Unit 1 and the James A. Fitzpatrick design.

As a result of those reviews, I think we documented in excess of 500 open items which either asked for some kind of an addition, deletion or change or an evaluation of design.

In 1982 we actually conducted a mini human factors review of the control room panels as they actually existed and were staged at the PGCC plant in San Jose, California.

For these reviews we contracted with some of the BWR owners group experienced personnel in their design review programs and a human factors specialist from MIT.

We formed two separate review teams, one the owners group review team and another review team utilizing experience in Niagara Mohawk operations and engineering personnel and Stone and Webster Engineering human factors people. We documented the findings from that review and we have incorporated a number of those findings into our panels.

MR. EBERSOLE: May I ask a question. As you know, historically we have the ECCS systems for large LOCA mitigation and segregated them and made them easy to look at in the control room, the theses being you would know, you could take a lumped view of them as an operator and see whether they were doing what they were supposed to. But historically over the years there has been no identification of the accident shutdown mode which did not involve LOCAs, but perhaps involved fires or other physical damage to the support systems.

For that reason there has never been an integral identification of the support block of equipment, nor no attempt to organize it on the control boards in any comprehensive way like ECCS was.

Did you all deliberately decide you didn't want to do that?

MR. PIKE: I guess I am not sure I understand the question.

MR. EBERSOLE: What I am saying there is a minimum set of shutdown equipment that is type represented on the aux shutdown board which you didn't have some years back. That typical set of equipment is scattered all over the control board.

MR. PIKE: Right, I understand.

MR. EBERSOLE: Yet, the ECCS was blocked as a set for supposedly operator efficiency and seeing that you could mitigate a LOCA which was the original only accident you ever had, when the real accident is always going to be degradation of support equipment for shutdown.

Has there ever been any discussion about organizing the critical shutdown equipment non-LOCA?

MR. PIKE: I don't believe specifically. I think rather our rationale was we normally have electrical boards with the electrical equipment on it.

MR. EBERSOLE: It is the classical compartmentalization.

MR. PIKE: Yes, that is correct, and the ECCS groupings are normally together on an individual panel.

MR. EBERSOLE: So they are compartmentalized in the old original functional logic?

MR. PIKE: Yes.

MR. EBERSOLE: And the operators are used to that and they like it?

MR. PIKE: That is correct.

MR. EBERSOLE: Okay.

MR. PIKE: As far as our Unit 2 specific program, we have developed our overall plan and it has been submitted to the NRC. Incidentally, we have received comments back from the NRC on that. I believe there were some 18 concerns identified and some five recommendations and we have recently responded to those.

Our overall program is managed by our own lead control room design review engineer and we have contracted with the human factors consultant to provide technical support in that area and prepare the kinds of implementing procedures and reports that are required.

(Slide.)

Just an overview of our organization.

We have an executive team made up of the Vice President of Nuclear Generation, Mr. Lempges, and our Nine Mile Point, Unit 2 Project Director to provide executive overview of the program and to approve the recommended changes in the final report.

We have a management team made up of basically the management level personnel on the project and within

Stone and Webster to basically again review recommended changes, the task completion reports and the final report.

Then we have the review team made up of our Niagara Mohawk team leader, advisory operations engineer from Stone and Webster, our architect/engineer, station shift supervisor from operations, a startup and test engineer from General Electric, a member of our training staff and again, the human factors people. And these are the people who do the day-to-day work of the program of developing the plans, conducting the surveys and so on and so forth.

(Slide.)

The phases of our program generally follow the guidance of NUREG 0700, that is operating experience review, inventory reviews, task analyses, panel surveys and the assessment improvement program.

(Slide.)

Some additional features of our program. We will be looking at the SPDS system integrated into the review process for compatibility from a human factors standpoint. We will also include the remote shutdown panel in our review, and we will also take a look at the technical support center in the emergency offsite facility layout from a human factors standpoint.

I just, incidentally, have a few pictures here of our control room which I thought I would run by quickly if you are interested, and I think you have pretty much seen these.

(Pictures shown.)

That concludes my presentation on the control room design review.

I believe the next topic is the safety parameter display system, and the basic requirements are from NUREG 0737 that you have a concise display of critical plant variables to aid the operator in determining the safety status of the plant and that they should be located in the main control room.

(Slide.)

The Unit 2 design features of our system. We utilize a Quadrex/Honeywell computer based system. Honeywell supplies the hardware and the emergency response facilities portion of this system. Quadrex has done the software for the SPDS displays themselves.

This is an integrated system. It is combined with a computer system that also provides control of the rad waste processes and monitors generator temperatures.

Our goals for this system, an unavailability of .01 when the reactor is operating and an unavailability of .2 when the reactor is shut down.

We do utilize two redundant 4500 central processors, Honey central processors with auto failover and we provide an uninterruptible power source for that system.

(Slide.)

As far as the SPDS displays themselves, we utilize all but one of the BWR owners group recommended SPDS displays. The one we don't use is the radioactivity controlled display. The reason we chose not to use that is we have a computer based digital radiation monitoring system that has equivalent information on a CRT right next to the SPDS displays.

Our displays are based on a subset of the Reg. Guide 1.97 post-accident monitoring parameters.

(Slide.)

On the left-hand screen you see a display of the level one overall display. This provides the overall status of the plant in real time values. We have a green/red color coding concept with green being a normal condition and red an abnormal condition. It provides rate and trend information for the variables. It provides on the bottom the overall status of the level two displays, and it also has a function to provide the position that the mode switch is in.

This is an example of our level two displays. This one happens to be reactivity control. Again, we have level two displays for reactivity control, core cooling, coolant system integrity and containment integrity.

The level two displays provide a time history of the past six minutes of the variables, real time values of the variables, trend rate information, again the status of the other level two displays, time, date, title information and again mode switch position.

(Slide.)

As part of this SPDS system, we also have what we call the ERF, or emergency response facility system. This data system is made up of all of the Reg. Guide 1.97 parameters, some 552 analogue and digital points. This provides information on the SPDS CRT's available in the control room, the technical support center and the emergency offsite facility.

You can compose as many as 180 graphic displays with a similar type thinking as the SPDS displays. You can put up bar charts, trending of variables, alarm summaries and group logs.

(Slide.)

We also have a historical event retrieval in this system. The retrieval system has trigger signals such as a LOCA, scram or other signals that initiate transients that start the recording and when initiated it will give you 2 hours pre-event and 12 hours post event information for those signals that are in the data base.

They are recorded in 1, 5 and 30-second time periods. And following this 12-hour post-event period, it has the capability to print out these variables every 15 minutes for up to 14 days.

This historical data system is accessible from the TSC and the EOF.

MR. EBERSOLE: May I ask you a question about this system. Again, historically it is oriented to the LOCA. It is not oriented to an industrial accident like a fire because it might be the victim of a fire. So it could get to be an attractive hazard because the operators just dearly love to use it. It displays virtually everything, but it is not qualified. It comes from cables that are intermixed with non-safety and safety and so forth.

So I want to ask you what do you do to counteract the tendency of the operators to get comfortable with this thing and not realize it is going to be with them for the non-LOCA accident?

MR. PIKE: I would have to say training.

MR. EBERSOLE: Well, do you scrub this every so often on an exercise and say we have some ---

MR. ZALLNICK: I think Mr. Abbott can respond to that.

MR. PIKE: I think maybe the operators should respond to that.

MR. ABBOTT: All our procedures written for the operator to operate the plant without the use of SPDS.

MR. EBERSOLE: Do you occasionally throw him into a transient on the simulator and then deprive him of all this nice information?

MR. ABBOTT: The SPDS is currently not available in the simulator.

MR. EBERSOLE: Oh, it is not?

MR. ABBOTT: No. But I think I mentioned yesterday that we will do our transients in the simulator without even the use of the computer to make ---

MR. EBERSOLE: Well, on the simulator you still have the privilege of scrubbing the non-qualified instrumentation as an exercise and watching him work out a transient. Do you do that where he is asked to resort to minimum instrumentation?

MR. ABBOTT: Yes, we will.

MR. EBERSOLE: Okay. Thank you.

MR. PIKE: The next presentation is control room habitability.

(Slide.)

Again, NUREG 0737 requirements basically requires radiological protection and toxic gas protection for the operators in the control room for accident events. The Unit 2 design basically complies with this for the use of shielding and a safety related heating and ventilation and air conditioning system.

(Slide.)

The accident scenarios that were used to develop the requirements, the four bounding accident cases were the LOCA, main steamline break outside containment, control rod drop accident and the fuel handling accident.

The results of the analysis were done in accordance with the regulatory guide and standard review plan for the 30-day habitability limits, and basically the results of our analysis showed that all 30-day limits were satisfied.

MR. EBERSOLE: Which was the worst of these, the LOCA I guess, right?

MR. PIKE: I am not sure. I think Mike Stockne' probably can address that.

MR. EBERSOLE: I think the SER shows the LOCA.

MR. PIKE: Yes, it is a LOCA.

MR. EBERSOLE: Now let me ask you this. The LOCA is an accident where very pessimistic viewpoints were taken for fuel failure and then that was then automatically then compensated for by the hypothesis of very tight containment.

There has been some question raised about, at least if you are depending on automatic containment closure whether it is as leak tight has it is supposed to after you do it automatically.

Do you ever look at extrapolating what might be the dose in the control room if the containment leaks a little bit more than you thought it should, especially in a two-unit plant because it would involve the second unit.

MR. PIKE: Well, I am sure that there was a lot of conservatism placed in the calculations and so forth. However, I don't know whether we postulated that a containment isolation valve did not isolate.

MR. EBERSOLE: Or leak somewhat more.

MR. PIKE: I would just as soon have Mike Stocknoff of Stone and Webster address that.

MR. EBERSOLE: It gets interesting for multi-unit plants.

MR. STOCKNOFF: We did look at leakage from the primary to secondary. It all goes in the secondary and we all assume that it gets through the charcoal filtration at that point.

We also did a 90-second exfiltration with no standby gas treatment system working, and that is also part of or analysis. We also looked at the effects on Unit 1 from a LOCA and also from a LOCA on Unit 1 to Unit 2.

MR. EBERSOLE: Well, what sort of margins do you have in respect to control room environment for increasing the outside dose beyond the standard dose that you get from an intact and perfectly working containment. Do you double the concentration of triple it?

MR. STOCKNOFF: We assume by what you mean by perfectly working containment ---

MR. EBERSOLE: With the specified leakage.

MR. STOCKNOFF: Theoretically the containment is supposed to hold and you exfiltration. We assume maximum exfiltration of 3500 standard cubic feet per minute. We base our analysis on that and we are still well within the 30-rem limit.

MR. EBERSOLE: I see.

MR. STOCKNOFF: So theoretically you would see, if it would be tight you would see zero, but we assume maximum leakage.

MR. EBERSOLE: So you think you have got a good margin for a control room environment?

MR. STOCKNOFF: Oh, yes.

MR. SIESS: When you are talking about exfiltration, are you talking about primary containment or secondary?

MR. STOCKNOFF: Secondary containment.

MR. SIESS: What are you assuming for primary containment filtration?

MR. STOCKNOFF: All leakage from the primary goes to the secondary, and then for 90 seconds nothing would get filtered from the primary to secondary. It is basically an instantaneous release from the primary to the secondary.

MR. SIESS: I am sorry, I didn't get the answer. What do you assume for primary containment leakage?

MR. STOCKNOFF: 1.1 percent.

MR. SIESS: 1.1 percent. Thank you.

MR. RADEMACHER: We also assume there are some leaks in the RHR system, a water leak into secondary containment. That is part of the analysis as well, and I think it was 10 gpm.

MR. STOCKNOFF: I think in the last analysis we assumed one gpm.

MR. EBERSOLE: Do you look at the elastomer seals on the RHR pumps for radiation dosage and the possibility of entraining insulation debris and griding them up?

MR. RADEMACHER: As part of our mechanical equipment qualification program we have looked at the radiation doses on elastomers as part of that program to verify that they are capable of withstanding the radiation doses after an accident.

MR. EBERSOLE: Does this plant still use hydroclones to strain the water that goes to the seals and journals against any contaminants that might damage those elements?

MR. RADEMACHER: I am sorry, I didn't hear the first part of your question.

dR. EBERSOLE: In the standard BWR's that I know, they have water lubricated and cooled seals and journals on their RHR pumps. These are protected against debris in the suction water presumably by little centrifugal separators called hydroclones in many designs. I never found a basis for the hydroclones knowing that they are to separate something with a specific gravity greater than one or less than one, but they will only do it one way.

This question always comes up when one looks at the kind of insulation you put on the equipment inside the containment and you begin to speculate on how much of it gets into the water and how much you concentrate in these hydroclones.

MR. TERRY: To provide a little bit of input on that or insight, we do have the separators, but in terms of taking a look at insulation and the possibility of that getting into there, first off, there is a limited amount of glass or bead type insulation that is used. Where it is used, it is encapsulated.

However, you could postulate certain scenarios where that could be blasted off. It would then have a path to take on down from the dry well floor. Bear in mind, we have got basically a curve on the downcomers themselves.

That would be the path that it would travel, anywhere form three to six inches. And I think you recall also that there is a cover on the downcomers themselves.

Then of course getting in to the pool, there are two types of insulation. One would sink and the other one would float and the intakes for the RHR are approximately 8 to 10 feet below the surface of the water. We did that to prevent vortexing.

MR. EBERSOLE: But in the initial stages isn't there such turbulence that you have pretty much a homogeneous mix of whatever the stuff is and don't you reconcentrate it in the hydroclones?

MR. TERRY: Well, yes, that would be to some degree. But what I was leading up to is even if it did get into the pump seals, GE has looked at that. My point was that it is a difficult path to get there. But even if it did, GE has taken a look at that and basically we don't anticipate any damage in terms of the pumps themselves.

MR. EBERSOLE: Well, I have heard two stories about these. Some owner/operators are getting rid of their hydroclones on the grounds they seem to offer more of a hazard than a protection. I don't know the full story, but I wonder if you have looked at the relative merits and demerits of that?

MR. TERRY: I don't believe we have in terms of the hydroclones themselves.

MR. SIESS: The argument has been a deep bed large filter would be better than that. Well, one argument is how do you know whatever you are going to strain out is heavier or lighter than water, and what is the design basis of the hydroclones. I have never found that yet. Have you? What is the source term?

MR. TERRY: I can't answer that.

MR. EBERSOLE: It is put there, but, by George, I have never found out what it is supposed to strain out. What are they supposed to strain out?

MR. TERRY: The types of debris?

MR. EBERSOLE: Evidently it must be something heavier than water.

MR. TERRY: Yes.

MR. EBERSOLE: But is it?

MR. TERRY: In terms of what we are talking about, like I say, there are a couple of different types and in one case it would not be. It would float.

MR. EBERSOLE: That is right. So that would feed this stuff to the very place you want to protect.

MR. TERRY: But, again, I think it is important to point out that even if that were to happen in terms of seal capability and pump capability, we would not jeopardize that.

MR. EBERSOLE: The interesting thing is the Westinghouse seals and journals are not subject to this problem. They are totally different. I think it would be interesting if you could find out just what the rationale is for the design of the hydroclones in the context of what is the source term of the contaminants and what is its characteristic, specific gravity and very nature that you are trying to get rid of it for.

Could you maybe do that?

MR. TERRY: Yes. We will have to take a look at that.

MR. EBERSOLE: Okay.

(Slide.)

MR. PIKE: We have also performed the toxic gas analysis for control room habitability looking at nitrogen, sulfuric acid, carbon dioxide, propane, halon, hydrochloric acid and chlorine. The basic conclusion is that there is no potential for operator incapacitation from these sources.

(Slide.)

Some of our design features, shielding-wise, and we are talking about the control room now. The walls and roof are two-feet thick reinforced concrete. The interior walls are at least one-foot thick. The floor is at least nine inches thick. The leak tightness of the control room. the concrete floors and walls are essentially leak tight.

We have leak tight access doors to the control room that are self-closing and our penetrations through the control room walls are sealed with leak tight fire retardant material.

(Slide.)

The screen on the left is a schematic of the control room HVAC system. The system is a fully redundant safety grade system seismically and environmentally qualified. It utilizes Class 1E instruments and controls. Air is taken from outside through two separate and redundant missile and tornado protected air intakes. They are approximately separated by a hundred feet and they are located on the east and west side of the control building.

This provides a control room boundary with 75 degree and 50 percent relative humidity conditions. It provides a positive pressure in the control room of about .125 inches of water.

MR. EBERSOLE: May I ask this. As Arkansas Nuclear 2 found out, the whole plant ran on the air conditioning system for the control room and they had a lot of trouble in line outage for many days because of some inadequacies in it.

Due to the new presence of solid state equipment in the control equipment, I believe this new shutdown equipment complex does contain numerous uses of solid state equipment, right, or does it?

MR. PIKE: Are you talking about the remote shutdown?

MR. EBERSOLE: No, no, no. Just the normal shutdown process, does it involve controls and auxiliaries which nowadays use solid state equipment?

MR. PIKE: Well, located in the main control room are the solid state trip units and so forth.

MR. EBERSOLE: What about other critical instrumentations for just maintaining shutdown, is that solid state?

MR. PIKE: I would say much of it is, yes.

MR. EBERSOLE: And is it in consoles or metal cubicles or cabinets?

MR. PIKE: It is within the control room panels.

MR. EBERSOLE: So then I will invoke the painters that come in with their drop cloths and hang them over these modules and thus cover the little fans that control the gradient to the room air of this equipment. Now what do you dd about that?

MR. PIKE: I would have to pass that to operations.

MR. EBERSOLE: You know what I mean, don't you, during shutdown or prior to shutdown. It will shut down sooner or later if you overheat these modules.

MR. ABBOTT: We wouldn't allow that to happen.

(Laughter.)

MR. EBERSOLE: The painter is going to get the word?

MR. ABBOTT: He won't be allowed in the control room with his drop cloths.

MR. EBERSOLE: I see. Okay. You are going to lock out the painter.

(Laughter.)

All right. Thank you.

(Slide.)

MR. PIKE: Basically, the normal mode of operation of these systems is that one trains draws in a small amount of outside air and the remainder is recirculated. We have safety related chillers that provide the air conditioning feature.

Again, we maintain a positive pressure, and the special filter trains which you see down in the right-hand corner are normally bypassed in the normal mode of operation through those two normally open isolation valves on the right-hand side there.

When you have either a LOCA signal or a detection of high radiation on either one of those two radiation monitors monitors that are shown down in the bottom right-hand corner. That bypass line will isolate, the filter trains will be automatically placed in service and then all your outside air is passed through those filter trains, the boosters fans on the discharge of the train start and they aid in pushing the air back to the inlet of the main ventilating fans and portions of the recirculated air also get fed back through those filter trains.

We still maintain the positive pressure in the control room under these conditions and the operator has the ability to select either one of those outside air intakes for his source of air.

(Slide.

Some additional features of our system.

We have smoke detectors located in the air intakes and at the discharge of the main fans also that alarm in the control room that would alert the operator to any possible intake of smoke from outside.

In any event both of those air intakes can be isolated, if necessary, and again you will continue to operate with your air conditioning system on to maintain temperature and humidity.

This system does have a smoke removal capability that would require the shutdown of the main system to aid in smoke removal from the control room.

We do utilize seismic fire dampers at the control room boundary that in the event of a fire in that system fuseable links would melt and isolate the boundary from the outside fire areas.

Breathing air is available in the control room for operators should it be required and we have lighting in the control room for all conditions.

MR. SIESS: Are the operators given some training on the simulator using the breathing apparatus?

MR. PIKE: I will pass that to training or operations.

MR. ABBOTT: No, that is not currently in our training program.

MR. SIESS: Any good reason why not?

MR. ABBOTT: They are not specifically trained to operate in the control room with a breathing apparatus, but they get training in the use of that type of apparatus with a face mask, et cetera, but not actually training in the simulator.

MR. SIESS: It seems to me that in reviewing your emergency operating procedures you do walkdowns and use a simulator to see if people can execute the various maneuvers in the appropriate time and so forth.

It just seems to me that their performance with breathing apparatus might be somewhat different than that without. This is masks I assume?

MR. ABBOTT: That is correct.

MR. SIESS: Do they have to have a cylinder carried separately or they plug into something?

MR. ABBOTT: They plug just into an air hose. They cylinder is remote from them.

MR. SIESS: I didn't hear you.

MR. ABBOTT: The cylinder is remote from them. It is not on their back. I am talking about in the control room.

MR. SIESS: So they are connected up by a tube?

MR. ABBOTT: That is correct with an air hose.

MR. EBERSOLE: What was the controlling factor in locating the opposite air intakes? Would it have to do with wind or hypothetical fire locations or what?

MR. PIKE: I believe that it was again to the extent practical to provide maximum separation for those.

MR. EBERSOLE: If I had one of those big transformers catch on fire, would I catch both intakes if the wind was right?

MR. PIKE: Well, I would say given any combination of wind conditions, no, that at least one of them should be okay.

Again, that control building is located basically south of the reactor building and the turbine. The prevailing winds at the site are generally in either a southwest or northwest to west direction which would generally be below any accident releases away from those smoke ---

MR. EBERSOLE: Are there any large fire sources like oil tanks or anything that might be in line with both intakes?

MR. PIKE: I can't think of anything at least outside the plant to the west of those -- and I am thinking more to the west because of prevailing winds from the west -- I can't think of anything outside the plant to west of those intakes like that.

If anyone knows, feel free to speak up.

That concludes the presentation on control room habitability.

(Slide.)

The next presentation is on our remote shutdown capability which we have been over quite extensively already.

Again, our design does provide remote shutdown panels to meet the NRC requirements. Also, those panels will fulfill all Appendix R requirements when coupled with some local operations.

(Slide.)

Some of the panel design features.

Designed to the same standards as the main control room, seismically qualified, Class 1-E redundant instruments and controls, divisional separation at least as good as Reg. Guide 175, and we have seen that that is even better in this instance.

The panels are again separated by a three-hour fire barrier and a positive pressure is maintained in the rooms.

MR. EBERSOLE: Is there in the specifications of this system some place identification of those discrete functions which you will not tolerate being inadvertently actuated and so you have to go find out where they possible inadvertent actuation process takes place like hot shorts?

MR. PIKE: Again, for our Appendix R input we assume a major fire in the control room that could cause spurious operations. Again, the transfer switches on the remote shutdown panel will take control away from the control room and they are independently fused.

So if the fire in the control room has caused a short that, for instance, blows fuses in a motor control center some place, when you throw these switches, it also cuts in an independent set of fuses.

MR. EBERSOLE: Right. Did you identify a discrete set though of functions which you would rather not have actuated on a spurious basis?

MR. PIKE: Yes, we did.

MR. EBERSOLE: How many were there?

MR. PIKE: Oh, I am just taking a wild guess, but I would say in the neighborhood of 70 to 80.

MR. EBERSOLE: That you have to lock out?

MR. PIKE: Yes. Mainly we would either take care of that by administrative control, such as running the plant with a valve shut and the power off, or we are in the process of installing disconnect switches which can be thrown upon exiting the control room which will disconnect those circuits from the control room and therefore remove the possibility of spurious operation.

MR. EBERSOLE: In a way that is sort of an extension of the auxiliary shutdown center, the things that you must guarantee will not work?

MR. PIKE: Yes, right, for the Appendix R main fire center.

MR. EBERSOLE: Would it take a man -- do you have a check list for him to do that?

MR. PIKE: Well, the operators will be developing procedures that will tell them exactly how to respond to that situation.

MR. EBERSOLE: Are you going to occasionally test him in this mode of operation and maybe shut down?

MR. PIKE: I guess I will let the operators address that.

MR. RADEMACHER: Once a refueling cycle we will verify the capability of the remote shutdown panel. That is part of our tech specs that we have submitted to the Commission.

MR. EBERSOLE: Thank you.

MR. PIKE: Also, we will be doing a startup test in accordance with the Reg. Guide to show that you can do this from the remote shutdown panel.

(Slide.)

Again, emphasizing the physical independence, we are two elevations below the main control room again electrically separated from the control room by key lock transfer switches.

Again, we have two safety related heating and ventilation and air conditioning systems, one for each room that maintain a positive pressure.

MR. EBERSOLE: This is a troublesome system to design and ensure that you have all the corners of it patched up right. I suppose you realize this UPPS system would make it unnecessary with out 1/20th of the elements.

MR. SIESS: Jesse, you had better be careful about saying it is unnecessary.

(Laughter.)

MR. EBERSOLE: All right.

MR. SIESS: The staff may not agree with you.

MR. EBERSOLE: Well, that won't be the first time.

(Laughter.)

(Slide.)

The systems provided on the panel, service water, RCIC, the shutdown and suppression pool cooling modes of RHR.

We have four of the SRV's that also provide the ADS function, and we have the remote shutdown room HVAC system controls.

We will be adding some additional controls to this panel to accommodate the Appendix R scenario. One example that I can think of right now is we are going to add some controls to allow a long-term supply of nitrogen to the four ADS valves should they be required.

MR. EBERSOLE: Let's see, in this case you will operate depressurized. I am thinking about containment temperature, hot and dry. Will you maintain containment cooling, which is not safety related?

MR. PIKE: Well, again, we would cool by blowing down of the suppression pool through the SRV's and then cooling the suppression pool.

MR. EBERSOLE: So you are really removing the heat source.

MR. PIKE: That is correct.

As far as the indication in the control room, we have made those totally independent of the control room. They are separate instruments from the sensors in the field right through to the remote shutdown panel with no connection to the main control room.

Again, we provide critical or system indication for systems operability, in other words, flows and pressures to operate the systems on the panel.

In addition, we have added critical redundant indication for RHR heat exchangers, inlet and outlet temperatures, reactor flange and bottom head temperatures and of course reactor level pressure, suppression pool water level and suppression pool temperature.

MR. EBERSOLE: I am interested in the reactor shell flange and bottom head. I never saw that before. Why did that get there?

MR. PIKE: I think it is basically to be able to control your cool-down rate to 100 degrees per hour.

MR. EBERSOLE: I see.

(Slide.)

Some additional features of this panel.

We will have a main plant computer CRT keyboard located in this room to provide additional information from the main plant computer data base.

We have a station that hooks into the plant-wide communication system also located there. We have provided lighting for all conditions, and again the panel will -- it should be panels will be included in the control room design review.

MR. EBERSOLE: Is the plant computer or CRT keyboard critical to this shutdown operation?

MR. PIKE: No, it is not>

MR. EBERSOLE: It is just a convenience.

MR. PIKE: It is additional information.

That concludes my presentation on remote shutdown.

MR. SIESS: Thank you.

We will go to the next item, which is emergency planning.

MR. WEINKAM: Emergency planning will be presented by Mr. Pat Volza.

Mr. Volza has nine years of nuclear power experience at Niagara Mohawk and Knowles Atomic Power Laboratory and is currently the Emergency Planning Coordinator.

(Slide.)

MR. VOLZA: Good afternoon, gentlemen.

I would like to briefly give you an overview of the Nine Mile Point emergency planning program, both our onsite and offsite programs.

As was previously stated by Mr. Mangan yesterday, the Nine Mile Point Nuclear Station currently has a very successful onsite and offsite emergency planning program. It is our intention to fold Nine Mile Point directly into this existing program.

It should be noted that both the local and State offsite emergency plans have been unconditionally approved by FEMA as well as our public alert and notification system.

This approval was obtained in accordance with current FEMA Rule 44 CFR 350 and is the first in the country to do so.

With that, I would like to proceed with a short discussion of our onsite and offsite programs.

(Slide.)

The Nine Mile Point Nuclear Station onsite emergency response organization begins with the minimum shift crew and may be expanded to include other personnel as they are needed and available. Three staffing levels for emergency response have been provided.

(Slide.)

The staffing levels are shown here on the left screen.

Staffing level one consists of the minimum shift crew under the direction of the Station Shift Supervisor and would provide the initial assessment and response to emergency condition.

For events that fall into the alert or higher emergency category or as deemed necessary by the Station Shift Supervisor, a large response organization would be required.

Staffing level two provides for the augmentation of the minimum shift crew by site and corporate staff. This organization is capable of handling any emergency of short duration as well as the initial phases of a large-scale or long-term emergency.

For events that fall into the category of site area or general emergency or have the potential for environmental consequences or otherwise as deemed necessary by the Site Emergency Director, an even larger response organization would be required.

Staffing level three provides for this augmentation by additional site and corporate staff and is capable of handling large-scale or long-term emergencies.

Augmentation of the site staff is provided as the situation dictates by our corporate support, the Institute of Nuclear Power Operations out of Atlanta, the General Electric Company, our NSSS vendor, local services support, by fire, medical and other volunteer organizations as well as support by nearby nuclear facilities.

It should be noted that Nine Mile Point is a co-signatory to a letter of agreement between the James A. Fitzpatrick nuclear power plant adjacent and east of the Nine Mile Point nuclear site, and the Robert Ginna power site located near Rochester, New York.

(Slide.)

In the event of a radiological emergency or other situation resulting in need for additional equipment and/or personnel assistance, these plants have agreed to provide aid to each other. In fact, this agreement proved valuable in supporting the Ginna accident during their 1982 emergency.

(Slide.)

As you can see from the slide on the left, this support includes anything from personnel assistance to making available our accounting facilities as well as our public information people.

(Slide.)

The Nine Mile Point Nuclear Station emergency plan has been coordinated with all appropriate governmental agencies. These agencies include at the State and local level the New York State Disaster Preparedness Commission and the Oswego County Office of Emergency Preparedness.

At the Federal, the United States Nuclear Regulatory Commission and the United States Department of

Energy Office at Brookhaven.

Our international coordination has been through the Canadian Ministry of the Solicitor General.

With respect to our emergency response facilities, they have been built in accordance with Supplement 1 to NUREG 0737 and are staffed and operated in accordance with NUREG 0654.

These facilities have numerous redundant assessment and communications systems to determine the extent and magnitude of an emergency.

As with all plans that are man-made, their effectiveness can only be guaranteed through the diligence of the personnel involved in the conduct of numerous well-defined and scheduled exercises and drills.

(Slide.)

Since 1981 Niagara Mohawk has conducted three site exercises, with the fourth currently being planned for November of this coming year.

That concludes my presentation.

MR. SIESS: Any questions, Jesse?

MR. EBERSOLE: No.

MR. SIESS: Thank you, sir.

The next item is the PRA.

MR. ZALLNICK: Yes, sir. The presenter for probabilistic risk assessment is Mr. Norman Rademacher who was previously introduced.

MR. RADEMACHER: Good afternoon. My name is Norm Rademacher. I am the Nuclear Design Coordinator. I am here to talk about PRA's for Nine Mile 2.

(Slide.)

A PRA is an analysis of the adequacy of core melt accidents. We have performed a mini-PRA for Nine Mile Point, Unit 2. This PRA was performed to meet an interim rule in the Federal Register, and basically PRA's are categorized as full and mini.

(Slide.)

Our mini PRA identified accident sequences important to core melt accidents and core melt risk. It included a loss of offsite power as an accident initiator and does not include external events, such as flooding, fire or seismic.

It does include sequences of plant systems needed to mitigate core melt sequences and includes various containment failure modes.

Also it includes the results of risks and socioeconomic impacts.

(Slide.)

PRA inputs. Basically we had for inputs ---

MR. SIESS: Excuse me. If the only thing that makes this mini the fact that it is limited to internal events? I am not familiar with the full and mini. There are some other categories that have been defined. This is the first time I have heard these, full and mini. You do go to consequences?

MR. RADEMACHER: Yes.

MR. SIESS: Okay.

MR. RADEMACHER: I guess in the newer terminology it does go through containment analysis and risk assessment to the public and that kind of thing.

MR. SIESS: But it is all strictly internal events?

MR. RADEMACHER: Yes.

Accident event trees and functional success criteria were established. We used site specific offsite power grid reliability values. We developed plant specific fault trees. They were modeled around the Grand Gulf fault trees taking into account of our plant specific differences.

We did use generic component failure rate data because we don't have plant specific failure rate data.

MR. EBERSOLE: Can I comment on that just a minute. How about going back and just taking one aspect of the plant design valves and perturbing that in some sort of a range, which you can pick as well as anybody else, but taking into account what I said earlier that the statistical data is not really valid because it doesn't show operation under duress and then see how that perturbs the answers. I think you may be somewhat surprised.

MR. SIESS: That is a sensitivity study. Did you do one.

MR. RADEMACHER: No, we haven't addressed that as part of our PRA.

MR. EBERSOLE: I think it might be a worthwhile almost innovation to look at in these various studies, to look at valve performance with a range of reliabilities. In other words, I think valves are sensitive points.

MR. SIESS: Well, there is a range of reliability. You do have uncertainties factored in, don't you?

MR. EBERSOLE: But they are generic overall, aren't they?

MR. RADEMACHER: That is correct.

MR. SIESS: Well, you don't know whether the uncertainties are broad enough to include your cases ---

MR. EBERSOLE: I don't want to look at the homogenized version. I want to look at the valve fraction.

MR. RADEMACHER: Okay. I guess we haven't looked at that in every aspect if you look at uncertainties. As Dr. Siess, you have a feel that, you know, individuals have indicated that ---

MR. EBERSOLE: I am not talking about piling up conservatisms.

MR. RADEMACHER: But that is basically what I think the intent is, the understanding that there are certain limitations to PRA's and uncertainties, and they also are due to such things as statistical modeling, missions, computational errors or whatever.

Although if you look at any specific components, someone could always argue with it, especially if you didn't have plant specific data.

MR. SIESS: But you don't use a single reliability value for valves?

MR. EBERSOLE: Oh, yes, he does, don't you?

MR. RADEMACHER: Yes. That generic valve data ---

MR. SIESS: There is no uncertainty put into that?

MR. RADEMACHER: Not at that level.

MR. EBERSOLE: No, they homogenize it. So it gets lost.

MR. SIESS: At what level do you put the uncertainty in?

MR. RADEMACHER: Well, basically when you come up with the end result you put a error band on the overall conclusions.

MR. SIESS: Well, that is not really propagating uncertainties then. That is just making a guess at the end as to -- well, that is not the way it is done in most PRA's.

MR. RADEMACHER: Well, we try and address that as some of the limitations in the study.

MR. SIESS: Yes, but PRA's have been made where there has been an attempt to propagate uncertainties through the process to the end.

MR. EBERSOLE: By the way, many valves will operate under normal circumstances. So what I have said about them is not true of all of them, but then there are a few that it is true about them, and I just am curious about to what extent they drive the answer.

MR. SIESS: Are you saying that you did not propagate uncertainties through the PRA and you just used the median value at each stage of the game or some point estimate probability in each stage and then guessed at a range when you got through?

MR. RADEMACHER: Yes. I believe that is exactly how we approached it.

MR. SIESS: That is not the way many of them have been done. That may be part of the mini.

(Laughter.)

MR. RADEMACHER: We did do generic component failure rate data, one year of hourly meteorological data and plant specific midlife population data, topographical data, socioeconomic data and plant specific emergency planning information.

(Slide.)

Basically the team that we had involved was from nuclear engineering. We did have participation by operations, project engineering, environmental and our radiation management/radio chemistry team.

MR. SIESS: Did you do all of this in-house?

MR. RADEMACHER: No. It was performed by Stone and Webster and we reviewed ---

MR. SIESS: Performed by who?

MR. RADEMACHER: Stone and Webster.

MR. SIESS: And these were just your people on the team? Oh, this was your review team.

MR. RADEMACHER: Yes, that was a review team by Niagara Mohawk.

MR. SIESS: And you reviewed at what levels, fault trees and event trees?

MR. RADEMACHER: Yes. The whole package was reviewed by ---

MR. SIESS: At each step.

MR. RADEMACHER: Yes.

Basically the intent of this was to fulfill the NRC requirement for a Class 9 accident and provide an estimate of plant risk. It does provide a check on emergency effectivness and evacuation and risk mitigating features, and it did provide an indication of dominant risk contributors.

MR. SIESS: What were the dominant risk contributors of Nine Mile Unit 2?

(Slide.)

MR. RADEMACHER: 106 of the 292 dominant cut sets contained terms with related to the service water hardware failures. 149 of the 292 dominant cut sets related to RHR system hardware. 189 of 292 contained terms relating to failure to recqver offsite power. 113 of the 292 dominant cut sets contained terms relating to failure of one or more of the plants emergency diesels. And 121 of 292 dominant cut sets contained contained terms relating to failure of RCIC mechanical hardware.

MR. SIESS: Now in terms of probability of core melt, did any sequence contribute more than 10 percent say?

MR. RADEMACHER: Yes. There was one sequence which related to -- it was a substantial sequence --

MR. SIESS: What did your overall probability of core melt come out?

MR. RADEMACHER: 2.4 times 10 to the minus 5th I believe.

MR. SIESS: 2.4 times 10 to the minus 5th?

MR. RADEMACHER: Right.

MR. SIESS: Now if the staff reviewed that it would end up somewhere between 2.4 times 10 to the minus 5 and 2.4 times 10 to the minus 4.

MR. RADEMACHER: I am sorry?

MR. SIESS: Based on about 20 PRA's that have been made by utilities, after the staff finishes reviewing them, the probability of core melt goes up by either one or two orders of magnitude, and this is just a calibration that I have made.

(Laughter.)

MR. RADEMACHER: Yes, I think you are absolutely right. In the environmental report the NRC ran a run on Nine Mile 2 and we basically fell in the middle of all plants.

MR. SIESS: Okay. You are about average. But what was the dominant sequence and how much of it?

MR. RADEMACHER: I will get that. The dominant sequence was 1.1 times 10 to the minus 5th, which was sequence T-23QW, transient followed by loss of long-term suppression pool coolant.

MR. SIESS: That is one Mr. Ebersole can wipe out for you.

(Laughter.)

MR. SIESS: What was the risk from that?

MR. RADEMACHER: 1.1 times 10 to the minus 5th.
MR. SIESS: And your total was 2.5?
MR. RADEMACHER: 2.4 times 10 to the ---
MR. SIESS: So that is about half of it.
MR. RADEMACHER: Yes.
MR. SIESS: So that reduced you by a factor of about 2 in this case is all.
MR. EBERSOLE: Well, that is the classic answer for a boiler, isn't it, with a suppression pool?
MR. RADEMACHER: Yes.
MR. SIESS: But the thing is I think on GESSAR they thought that the UPPS would reduce the core melt risk by a factor of about four or five, and that might be true here, but you have got other sequences.
MR. EBERSOLE: Well, that was with the narrow scope UPPS oriented just for power failure. You know, you can orient it virtuzlly toward any target or function you want.
MR. SIESS: But there are some sequences it just doesn't affect.
MR. EBERSOLE: It depends on how you design it.
MR. SIESS: Okay. Go ahead.
(Slide.)
MR. RADEMACHER: I have some more slides after this as a summary. Further PRA's, right now we are

performing one for Unit 1 and after we have some operating experience, we are considering the possibility of upgrading our current, what I call, a mini PRA to more of a full one.

We kind of wanted to get some experience on Nine Mile 2 and then go to Nine Mile 1 and perform that study and then go back to Unit 2 after we get operational.

We have set up some training courses basically as part of mitigating core damage. Our training people are considering looking at dominant cut sets and that kind of thing to incorporate into the operator training. Right now they are going through enginering training on Unit 1 to bring some more people up to speed relative to PRA's.

MR. SIESS: Did your PRA suggest any placees where a relatively modest physical change would be significant, some weakness or outlier?

MR. RADEMACHER: Well, if you go back through the steps, I believe one of the dominant concerns was RCIC, maintenace of the valves and leaving them in the wrong position. That was one of the dominant conditions of that.

MR. SIESS: Well, that you couldn't fix with a hardware change, but you could do a lot procedurally.

MR. RADEMACHER: Yes, with a maintenance, and some of that was due to the fact that we now currently use double verification and that kind of thing. So if you assume that they left it there, it would have to be incorporated directly into the procedure to update that.

MR. SIESS: Well, in figuring the probability of the valves being in the wrong position, did you assume in the human factors part that it was double verification?

MR. RADEMACHER: No, we did not.

MR. SIESS: It was a single operator error.

MR. RADEMACHER: Yes.

MR. EBERSOLE: We mentioned earlier that the RCIC might be somewhat dependent on electrical fan cooling, but you were going to look at that. Was that put in the PRA, environmental control?

MR. RADEMACHER: Oh, no. That was part of station blackout. I think what we said was, and maybe I misunderstood your question, but as part of the station blackout study, we will be looking at the room coolers to verify that RCIC won't be limited to operation because of that.

Now one of the things we can do is open up the door and have operator action or something like that.

MR. EBERSOLE: I have here some place, and I don't need to dig it up, a piece of paper that says that RCIC will operate without its gland seal function; is that right?

MR. RADEMACHER: I would have to defer that to General Electric. I don't believe that it can.

MR. EBERSOLE: I can't either, but there is a printed report here to the extent that you don't need it, and I don't understand that.

MR. PIKE: I have a little familiarity with that from the EQ programs. I believe that GE takes the position that it will operate without the gland, but the consequences are you are going to get some steam in the RCIC pump room.

MR. EBERSOLE: You want brake sunction because of air ingress. It is on the high pressure side; is that right?

MR. PIKE: I am not sure.

MR. EBERSOLE: I just didn't understand how it would work. That is all.

MR. RADEMACHER: George Moyer I believe has a response to that.

MR. MOYER: On our RCIC, the air pump just provides air to prevent an outflow of steam. It would operate and we would just get some steam into the room.

MR. EBERSOLE: It would just leak, okay. Thank you.

(Slide.)

This slide basically summarizes what our results for our PRA were and compares this against other contemporary plants.

MR. SIESS: Are these the original values for the other plants before the NRC review?

MR. RADEMACHER: Are these the original values for the other plants before the NRC review?

MR. RADEMACHER: Yes. These are the industry submittals and not the NRC review.

MR. SIESS: Okay.

MR. RADEMACHER: That basically summarizes my presentation on PRA.

You did have a question relative to what happens on RCIC, a line break which is unisolable. RCIC is in a cubicle room by itself and what happens is the RCIC cubicle blowup panels would go which would allow the steam to go up the RCIC pipe chase and deliver this steam in the reactor building.

MR. EBERSOLE: Was that an intentional delivery path so it wouldn't hurt anything?

MR. RADEMACHER: I believe that is the case of how we looked at it, yes.

MR. EBERSOLE: How about reactor water cleanup?

MR. RADEMACHER: I believe that also has blowup panels, but I don't know exactly what the ---

MR. EBERSOLE: Why don't you have a look as to what the escape route was for those. I only want to find out if you have the Limerick logic, you know, which was to let it go some place without hurting equipment.

MR. RADEMACHER: Well, you have to understand that I think that was part of our concept, but it might raise up through the building. However, major pumps and components are located basically in other cubicles. So they would be afforded protection.

MR. EBERSOLE: Let me ask this. The Division 3 diesel, does it have ventilation cooling equipment which is driven by its own output?

MR. RADEMACHER: Yes.

MR. EBERSOLE: It is only the service water you didn't drive?

MR. RADEMACHER: That is correct.

MR. EBERSOLE: Thank you.

MR. ZALLNICK: Mr. Rademacher also has responses for questions that were asked this morning, if you would like to take the time now to cover those.

MR. SIESS: I think we will take the time now.

MR. RADEMACHER: The first question was relative to the Tip system and whether it was the same as other GE BWR's, and the information from GE indicates that it is.

Then we had a response relative to the CO2 tank failure. The CO2 tanks are located in the access area which is non-safety related area, and if that tank were to explode, it would not have an effect on safety related components. Now obviously it does have relief devices.

MR. EBERSOLE: Well, let me ask you this. That was a large tank. So I could guess that you used a fraction of its contents, which is selectively parceled to certain areas.

MR. RADEMACHER: Yes.

MR. EBERSOLE: And I am going to talk about Watts Bar now which blew up some rooms because they didn't have a safety grade cut-off device. They just blowing it into the room until something gave. How do you preclude that?

MR. RADEMACHER: We don't have safety grade components on that just like Watts Bar. We would basically either pop a fire projection penetration, plus we would have leakage out the door.

MR. EBERSOLE: Then that wouuld invalidate the CO2 performance if it was trying to cope with a fire, because that would allow ---

MR. RADEMACHER: No. It would seal up to a certain pressure.

MR. EBERSOLE: But then it would blow, and that would destroy the hermetic seal of the room and air would come back and the fire would ---

MR. RADEMACHER: No. I would say that we have something like 13 tons of CO2 in those tanks, and the inlet flow is about 450 pounds per minute.

MR. EBERSOLE: Well, are any safety functions predicated on the notion that these cut-off valves will in fact always stop at their desired number of cubic feet, and if you exceed that you get in trouble one way or another?

MR. RADEMACHER: That is correct. We were relying on those valves to stop the flow of CO2. However, when this would happen, if a fire were to occur, there are certain emergency procedures that require the fire personnel to respond directly to the fire, and they also have been trained, if it did not turn off, to manually shut off the valve outside the room.

MR. EBERSOLE: I was about to say they would walk into a wall of CO2 and promptly suffocate.

MR. RADEMACHER: I believe there are local shutoff valves as well as near the tank.

MR. EBERSOLE: Isn't it worthwhile to upgrade the parceling valves to some redundant configuration or else to not even use common tankage?

MR. RADEMACHER: I have to maybe clarify one point. Each zone that is provided CO2 and, for example, there are three zones in the control building and one in the reactor building. They are switch gear rooms. That is where we use CO2 in safety related buildings and each one has its own separate pipe that runs to that. So it would not inject into all of them at the same time.

MR. EBERSOLE: I know, but it has a common tank, doesn't it?

MR. RADEMACHER: Yes.

MR. EBERSOLE: So back there at the tank there is a time valve which is going to lock up wide open and will never shut off.

MR. RADEMACHER: I am sorry. Maybe I misunderstood your question.

MR. EBERSOLE: Where you have a common storage volume for a number of enclosures and you intend to parcel a fraction of the discharge as the uppermost limit of cubic footage that you are going to put in a certain place, that cut-off function is executed by some kind of a valve which probably is nowhere near safety grade, and I don't know what the reliability is, but there is a distinct chance that it will stay open and you will discharge the entire tankage into that one space with unknown consequences.

MR. RADEMACHER: As I indicated before, the fire brigade would be dispatched to that team and ---

MR. EBERSOLE: Suppose they meet an unexpected wall of CO_2 and they all drown.

MR. RADEMACHER: Then they can go back to the tank itself and turn it off.

MR. EBERSOLE: They will be dead.

(Laughter.)

MR. RADEMACHER: I am sorry. I didn't hear that.

MR. EBERSOLE: I don't know how you do this.

MR. SIESS: Would the fire brigade open a door to a room that had CO2 in it?

MR. RADEMACHER: Let me introduce Mike Kammeron from fire protection.

MR. KAMMERON: My name is Mike Kammeron and I am a fire protection engineer. We have massive selector valves and selector valves. If you respond to a fire in a plant, the fire department is obviously going to let the suppression system do its work first off.

MR. EBERSOLE: What if it overdoes its work?

MR. KAMMERON: Excuse me?

MR. EBERSOLE: What if it overdoes its work by discharging the whole tank?

MR. KAMMERON: Well, first of all, the CO2 system is going to discharge into the room and immediately you are going to have fire alarms and pressure switches which activate the control room and alert the fire brigade.

Upon the fire brigade arriving at the scene, if they find that the CO system is still discharging beyond its design limitations, they can manually shut off the CO2 at the location.

MR. EBERSOLE: How will they know that is occurring? They can't see it, can they?

MR. KAMMERON: Well, you can see CO2 all over the place.

MR. EBERSOLE: They just see the fog that is developed by the cold gas?

MR. KAMMERON: Yes.

MR. EBERSOLE: Isn't there a considerable amount of that anyway that comes through the room leakage?

MR. KAMMERON: Oh, yes. There is some leakage out of the room. You have to design to some extent to some pressure leakage so you don't overpressurize the room.

MR. EBERSOLE: What I am really trying to do is understand how they avoid going into a saturated area which is 10 times bigger than they thought it was going to be.

MR. KAMMERON: Well, first of all, the fire department is also going to respond with some Scott air packs.

MR. EBERSOLE: Okay. Is that always the case?

MR. KAMMERON: They are not going to walk into an unknown atmosphere.

MR. EBERSOLE: Yes, but will they walk to an area that they thought was not contaminated with CO2, but in fact it was because the panels had blown out?

MR. KAMMERON: Well, if the fire department didn't, let's say, have their Scott air packs on and they walked into a room, the CO2 systems are supplied with

Wintergreen capsules. So the discharge of CO2 crack these capsules and you will get an oderant in the air.

MR. EBERSOLE: Okay. I've got you. I didn't know it had an odor tracer.

MR. KAMMERON: This is all taught to the fire brigade in general fire fighting training.

MR. EBERSOLE: Is that standard that all CO2 has an odor tracer like natural gas?

MR. KAMMERON: Well, we use it at Nine Mile Point, Nine Mile 1 and 2 for CO2 hazards and also for the halon.

MR. EBERSOLE: You can smell it before it will asphyxiate you?

MR. KAMMERON: Oh, sure.

MR. EBERSOLE: Great. I didn't know that.

MR. SIESS: Is the odor added to the CO2, or did you mention some capsule?

MR. KAMMERON: Well, what happens is you have a key off the discharge line, the CO2. And as soon as you get your initial shot of CO2 in that line you are going to break a glass capsule which contains a concentrated wintergreen odor.

MR. SIESS: So it is added to the CO2 as it discharges.

MR. KAMMERON: As it discharges, right.

MR. SIESS: Have you got some more?

MR. RADEMACHER: Yes, two more.

The next one was relative to Mr. Ebersole's question regarding the pressure on an ATWS, and we will have Mile Colomb respond to that question.

MR. COLOMB: I would like to answer your question on pressure control during an ATWS. Our initial instructions to the operator on ATWS is to reduce pressubr only as far as 940 pounds. This is to stop the reactivity addition associated with a cool-down. An analysis has shown that this, along with lowering of the level, as also described in the EOP's will produce sufficient decrease in recirculation flow and voiding in the core to keep power down to where ATWS can be sustained.

MR. EBERSOLE: And then if for one reason or another he doesn't get standby liquid control, do you go to a lower pressure?

MR. COLOMB: Yes. Lowering the pressure would then be driven by the inability to maintain level or containment parameters, yes. That would cause him to further lower the pressure.

MR. EBERSOLE: Down to how far?

MR. COLOMB: As far as possible.

MR. EBERSOLE: But stil maintain the level?

MR. COLOMB: If possible, yes.

MR. EBERSOLE: Which would be a very ---

MR. COLOMB: Yes. You are always instructed to maintain level at top of active fuel if it can be done.

MR. EBERSOLE: That would mean a core coolant structure which would be largely froth, wouldn't it?

MR. COLOMB: Yes, that is true.

MR. EBERSOLE: Would that louse up the level control signals very much since it only reads equivalent solid heighth? Would that automatically synthesize a new reading on the level instruments?k

MR. COLOMB: The froth would be inside the core area. Our level indication is external to the downcomer.

MR. EBERSOLE: So the froth might be quite high?

MR. COLOMB: Yes.

MR. EBERSOLE: Thank you.

MR. RADEMACHER: The last question was a clarification on the clarification.

(Laughter.)

On the MSIV and the weight of the water I indicated that it was initiated by level, and the correct initiation could be caused by the loss of condenser vacuum. So I just wanted to make that clarification.

Those are all my clarifications.

MR. SIESS: I propose that we take a break now, and we will reconvene at 2:45.

(Recess taken.)

MR. SIESS: Let's go now to Item 23 on the agency, the intergranular stress corrosion cracking.

MR. ZALLNICK: Item No. 23, IGSCC, is going to be presented by Mr. Donald Pracht. Mr. Pracht has 21 years of nuclear experience in the design of Nine Mile Point, Units 1 and 2. He is the lead mechanical engineer.

(Slide.)

MR. PRACHT: Good afternoon, gentlemen.

My name is Donald Pracht. I am the lead mechanical engineer, and I have been with the project since its inception in '71.

(Slide.)

This afternoon I would like to provide you with Niagara Mohawk's position on IGSCC or intergranular stress corrosion cracking with specific reference to Nine Point Point, Unit 2.

By virtue of operating a BWR since 1969, we feel Niagara Mohawk has gained an overall understanding of the problem of IGSCC. This knowledge and concern has been factored into the design of Nine Mile Point, Unit 2.

Specifically out approach to IGSCC can be summarized in three broad categories.

First, attention was given to the parameters as outlined in NUREG 0313, Rev. 1.

Second, where circumstances permitted and prudency dictated, material and piping systems wee upgraded.

Third, in those cases where additional techniques could be utilized in the future to enhance the system's resistance to IGSCC, due consideration was given as to when and if they should be employed at this point in time.

Now let us explore how this philosophy was implemented on the project.

(Slide.)

Plant systems overall were reviewed for comformance to NUREG 0313, Rev. 1. Materials were chosen a basis of normal system operating conditions. In those systems where the normal temperature exceeded 200 degrees Fahrenheit for greater than one percent of the life of the unit, either an L or an NG material was chosen.

For austenitic stainless steel, including those systems employing stainless steel below the threshhold conditions, either one of two options was invoked, solution annealing heat treat followed by a water quench was utilized or, if not water quenched, the material was required to meet the ASTM A262 Spec Practice A.

The control of site procedures and practices has been a concern throughout the life of the project.

Specifically all welding, other than that done on the recirc piping, which was independent in terms of control, has had strict controls on interpass temperature.

Site fabrication techniques for small bore stainless steel piping have controlled the amount of cold work that could be performed on the material by limiting the diameter, thus limiting the strength and the resulting residual stress levels.

Cleanness, especially with respect to providing a halogen and sulfur free environment, has always been a project concern. Compliance with Reg. Guide 137, or ANSI 452.1 has been accomplished in numerous ways. Such things as shop and site control of expendable products has been rigorously enforced in terms of prohibition of calcium chloride for dust control in the summer months or snow melting in the winter, the exclusion of Teflon in CAT 1 systems tend to exemplify our methods of compliance with the Reg. Guide.

As an additional measure, stainless steel systems are either kept in a dry state or laid up wet with deaerated demineralized water.

(Slide.)

Now probably the most important to you. Nine Mile Point 2 has undergone several significant upgrades during the design and construction phases. These upgrades were brought about due to problems identified in operating reactors.

Early in '79 General Electric presented Niagara Mohawk with the complete history of the recirc. inlet safe end problems at Duane Arnold that had developed during '78.

The Duane Arnold design consisted of a welded inconel thermosleeve which had an inherent crevice due to the welded design concept.

Additionally, the stresses which were present during operation further aggravated the problem. The later designs of Brunswick, LaSalle 2 and Nine Mile 2 all attempted to mitigate the problem by reducing the crevice length along with reduced stress levels.

However, it is not possible to design a fabricated section without any potential of a crevice being present. Realizing this, General Electric designers turned to a forging utilizing the tuning fork design.

(Slide.)

And for clarification, the top portion of the slide is the Duane Arnold design, and I think you can easily see where that crevice is to the right of the area indicated as a field weld, which would be the weld made from the nozzle to the vessel internals in terms of getting the water to the jet pump risers.

The forging at the bottom very neatly eliminates the crevice problem that is inherent in a welded design. Nine Mile 2 was effectively the same basic design as what Duane Arnold is represented as.

During the meeting when GE was presenting all this to us, I was a participant and GE offered to supply Niagara Mohawk with the new forged 316 L safe ends with the integral thermal sleeve of the tuning fork design.

Now this was quite a blow in a sense since the vessel had already been totally fabricated. This meant that Niagara Mohawk would have to incur the cost of having the existing safe ends removed and the new ones installed.

Considerable discussion took place within Niagara Mohawk regarding this decision, and in October of '79, and I think it was alluded to by Mr. Mangan yesterday, General Electric was advised that the safe ends were to be replaced.

I might add that this was done on the basis of prudency on Niagara Mohawk's part when looking at the long term and was well before the Nine Mile 1 problem ever surfaced regarding safe ends.

Actually as a part of the safe end replacement program General Electric also advised that our recirc. piping, which was then to go into fabrication with 304 stainless steel, could be supplied on an upgraded basis with 316 NG material or a nuclear grade.

Consider data existed supporting the superiority of the 316 NG versus attack by IGSCC. As was indicated by Mr. Mangan yesterday, Niagara Mohawk has always taken, they feel anyway as they have always taken a responsible position with respect to improvements in plant design.

This decision was made on the basis of a definitive plan improvement and GE again was advised to supply our recirc. piping as 316 NG.

(Slide.)

Another upgrade,,e which I think was also mentioned by Mr. Terry earlier, was to change the rod drive insert and withdraw piping from 304 to 304L. This was based on the latest GE design evaluation of the system operating temperatures during various modes of operation.

(Slide.)

Some additional considerations merit discussion with reference to an integrated approach to IGSCC.

Undoubtedly, one of the most important operational aspects is with reference to reactor water period. It is definitely Niagara Mohawk's intent to meet the EPRI guidelines of less than .3 micromoes per centimeter regarding conductivity.

Niagara Mohawk has been active in the water chemistry programs with General Electric and others since Nine Mine 1 went on line in '69 and fully recognizes the merit of good plant water chemistry.

(Slide.)

Much attention as of late has been focused on the potential means of further mitigating IGSCC. These are hydrogen water chemistry and induction heating stress improvement, commonly known as HWC and IHSI, both of which are being seriously considered by the nuclear industry at this time.

However, for Unit 2 Niagara Mohawk has taken the position that we will not commit to any further upgrades than have already been implemented at this time.

It should be remembered that Nine Mile 2 has the 316 NG with the 316 L forge to recirc. nozzles, which definitely are the primary means of deterring IGSCC.

(Slide.)

In conclusion, we believe it can be demonstrated that Niagara Mohawk has had considerable experience with IGSCC, and this is based on the various points that are tabulated here, involvement with the BWR owners group since '78, participation with the NRC, EPRI and GE on IGSCC issues, recirc. piping replacement program at Unit 1 with the prime responsibility being borne by Niagara Mohawk's own engineering department. That was a first. The engineering staff obviously is well aware of the emerging issues on IGSCC and, lastly, monitoring of the various mitigating actions presently being discussed in the technical forum, such as hydrogen water chemistry and induction heating stress improvement.

The one that I didn't refer to up there is that we are also well involved with the NDE facility down in Charlotte.

MR. SIESS: Any questions, Jesse?

MR. EBERSOLE: No.

MR. SIESS: Thank you. I don't think there are any questions now.

MR. PRACHT: Thank you.

MR. SIESS: The next item is the environmental qualification program.

MR. ZALLNICK: The presentation on the environmental qualification of equipment will be given by Mr. Doug Pike, who was previously introduced.

MR. SIESS: Mr. Pike, it seems to me we might be able to save some time here. I have looked through the slides and they seem to refer to your commitments to the NRC requirements on various categories, and I think we might reasonably assume that the staff is going to see that you meet those commitments and that your equipment is environmentally qualified. And if that is true, I wonder if you could simply tell us whether there is anything unique about your program or anything that you are doing that somebody else is not doing or any problems you have with environmental qualification and, if there is nothing special, I don't know why we need to spend much time on it.

MR. PIKE: I don't think there is anything particularly unique, other than perhaps our mechanical environmental equipment program where we take a look at the safety related mechanical equipment and identify organic materials in that equipment and then determine their useful life from a thermal and radiation aging standpoint and document their qualified life from that standpoint.

That is a fairly new issue that we have taken this action on.

MR. SIESS: It is a new issue that has been raised by the staff or a new approach?

MR. PIKE: I think the environmental qualification mechanical equipment, there has been really no real clear direction from the NRC on exactly what needs to be done there. In fact, our program is based on some direction given in response to I believe it was a question during FSAR review.

MR. SIESS: Now one issue of some current interest in the research area, for example, is aging and also in the regulatory area because plants are getting older.

You have got a plant that is 15 years old. Have you had good experiences with aging and how that might relate to -- was the Nine Mile Point 1 equipment qualified the same way your Nine Mile Point 2 is?

MR. PIKE: It wasn't qualified to the same standards that Unit 2 was being qualified to. However, they are going through a program to upgrade the qualification of the equipment at Unit 1 at the present time.

MR. SIESS: Does your experience with Nine Mile 1 tell you anything about the significance of equipment qualification and what it tells you about how good it is going to do or is it strictly accidents that we are worried about that we will never find out until we have one?

MR. PIKE: To my knowledge, I am not aware of any specific problems with the equipment at Unit 2 due to age related effects, and of course they have not been exposed to accidents over there.

MR. SIESS: Have you had failures that you would attribute to age?

MR. PIKE: I guess for specifics I would have to have someone from Unit 1 address that.

MR. SIESS: Has anybody ever asked that question before?

MR. ZALLNICK: Mr. Perkins is going to try to respond to that question.

MR. PERKINS: I am Tom Perkins, General

Superintendent of the Nine Mile site. To the best of my knowledge we have found no instances of failure due to aging. I believe that most of the effort is in the accident area and not in normal operation.

MR. SIESS: Anything else you would like to say about the program.

MR. PIKE: I might just give you a brief status of where we are at the present time.

(Slide.)

In one Class 1E electrical harsh environment program and our seismic qualification of mechanical components requiring motion to fulfill their safety function we have approximately 4100 components in that program. At the present time 78 percent of those are qualified, and by the end of March we expect that to be about 89 percent.

In our mechanical environmental equipment program we have some 3850 components and at present 65 percent are qualified. We expect that to be about 86 percent by the end of this month.

MR. SIESS: When do you plan to start operation?

MR. PIKE: Fuel load is February of 1986.

MR. SIESS: And everything will be qualified before then I assume.

MR. PIKE: We expect to have everything qualified before then.

MR. SIESS: I think that is about all I need to hear.

Jesse, do you have any questions you would like to address to him while he is here?

MR. EBERSOLE: I would to, yes. In the table of equipment qualifications it defines that the parameters of interest are temperature, pressure, humidity and irradiation, and those are of course the the parameters of interest, but I think you would find it profitable to read a completely candid Sandia report that shows how certain qualification requirements have been met by systematic attention to the sequence of operations during testing.

For instance, if you discover that you have post humidity condensation on a component which is normally cold, you can preheat it before you test it and you will never have surface condensation and it will pass. But if it is initially cold and goes through a heating or condensation transient, it will leak. I am talking about terminal blocks.

That report I think is very illuminating and I suggest you ask the staff to give it to you and see what these laboratories have been doing to certify as qualified certain pieces of equipment. There is some pretty shady business about qualification process. Have you see this?

MR. PIKE: No, I have not personally seen this.

MR. EBERSOLE: Well, there has been deliberate, for instance, preheating to avoid the surface condensation.

MR. SIESS: I think you might ask the staff if they have seen it.

MR. EBERSOLE: Oh, I am sure they have seen it. There was a big controversy about whether or not they are going to impose some trouble to Sandia for being so candid.

(Laughter.)

Do you know about this?

MR. SCHWENCER: No, not personally, but I suspect that our equipment qualification people in the incident in shop are well aware.

MR. EBERSOLE: Right. I wish you would impart some of that to the applicant.

MR. SCHWENCER: We can get the reference to it and provide it to Niagara Mohawk.

MR. EBERSOLE: I just want this applicant to know about what has been used to qualify equipment, which has resulted in non-qualified equipment.

MR. SIESS: You are being warned that the rules may be changed.

MR. EBERSOLE: In fact, there have been synthesized qualification processes that resulted in non-qualified equipment.

MR. PIKE: Again, we qualified to the requirements of IEEE 323 1974 as far as testing sequences. We review and approve any vendor qualification testing plans and reports. I am sure if we were aware of these things we certainly would not allow it.

MR. EBERSOLE: Why don't you check and see and I think you will not find this sequential process of imparting a human atmosphere to an initially cold component.

MR. SIESS: Put up your first slide.

(Slide.)

There are three NRC documents and two industry standards that I believe have been approved by Reg. Guide. Will they give you any guidance as to the sequence of qualification?

MR. PIKE: Obviously the testing sequence is you environmentally and seismically age the equipment before you expose it to the accident. That is the general sequence.

MR. EBERSOLE: These are broad sequential things.

MR. SIESS: Well, I would be surprised if they are not pretty darned detailed.

MR. EBERSOLE: I don't think there is a trace of this condensation phenomena in any of those ---

MR. PIKE: Well, I am not aware of it to that level of detail.

MR. SIESS: We have got five documents that probably cover 100 things we haven't thought of.

MR. EBERSOLE: Let me put it this way. There is almost a staff controversy, Chet, about whether you can tolerate leakage currents on terminal blocks from this phenomena and, as far as I know, that is up in the air at this point in time.

MR. PIKE: Again, as far as our plant specific design in that area, again as I mentioned earlier, if we have an area that is subjected to that kind of condensation, we wouldn't use terminal blocks but we would use qualified splices.

MR. EBERSOLE: You are right and you wouldn't b in any trouble, but there are many plans that have used terminal blocks.

The other thing is the environment that you specify may be in fact controlled by an equipment which has environmental qualification in its own right. I mentioned earlier about the valves which may not close which changes rather drastically what is the nominal mild environment. So that has a regressive effect which has to be considered in the light of making the valves work or living of the consequences of it not work, and I don't think that is in that set of up at all either because it is rigidly tied to the hypothesis that redundancy always works.

So other than this, I don't have anything else.

MR. SIESS: Thank you then, Mr. Pike.

The next item on the agenda then is the radiation protection program. I assume that is occupational protection, right?

MR. ZALLNICK: The presenter for the radiation protection program is Mr. Volza who was previously introduced. He is the Supervisor of Radiological Support.

(Slide.)

MR. VOLZA: Good afternoon again, gentlemen.

I would like to briefly give you an overview of the radiation protection program currently in place to support Nine Mile 2.

MR. SIESS: Excuse me. As a rehersal for the full committee, let's skip the organization charts.

MR. VOLZA: Okay, will do.

(Slide.)

The Nine Mine Point radiation protection program is designed to provide for the protection of all permanent and temporary personnel and all visitors from irradiation and radioactive materials in a manner consistent with Federal and State regulations through all phases of plant operation.

In order to achieve this program objective, Niagara Mohawk drafted and implemented a company policy to address these goals.

On the left here is our Nine Mile Point ALARA commitment which describes our commitment to maintaining radiation exposures ALARA.

In addition to this statement, it should be noted that Niagara Mohawk has formerly endorsed INPO's five rem per year exposure guideline for all utility and contract personnel. This is further evidence of the Nine Mile Point commitment to keep exposures ALARA.

(Slide.)

The responsibility for implementing the radiation protection program and the ALARA commitment rests with the Superintendent of Chemistry and Radiation Management, Mr. Edward Leach and members of his staff. Mr. Leach reports directly to the General Superintendent of Nuclear Generation, Mr. Tom Perkins who is responsible for all aspects of site operations, including the onsite radiation protection program.

The site departmental superintendents and supervisors are also responsible to the General Superintendent for the radiation protection program within the site and support the radiation protection organization in forming and implementing the site program for maintaining exposures ALARA.

(Slide.)

The Niagara Mohawk management is committed through its radiation protection program to maintain occupational exposures as low as reasonably achievable. In order to achieve compliance with this commitment, management oversight of the radiation protection program has been provided through four separate groups, the site radiation protection staff, the site operations review committee, safety review and audit board and the quality assurance department.

You have already heard about the latter three groups. I would like to briefly describe the RP staff and its responsibility.

(Slide.)

Is the responsibility of the Nine Mile Point Radiation Protection organizations to conduct surveillance programs and investigations to ensure occupational exposures are as far below specified limits as is reasonably achievable.

The responsible is shared between respective site and corporate health physics groups and, as you wish, we will skip the actual organizations. Let it be sufficient to say that the site organization reports through the Vice President for Generation and is headed by Mr. Leach, and the corporate organization reports through the Vice

President of Nuclear Engineering and Licensing and headed by our lead corporate health physicist.

And if you would like to see it later, I could provide you with the personal experience and qualifications of all of the individuals on staff.

MR. SIESS: I don't think that will be necessary. The slide on the right, what is the significance of formally endorsing INPO's five rem per year guideline? At Nine Mile Unit 1 have you had exposures exceeding five rem per year?

MR. VOLZA: We have had one since 1979, I believe. It has not been a typical occurrence. Recently at the tail of last year then INPO Director Mr. Wilkinson sent a letter to all utilities asking them to voluntarily comply to a five rem per year exposure guideline to show the industry's commitment to reducing radiation exposure.

MR. SIESS: Now that five rem per year applies to contract personnel?

MR. VOLZA: As well. It will apply to both utility and contractors arriving at the Nine Mile Point site.

MR. SIESS: Thank you.

(Slide.)

MR. VOLZA: I would like to briefly discuss the Nine Mile Point ALARA program as it would relate to both the design and operation of Nine Mile Point Unit 2.

The general design considerations and methods employed at Nine Mile to maintain in plant radiation exposures ALARA have two objectives, and those objectives are listed on the overhead on the right.

Basically they are to minimize the amount of time plant personnel spend in radiation areas, and also to minimize radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require the attention of plant personnel.

With this in mind, equipment layout, shielding, penetrations and piping locations have been reviewed by engineering personnel, including radiation protection personnel during the development of the design drawings for implementation of the ALARA philosophy and minimizing radiation occupational exposures.

(Slide.)

Examples of design consideration are shown here on the left and includes such things as ensuring that interface control panels for systems are placed in low background areas to the use of reach rods in our reactor water cleanup system to minimize operator exposures. And as far as developing a Unit 2 model, which was extensively used by the ALARA review committee for ALARA design and used as a reference to evaluate access control, shielding design, pipe runs and equipment placement.

(Slide.)

As a result of our design efforts, we have estimated that approximately 517 man rem per year will be expended to operate Nine Mile 2. When compared to the industry standard, this is very favorable and below the industry average.

As you can see on your left, I have superimposed here the Nine Mile 2 estimate over the current Nine Mile 1 exposures since 1974 and the industry average for a typical BWR.

MR. SIESS: Now.looking at that chart, that shows a very significant decrease for Nine Mile 2 as compared to Nine Mile 1. Now since your administrative controls clearly can be the same for both units, I would have to assume that that decrease is attributable to the design features. Is that correct, or is it just some optimism involved in there?

MR. VOLZA: No, it is as a result of the inherent design features of Nine Mile 2. I would also like to add that again with the ALARA program taking a more prominent role in the day-to-day operation of the plant, we are also anticipating that the coming year the man-rem exposure for the Nine Mile 1 plant will be approximately 400 man-rem.

And basically we are taking this this lead and have taken Dr. Harold Denton's challenge to reduce radiation exposure by 20 percent over the next year.

MR. SIESS: Those three big peaks there on Nine Mile 1, what do they correspond to?

MR. VOLZA: The peaks correspond to outage years and the resulting peak between '81 and '82, you did not see a dramatic increase because of the extended recirculation outage that we had to undertake at that time.

MR. SIESS: That was when you replaced the recirc.. pipe?

MR. VOLZA: Correct.

MR. SIESS: Did you decontaminate the primary system before you replaced those pipes at Nine Mile?

MR. VOLZA: We didn't decontaminate the entire primary system but portions thereof.

MR. SIESS: And the peak in '77 was an outage?

MR. VOLZA: Yes.

MR. SIESS: Was it a major repair or what was it?

MR. VOLZA: A feedwater sparger.

MR. SIESS: Does that account for most of that because that is a thousand man-rem about your target of 800.

MR. LEACH: One of the large jobs in 1977 was the feedwater sparger. There also were associated with that outage some of the first I believe Browns Ferry fire modifications.

MR. SIESS: Now the flat line I see up there for Nine Mile 2 then I would assume assumes that there are not going to be any major repair outages for Nine Mile 2.

MR. VOLZA: The line you see there is an estimate of normal operating conditions, excluding outage situations. We would anticipate that there would be some deviation up and down, depending on the amount of work that had to be done during the outage period.

MR. SIESS: What would the average BWR curve look like if I did it on the same basis?

MR. VOLZA: The average BWR curve is the middle curve that you see there.

MR. SIESS: But that also includes all the outages and recirc. pipe repairs and replacements and so forth.

MR. VOLZA: Correct.

MR. SIESS: So it is not really comparable to your Nine Mile 2 curve. Neither one are.

MR. VOLZA: The only information that I had readily available to me at the time was a comprehensive total of all exposures.

MR. SIESS: I realize that, but they are just not comparable curves. If I took the rest of the industry and took out all the outages, major repairs and major modifications, that curve would have to come down, just like your Nine Mile Point 1 curve would come down.

MR. VOLZA: True. We just completed an outage situation this past year in 1984. The total man-rem expended for 1984 was approximately 860. And we had anticipated that our exposures for this coming year would have been well over 1100. So we were able to reduce exposures for this past year by 300 man-rem.

We anticipate that, yes, there is going to be some fluctuation of approximately two to three hundred man-rem possibly because of the outage, but because of our diligence in trying to reduce exposure, we would not anticipate it going any higher than that.

MR. SIESS: That is good.

MR. EBERSOLE: Can I ask a question?

MR. SIESS: Yes.

MR. EBERSOLE: I hear that, unfortunately, if you find you have got to use hydrogen injection to assist the stress corrosion cracking problem, the dose levels, short term at least, go up. I don't quite understand the mechanism that causes that or whether it is just a transient high level or is it residual higher activity or whatever. Are you hedging with additional shielding or anything in contemplating the possibility you will put hydrogen in the primary loop?

MR. VOLZA: We have discussed the use of hydrogen and we would anticipate that it would increase exposure levels because of the additional N-16 carryover. At the present time we haven't looked at it totally because it is still in a state of flux and until our reviews are completed at the engineering level, then we would look at it from an alara standpoint and from an operating standpoint.

MR. EBERSOLE: Well, if it just an N-16 increase in level, it would imply maybe nothing more than maybe some additional shielding on the turbine, right?

MR. VOLZA: Possibly.

MR. EBERSOLE: Is there any long-term activity increase beyond N-16?

MR. VOLZA: Not that I am aware of, unless, Ed, do you know?

MR. LEACH: I can address that a little bit. The problem in the increased exposure is almost exclusively associated with the N-16, least as far as I know at this point.

For the areas near steam lines where inspection may be necessary during operation, hydrogen injection can be cut off and the levels will drop down rather rapidly without any overall detriment to the hydrogen injection program.

So expect that, yes, there will be some increase in operational exposure for the plant personnel, but it does not look at first glance to be prohibitive. There is also a possibility that there could be some net gains certainly if you lower exposure due to maintenance and replacement.

MR. EBERSOLE: Yes. What factor of increase in N-16 dose level do you get?

MR. LEACH: There is some indication that it could be as much as three to five times. There is also some recent indication that says this is true when you first start the hydrogen program, but later on if you are able to reduce hydrogen level, or hydrogen injection rate, the increase then may drop accordingly and it may be only a factor of maybe two or three times.

MR. EBERSOLE: Well, would that cause you to alter your turbine shielding at all?

MR. LEACH: At this point we are not contemplating make any changes to the turbine shielding. We would want to operate first and take a look at just where the problems were before we proposed a fix to it.

MR. EBERSOLE: What is the dose on the roof of the turbine hall? Do you have top shielding or is it bare?

MR. LEACH: I would have to defer to Mike Stocknoff on those numbers.

MR. RADEMACHER: On the moisture separater reheater we have three foot of concrete. That is one of the major contributors to roof exposure or sky shine.

In addition, I believe on the turbine floor itself we have some steel, and Mike could probably describe those a little better than me.

MR. STOCKNOFF: Yes. On top of the turbine itself, the roof of the turbine building is just sheet metal. But on top of the cross-over pipe and the moisture separator reheaters we do have steel to minimize potential exposure.

MR. EBERSOLE: Is the roof off limits to ordinary people walking around on it?

MR. STOCKNOFF: Yes. There is nothing up there. Typically it is around 100 MR on the top of the roof, too. It is not that high.

MR. EBERSOLE: Thank you.

(Slide.)

What we have here on the left is a breakdown of that 517 man-rem per the different groups and how it would be broken down for the different work functions. Basically you can see in all respects again adding some factor there to account for outage situations, the typical man-rem exposures anticipated at Nine Mile 2 would be below the industry standards.

MR. SIESS: Why is your in-service inspection higher?

MR. VOLZA: Basically because of the plant and the sophistication of the equipment and the preventive maintenance, it does require a lot more in-service inspection to maintain some knowledge of what is going on in the respective systems.

MR. SIESS: Do you think you will be doing more in-service inspection than other BWR's and not that you will be getting higher exposure per inspection?

MR. VOLZA: I would anticipate we would, but I would defer that to Norm for any further clarification on that.

MR. TERRY: I think quickly the best way to answer that is, first off, as opposed to the average, I am sure we have more piping and more welds requiring in-service inspection.

Additionally, in terms of our programs, we are essentially implementing the latest and the more extensive ISI programs. So based upon that in comparison with the average, one would expect it to be higher.

MR. SIESS: Okay. That is with the average, which includes a lot of older smaller plants?

MR. TERRY: Absolutely, and there is a substantial amount.

MR. SIESS: Compared to plants of the same vintage and type you would be about the same I assume?

MR. TERRY: Oh, yes, I would expect so.

MR. SIESS: I see. Thank you.

(Slide.)

MR. VOLZA: In addition to the design consideration, there are also operational considerations that we have considered, and basically the objectives of our operational ALARA program is to maintain both individual and collective occupational exposures as low as reasonably achievable.

The essential ingredients of the Nine Mile Point Operational ALARA program consists of the five elements that you see her on your right, a dedicated ALARA group, pre-ALARA work reviews, post-ALARA work reviews, ALARA goals program and ALARA awareness program.

I would like to briefly touch upon each area.

The dedicated ALARA group currently consists of an ALARA coordinator, an ALARA specialist for each unit and appropriate contractor and support personnel reporting through me to Mr. Leach for controlling all of the ALARA functions on the site.

In addition to this group, this group is responsible for performing pre-ALARA work reviews to determine and minimize both individual and collective doses for a particular job, action levels have been set, as you can see here on your left to determine the depth of the ALARA review to be undertaken.

Once this has been accomplished, appropriate recommendations are provided to reduce the estimated exposure as low as reasonably achievable.

Subsequent to the completion of the work reviews, post-ALARA reviews are performed to evaluate the overall effectiveness of the station's ALARA job planning process, it provides a basis for evaluating the usefulness of our ALARA measures undertaken so that we can plan for future projects.

Tasks requiring post ALARA reviews are determined by the ALARA coordinator using the criteria that you see here, and typically the ALARA coordinator will take a certain percentage of routines, special and other outage type jobs to determine the effectiveness of the overall program.

The Nine Mile Point ALARA program also provides for the establishment of annual exposure goals and specific ALARA goals. An annual exposure goal is an anticipated exposure value calculated by estimating the radiation exposures to personnel performing plant cyclic activities.

It is the responsibility of the site ALARA committee to establish and evaluate the effectiveness in achieving this goal, and we receive support from our corporate health physics groups in preparing the annual exposure estimates.

It should be noted, as I indicated earlier, that the 1985 goal has been prorated such to meet the challenge placed upon the industry by Dr. Harold Denton, and we anticipate that our exposure will be less than 400 man-rem for the year 1985.

The final program element in the ALARA program is the ALARA awareness program. It consists of four components to promote an active participation in the ALARA process by site employees and management.

Components of this ALARA awareness program include an ALARA suggestion program, training, ALARA committees, both a site committee and individual unit committees, as well as management overview through the site ALARA committee, the stations operations review committee and the safety review and audit board, which performs an annual audit of our radiation protection program.

That concludes my presentation.

MR. SIESS: Thank you.

Any questions?

MR. EBERSOLE: No.

MR. SIESS: We only have one item left on the agenda which is plant security which will have to be covered in a closed session.

What I propose is to skip that item and go to the last item on the agenda, which will be some comments by me regarding the presentations to the full committee, following which we will adjourn the open portion of the meeting. We will retire to another room ---

MR. ZALLNICK: Yes, sir. We have Conference Room F down at the end of the hall set aside.

MR. SIESS: All right. ---to discuss the plant security.

So I have the following comments to offer to you for the full committee agenda.

You have a copy of the agenda with a number of items as I do, and I will refer to them by numbers.

As far as the staff's presentation is concerned, pretty much the same way you did it here with whatever appropriate changes to the open item part.

I think as far as the ACRS is concerned, the containment items are ones we would just as soon leave to the staff and probably don't need any discussion on them. You can mention what they are, but we won't ask the applicant to have a discussion on that. They are resolvable, in my opinion, to the satisfaction of the staff.

MR. WEINKAM: So you won't require an in-depth discussion as I did on the first five?

MR. SIESS: No.

We will expect to hear from the representative from the region. I think Mr. Collins said he would be at the full committee meeting and we will notify him of the time.

For the applicant the presentation under Item 6, well, that was fairly short.

Item 7 on organization and management we would like to hear.

Item 8 on the safety review committee I don't think we need to hear. If there are questions about it, we will expect you to respond.

Now there are a number of items in the category of no presentation of no presentation, but be prepared to answer questions. I would suggest though that I will try to keep the questions fairly specific, and I will be chairing that portion of the meeting. So a question on a particular subject should not elicit an answer involving the whole presentation that we heard here today.

I will try to keep it specific and try to keep the answer as specific as possible and using only those slides that are pertinent to the answer.

I don't want a question to trigger a long presentation. I want it to trigger an answer, which may be a long presentation, and there will be more than one question.

MR. ZALLNICK: Yes, sir. I understand.

MR. SIESS: So the safety review committee we will keep as a questions only, and I think the open item on that was one that wasn't of too much concern at this time.

The industry interactions I think we could not have a formal presentation on.

The operations staffing and training, yes. I think it could be abbreviated a little, but I gave some specifics on what slides in terms of the slides there.

MR. ZALLNICK: Yes, sir.

MR. SIESS: There will be some questions and it is probably better not to try to give everything. Save some of it back to answer the questions.

MR. ZALLNICK: Yes, sir.

MR. SIESS: On emergency operating procedures, that is very much in the preliminary stage and you have nothing much to say except you are developing them, and again we will leave that for questions if somebody has a question. It will be much more specific than anything you had to present there.

The seismic issue I will leave for questions, although the staff may want to have somebody there. I can't anticipate, but it wouldn't hurt to have somebody there to talk about the faults.

Systems interactions I would like to leave for questions. I suspect there will be some, but it would be a somewhat shorter presentation to answer the questions.

On the decay heat removal item, the material on leak detection I think we could eliminate, but the auxiliary building extension bays and that arrangement of the RHR systems I think is unique and is of interest. So that portion of it would be of special interest I believe.

MR. McKINLEY: Chet, you passed up Item 13.

MR. SIESS: Oh, AC/DC power systems. I thought we would leave that to be handled by questions. The electrical separation comes under another item. That I am particularly interested in, but it wasn't under the AC/DC power systems.

Decay heat removal I mentioned.

The containment, the open item parts I don't think we will need to discuss again, but the unique features of the containment are the things that should be presented. That is the first part of it.

Instrumentation for detecting inadequate core cooling seems to be pretty much the industry and the owners group position. Let's save that for questions.

ATWS, the ATWS as far as the scram systems is concerned is pretty much straight forward. You could simply mention what you provided without going into the details.

But the stuff on the scram reduction efforts I think will be of interest. The other may come up in questions of course.

It is my intention to give the committee a list of areas that you are prepared to answer questions on, which will sort of put them in that direction, but that does not mean that the questions will be limited to those.

MR. ZALLNICK: Yes, sir.

MR. SIESS: You have been around and you can get questions on anything of any sort.

On fire protection, yes, with an emphasis on the separation and some better slides on that. Mr. Rademacher knows what I am talking about.

On the control room, the control room design review is not far enough along or enough different to warrant the time I think there.

The safety parameter display system I would like to have the presentation on that pretty much as it was done here. I don't think the committee has looked at an SPDS recently and you have more detail than I have seen on some of them.

The control room habitability, probably we will save that for questions. Dr. Moeller may have one or two or three.

(Laughter.)

But the remote shutdown capability, we would like to hear a presentation on that, the two units and so forth.

Emergency planning we will save for questions. We may not get any. This is a site that has had emergency planning and has pretty well proved out.

The PRA, I think we would like to hear about that, but the point should be made clearly that this was the PRA under the NTOL. It was mainly for the environmental review, but the uses you are making of it will be of interest.

Intergranular stress corrosion cracking, let's save that for questions. Dr. Shewman may want a little more detail.

Environmental qualification, no presentation, just questions.

The same on the rad protection program, although you will be certain to get a question, but I would prefer to focus it with the question.

And on plant security, we will settle that later.

Now my estimate is that we can reasonably do this and it will give the committee the things I think they are most interested in with time to ask some questions.

Mr. McKinley and Mr. Schiffgens will be there working up a more detailed agenda for the meeting, and if we do have a problem, there may be another item or two that would be cut back to questions only, and one candidate for that right now would be the control room part. I have got it down to two items under control room, the SPDS and the remote shutdown, either of which could be handled by questions. They were both pretty short and I sort of tended to leave them in there.

Do you have any questions about that?

MR. ZALLNICK: No, sir.

MR. SIESS: Jesse, do you have any comments?

MR. EBERSOLE: No.

MR. SIESS: Then I declare the open part of this meeting adjourned. We will not readjourn in this room.

The meeting is adjourned.

(Whereupon, at 3:45 p.m., the meeting adjourned.)

\- - -

CERTIFICATE OF OFFICIAL REPORTER

This is to certify that the attached proceedings before the UNITED STATES NUCLEAR REGULATORY COMMISSION in the matter of:

NAME OF PROCEEDING: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON NINE MILE POINT NUCLEAR STATION, UNIT NO. 2

DOCKET NO.:

PLACE: SYRACUSE, NEW YORK

DATE: THURSDAY, FEBRUARY 21, 1985

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission.

(sigt) Mary Simons/sg

(TYPED)

MARY SIMONS
Official Reporter
ACE-FEDERAL REPORTERS, INC.
Reporter's Affiliation

PASQUALE VOLZA

Supervisor of Radiological Support

NIAGARA MOHAWK

RADIATION PROTECTION PROGRAM

I. Policy

- NMPC Policy is to Ensure that Exposure of Station Personnel, the General Public, and the Environment Will be ALARA
- NMPC has Formally Endorsed INPO's Five REM per Year Exposure Guideline
- Management Oversight Provided to Ensure Compliance

II. Organization

- Experienced Site and Corporate Health Physics Staffs

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RADIATION PROTECTION PROGRAM

III. ALARA Design Objectives

- Minimize Amount of Time Plant Personnel Spend in Radiation Areas
- Minimize Radiation Levels in Routinely Occupied Plant Areas and in the Vicinity of Plant Equipment

IV. Operational ALARA Program Elements

- Dedicated ALARA Group
- Pre-ALARA Work Reviews
- Post-ALARA Work Reviews
- ALARA Goals Program
- ALARA Awareness Program

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SECOND PROJECTOR SLIDES

RADIATION PROTECTION PROGRAM

Nine Mile Point ALARA Commitment
(Policy 6.1.7.1)

Purpose: The Purpose of this Policy is to Acknowledge the Company's Responsibility for a Radiation Exposure Objective to be as Low as Reasonably Achievable (ALARA)

Policy: "The Company Policy is to Commit Sufficient Resources to Ensure that Exposure of Station Personnel, the General Public, and the Environment Will be as Low as Reasonable Achievable . . . etc."

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RADIATION PROTECTION PROGRAM

Management Oversight to Radiation Protection Program

- Site Radiation Protection Staff
- Site Operations Review Committee (SORC)
- Safety Review and Audit Board
- Quality Assurance Department

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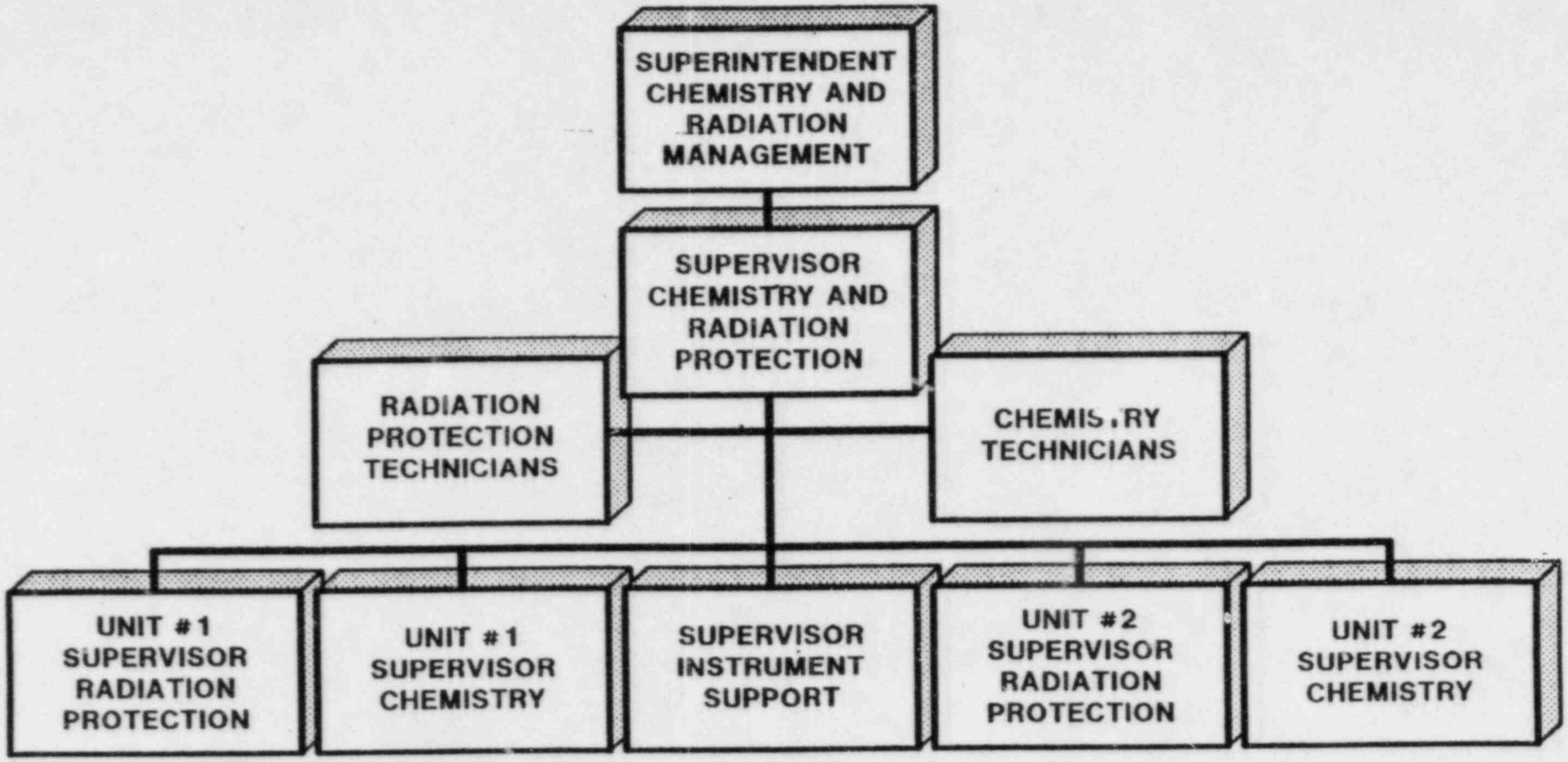
RADIATION PROTECTION PROGRAM
CHEMISTRY AND RADIATION PROTECTION OPERATIONS GROUPS



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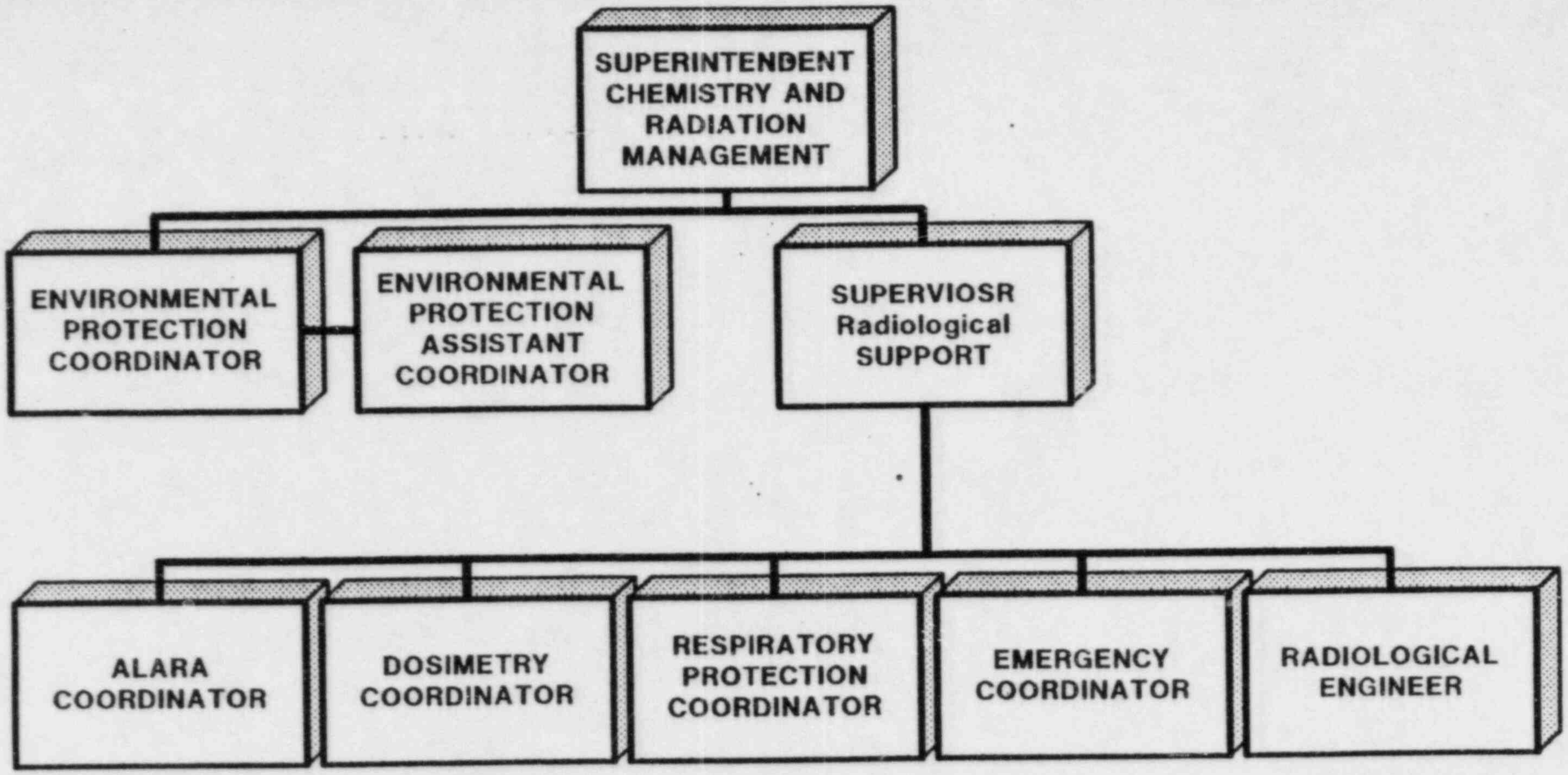
RADIATION PROTECTION PROGRAM
RADIOLOGICAL SUPPORT AND ENVIRONMENTAL PROTECTION GROUPS



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RADIATION PROTECTION PROGRAM

Examples of Design Considerations

1. Operator Interface Control Panels for Systems Containing Radioactive Materials have been Placed in Low Background Areas (i.e., Rx Water Cleanup System)

2. A Dedicated Control Rod Drive Maintenance Facility with a Supporting Ventilation System has been Constructed to Minimize Radiation Exposure to Other Work Groups

3. Generally, Components (Pumps, Valves, etc.) have been Separated by Shield Walls to Reduce Exposure Levels to Workers from Unrelated Components (Tanks, Pumps, etc.)

4. Reach Rods to Valves have been Provided where Appropriate for the Manual Operation of the Reactor Water Cleanup System

5. Systems Containing Radioactive Materials have been Reviewed for Flushing and Decontamination Fixtures. The Review Resulted in the Verification of Existing Connections or the Recommended Placement of New Connections

6. The Nine Mile Point Unit 2 Model was Extensively Used by the Alara Review Committee for the Alara Design Review. The Model was Referenced to Evaluate Access Control, Shielding Design, Pipe Runs and Equipment Placement

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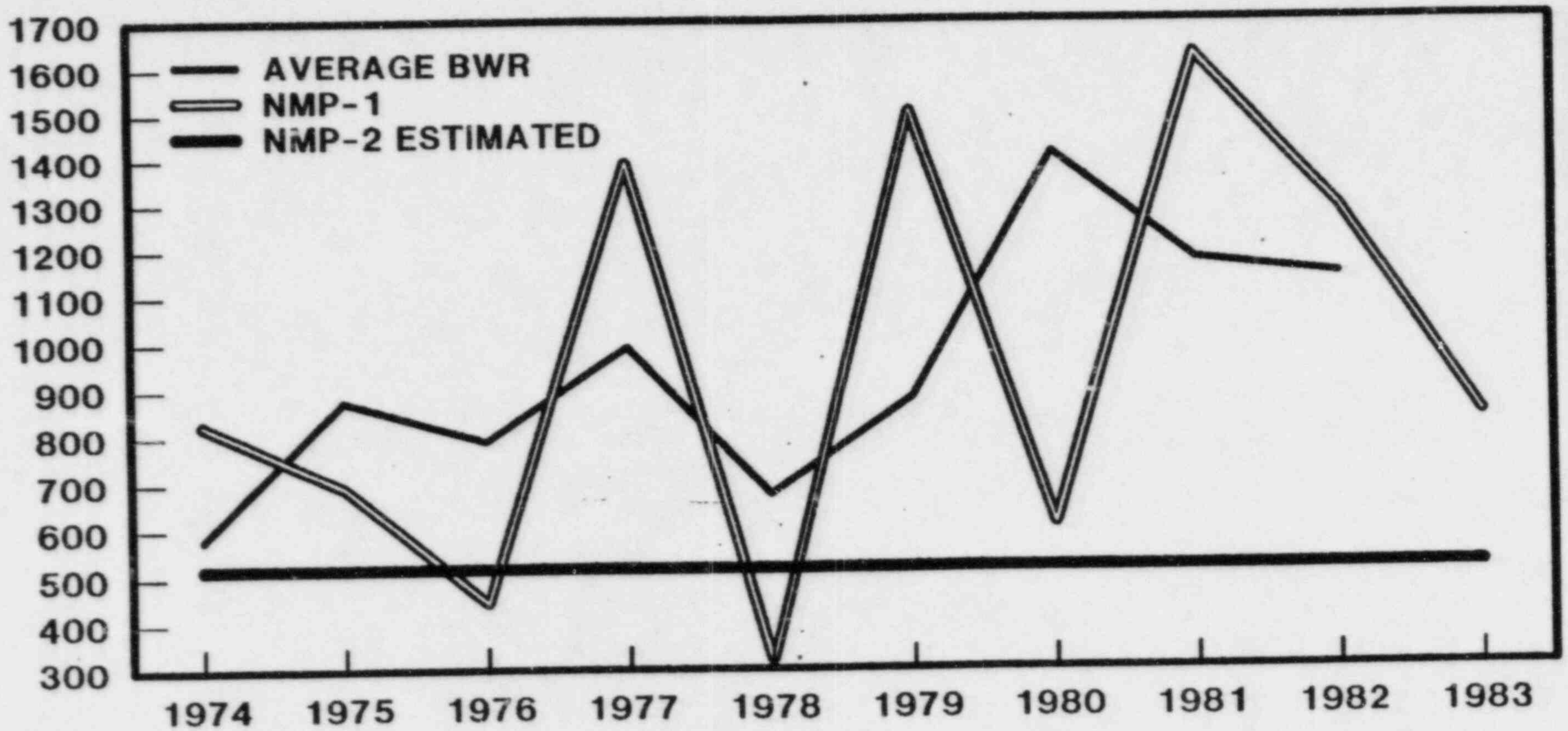
RADIATION PROTECTION PROGRAM
ANNUAL OPERATIONAL MAN-REM EXPOSURE



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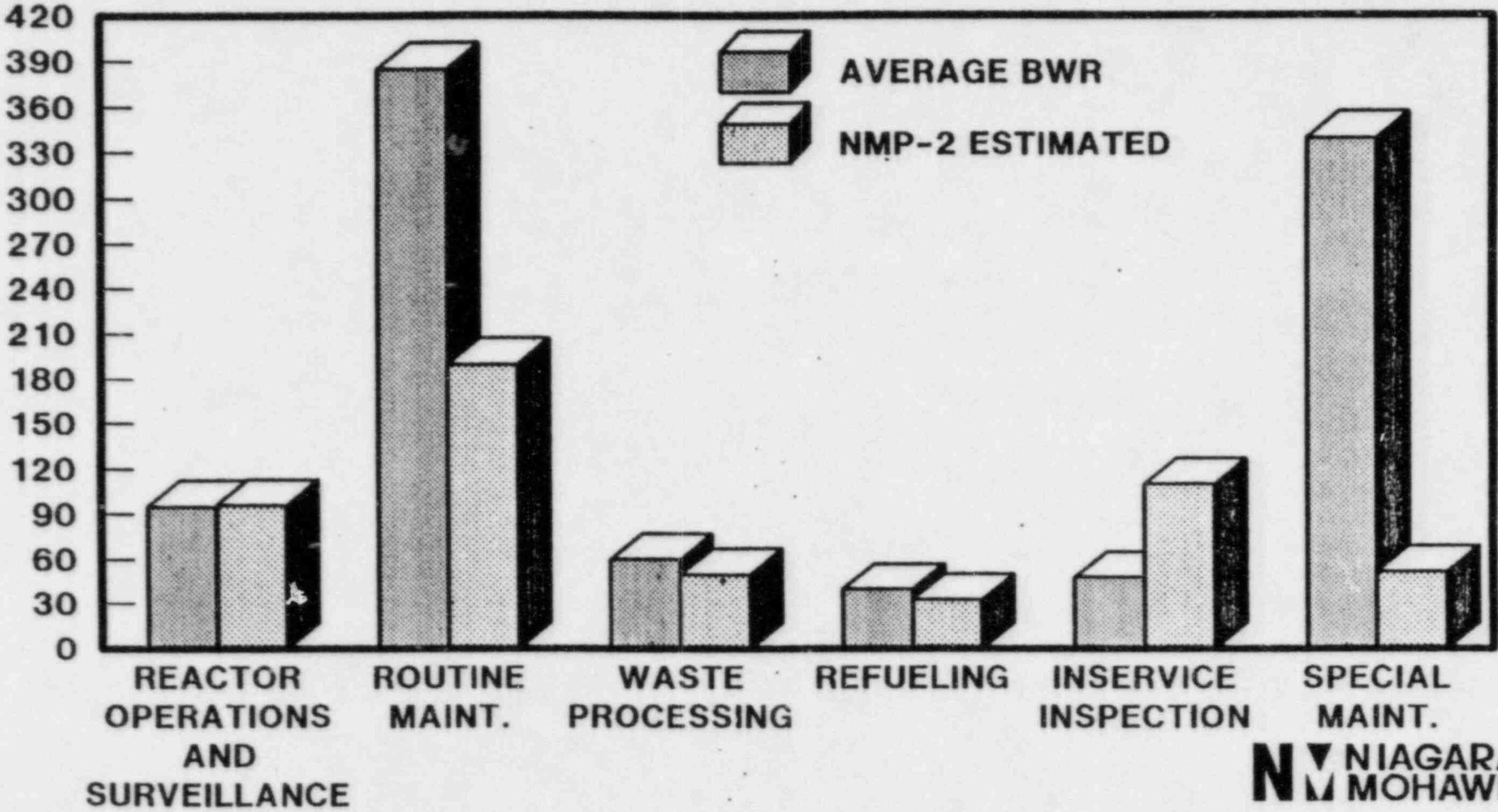
RADIATION PROTECTION PROGRAM

OCCUPATIONAL RADIATION DOSE BY WORK FUNCTIONS



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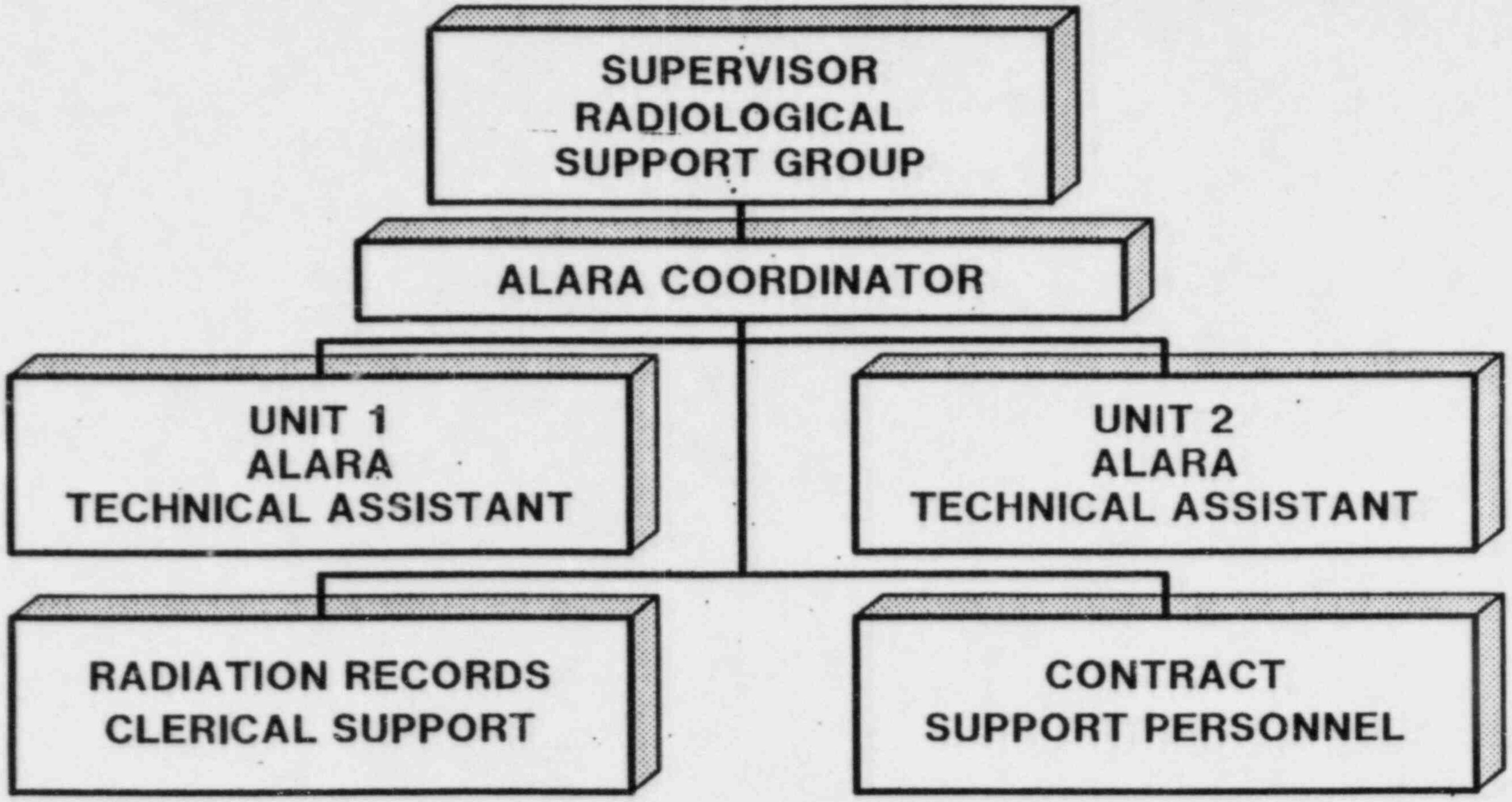
RADIATION PROTECTION PROGRAM
ALARA GROUP ORGANIZATION



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RADIATION PROTECTION PROGRAM

Pre-ALARA Work Review Action Levels

Estimated Dose Range		Review Performed By
Collective:	Dose < 1 MAN-REM	Rad. Prot. Technician
Individual:	Dose < .25 REM	
Collective:	> 1 MAN-REM < 5 MAN-REM	Alara Group
Individual:	> .25 REM < .5 REM	
Collective:	> 5 MAN-REM < 25 MAN-REM	Alara Group
Individual:	> .5 REM < 2.5 REM	
Collective:	> 25 MAN-REM < 50 MAN-REM	Alara Committee
Individual:	> 2.5 REM < 5.0 REM	
Collective:	> 50 MAN-REM	Alara Committee
Individual:	> 5.0 REM	

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RADIATION PROTECTION PROGRAM

Post-ALARA Work Review Criteria

- Routine Tasks
- Special Tasks
- Outage Tasks
- Tasks Performed on Extended RWP's
- Tasks Receiving Greater than 25 MAN-REM

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RADIATION PROTECTION PROGRAM

Nine Mile Point ALARA Goals Program

The Annual Exposure Goal is an Anticipated Exposure Value Calculated by Estimating the Radiation Exposure to Personnel Performing Planned Activities

A Specific ALARA Goal is a Statement of a Desired Objective Clearly Exemplified with a Means of Measuring Goal Achievement and a Specific End Point

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RADIATION PROTECTION PROGRAM

ALARA Awareness Program

- ALARA Suggestions
- Training
- ALARA Committees
- Management Overview
    - Site ALARA Committee
    - SORC
    - SRAB

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EDWARD R. KLEIN

Assistant Manager
Project Engineering

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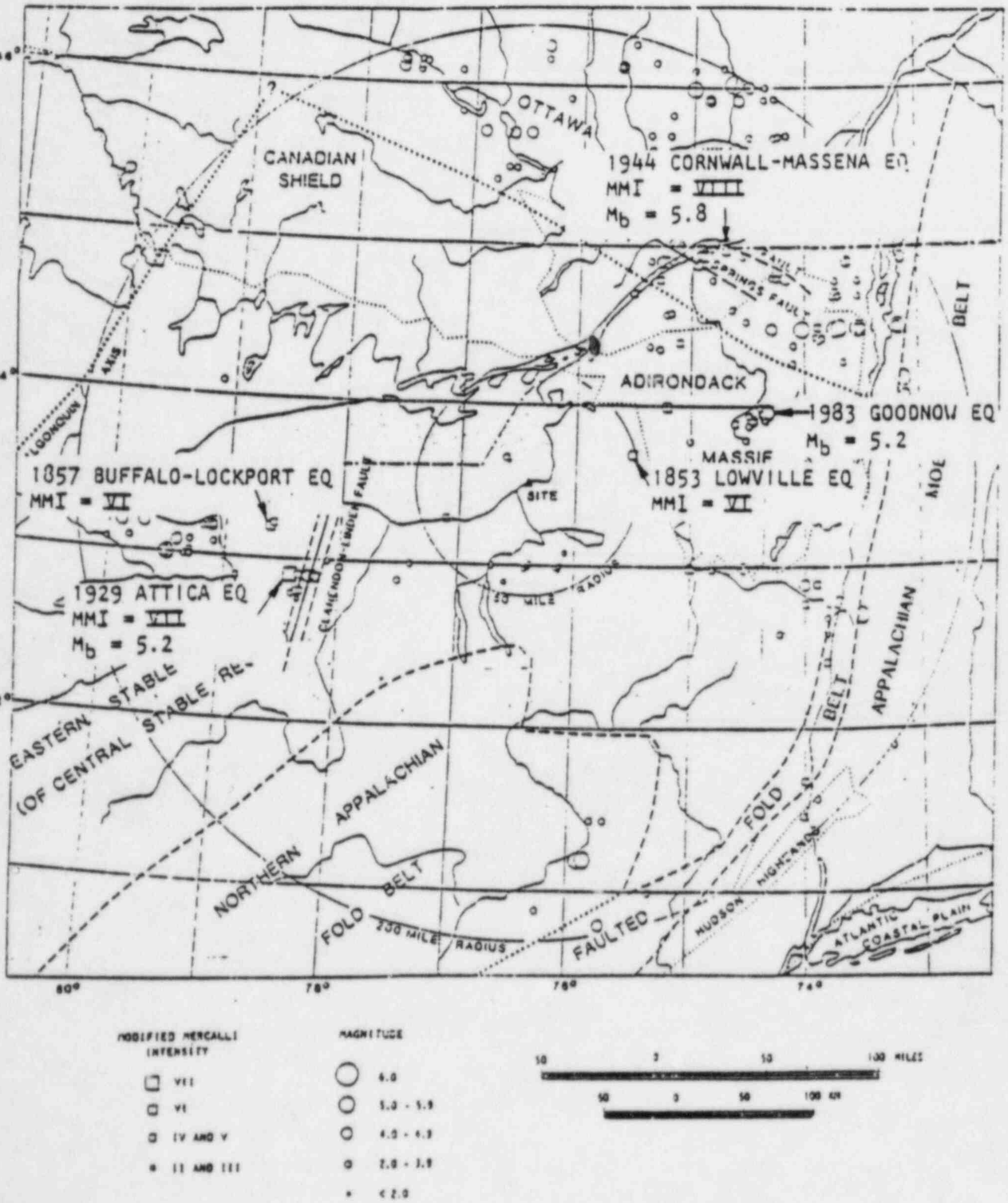


TECTONIC PROVINCES SHOWING SIGNIFICANT EARTHQUAKES

FIGURE 1

REC. ID. (1)	EARTHQUAKE	DATE/TIME (2)	STATION	DISTANCE	MAGNITUD
S68/8	OROVILLE	8/8/75/0700	STATION 6	LESS THAN 20	4.9
S89/27	OROVILLE	9/27/75	STATION 8	LESS THAN 20	4.6 (3)
S99/27	OROVILLE	9/27/75	STATION 9	LESS THAN 20	4.6 (3)
IBO37	PARKFIELD	6/27/66	TEMBLOR	6	5.5
IW335	LYTLE CREEK	9/12/70	ALLEN RANCH	19	5.4
ROC132	FRIULI	9/11/76/1631	S. ROCO	16	5.5
	C. MENDOCINO	6/7/75	CAPE MENDOCINO	20	5.3
BO25	HELENA	10/3/35	CARROLL COLLEGE	7	5.7 (4)
U297	HELENA	11/28/35	FEDERAL BLDG.	5	5.0 (4)
	OROVILLE	8/1/73	SEISMOGRAPH STA.	15	5.7

(1) RECORD ID USED FOR RECORDS FROM NUREG/CR-1582

(2) TIME IS GIVEN ONLY IF REQUIRED TO IDENTIFY EARTHQUAKE

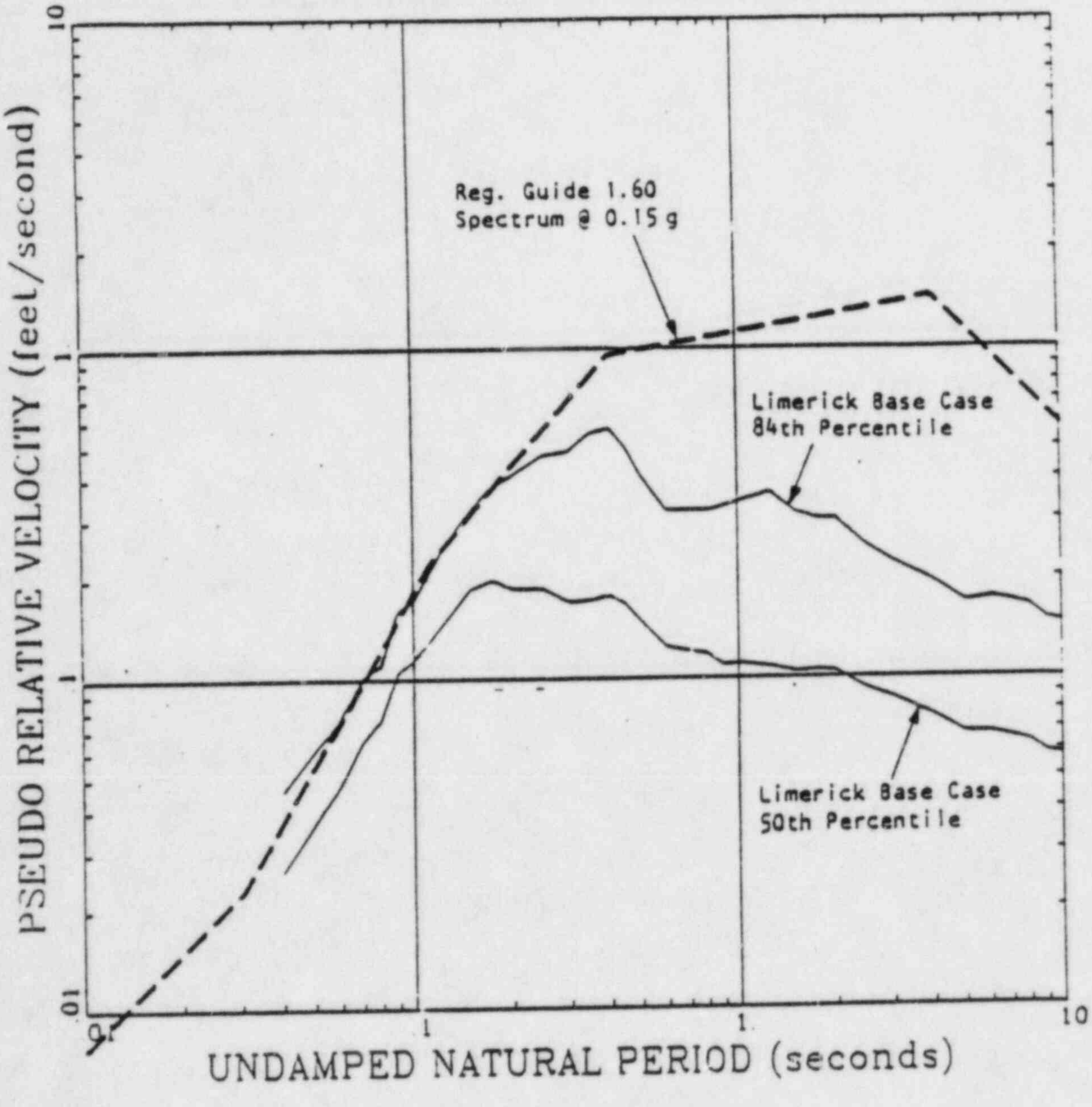
(3) SOMEWHAT LOWER THAN RANGE DESIRED BUT USED BECAUSE SO FEW ROCK RECORDS ARE AVAILABLE

(4) KANAMORI AND JENNINGS, BSSA (68), 1978

ROCK RECORDS USED (BASE CASE) FOR SITE SPECIFIC SPECTRA

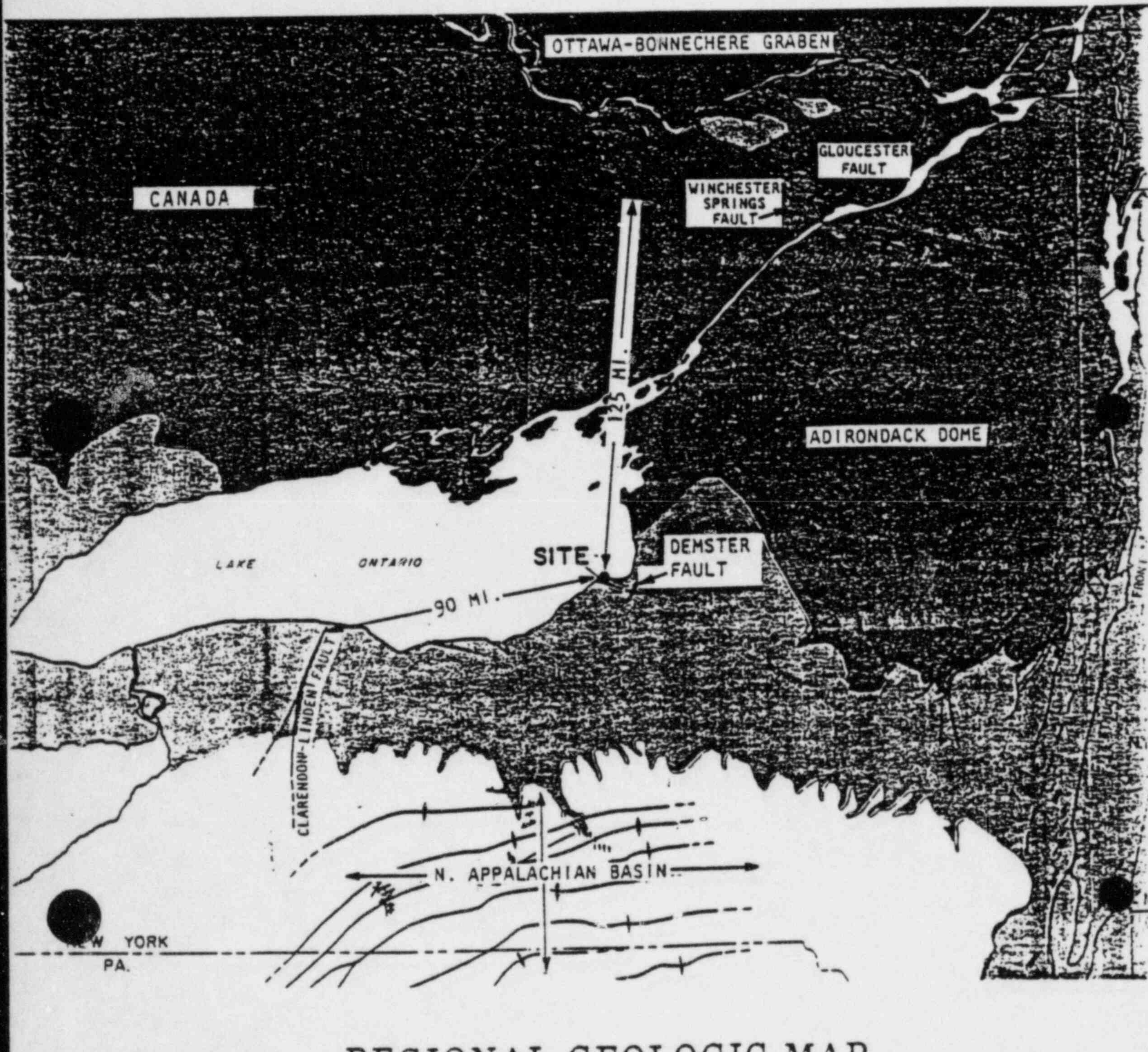
FIGURE 2

REF: LIMERICK SER

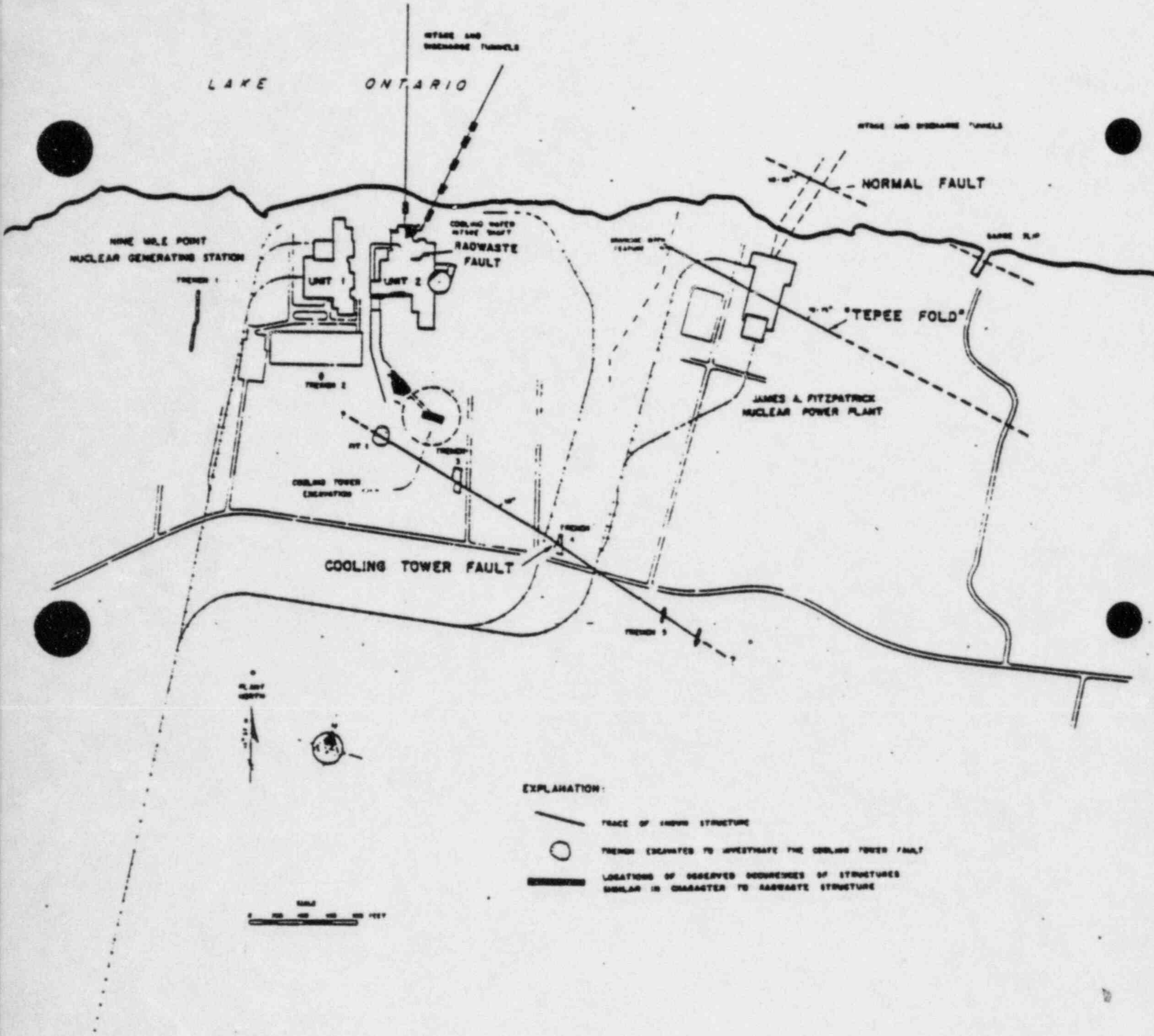


SITE SPECIFIC SPECTRA COMPARED TO REG. GUIDE 1.60

FIGURE 3

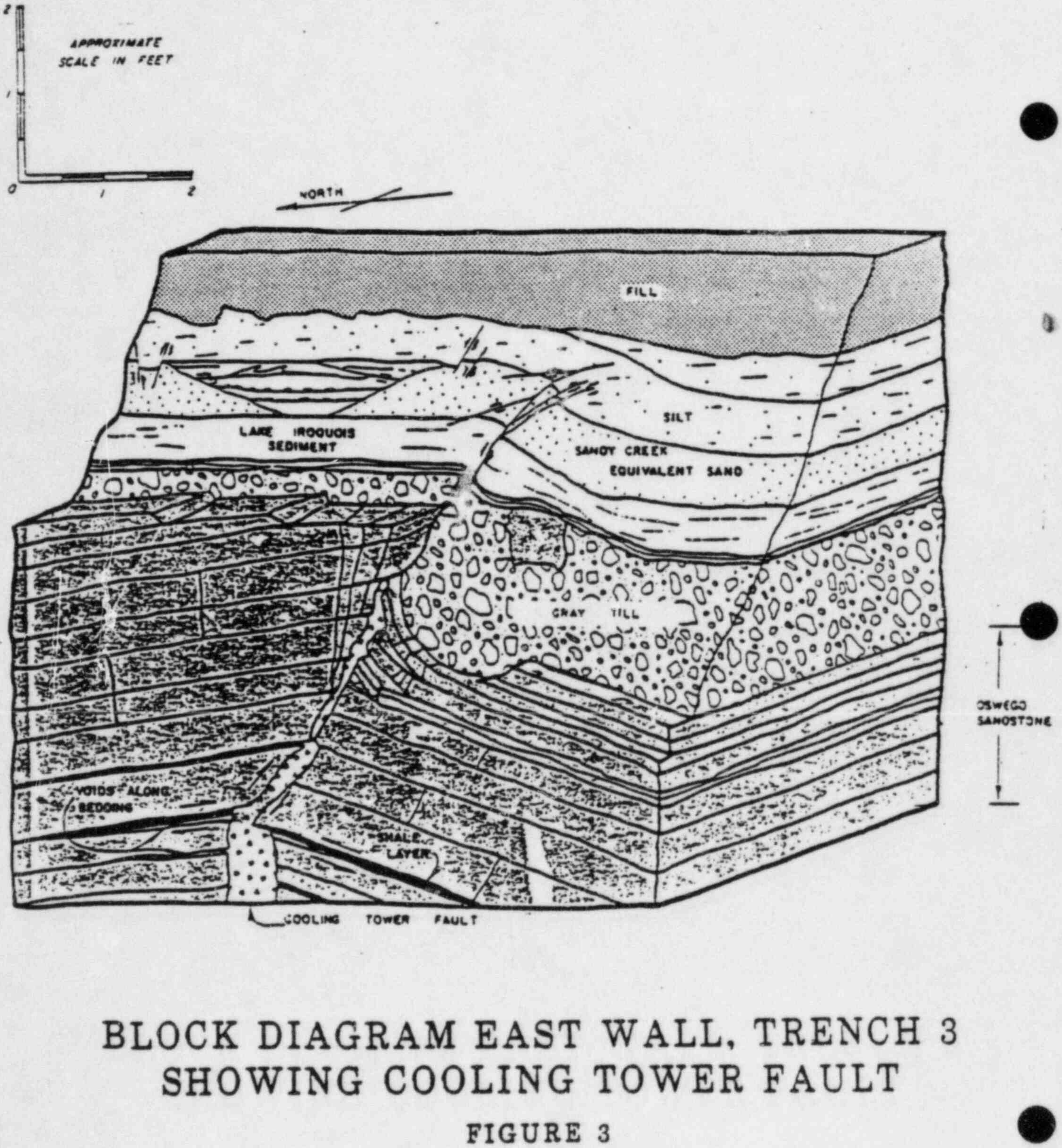


REGIONAL GEOLOGIC MAP
SHOWING SIGNIFICANT STRUCTURES
FIGURE 1



SITE LOCATION MAP SHOWING TRENCHES AND KNOWN GEOLOGIC STRUCTURES

FIGURE 2



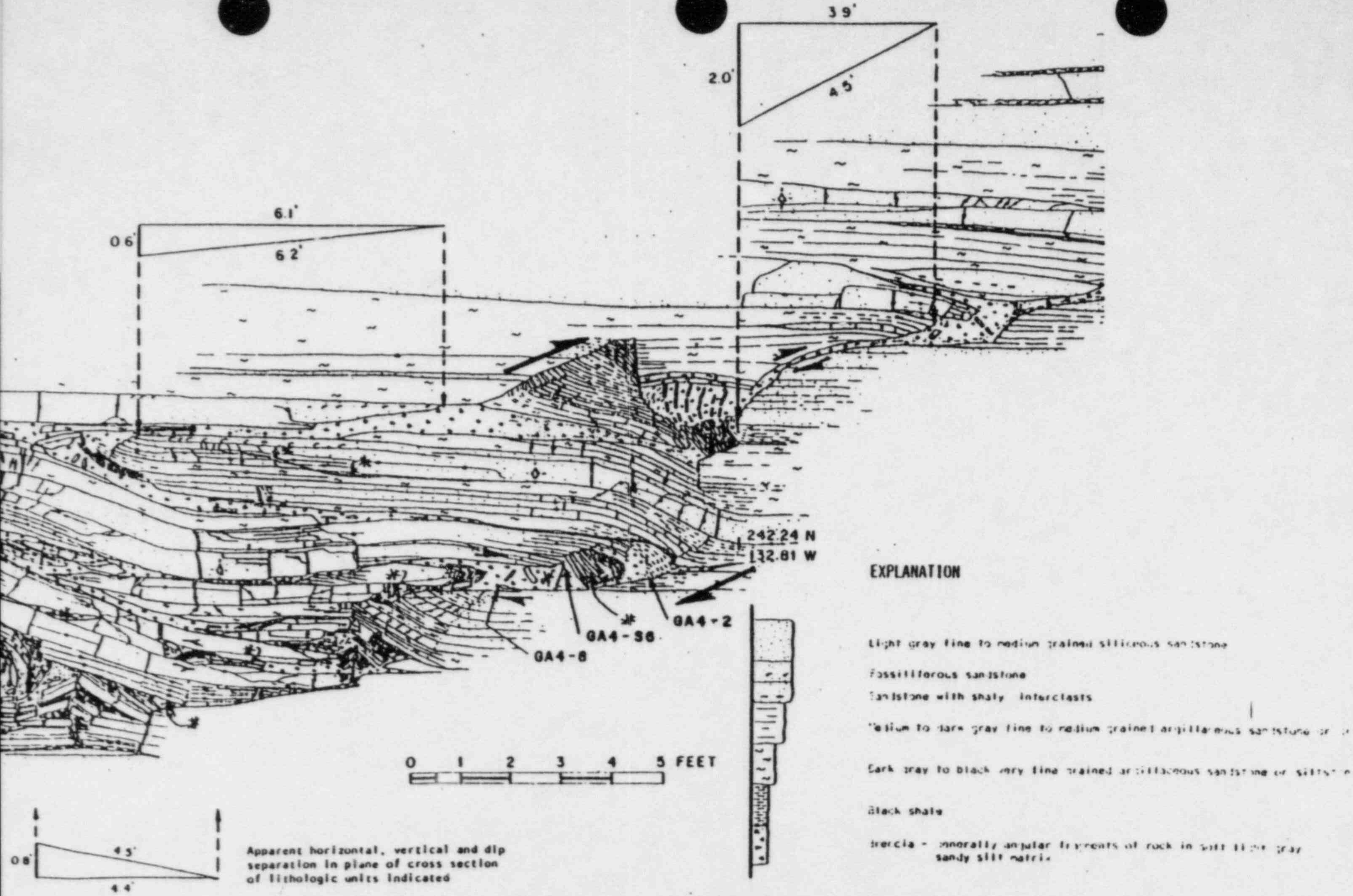
BLOCK DIAGRAM EAST WALL, TRENCH 3
SHOWING COOLING TOWER FAULT

FIGURE 3

COOLING TOWER FAULT

- INITIALLY DEVELOPED AS STRIKE-SLIP FAULT IN LATE PALEOZOIC (225 M.Y.B.P.)
- REACTIVATED AS NORMAL FAULT IN MESOZOIC (100 M.Y.B.P.)
- MOST RECENT MOVEMENT IS BUCKLING TRIGGERED BY GLACIAL PROCESSES
- NON-CAPABLE --- PROCESSES THAT CAUSED MOVEMENT NO LONGER ACTIVE

FIGURE 4



RADWASTE THRUST FAULT CROSS-SECTION, SW WALL
NORTH RADWASTE TRENCH

RADWASTE THRUST STRUCTURE

- RESULTED FROM RELIEF OF STORED STRAIN ENERGY TRIGGERED BY FORMATION OF EROSIONAL VALLEY IN BEDROCK TO THE WEST OF THE SITE

- INITIAL DEVELOPMENT ASSOCIATED WITH CRUSTAL LOADING AND UNLOADING DURING WISCONSONAN GLACIATION

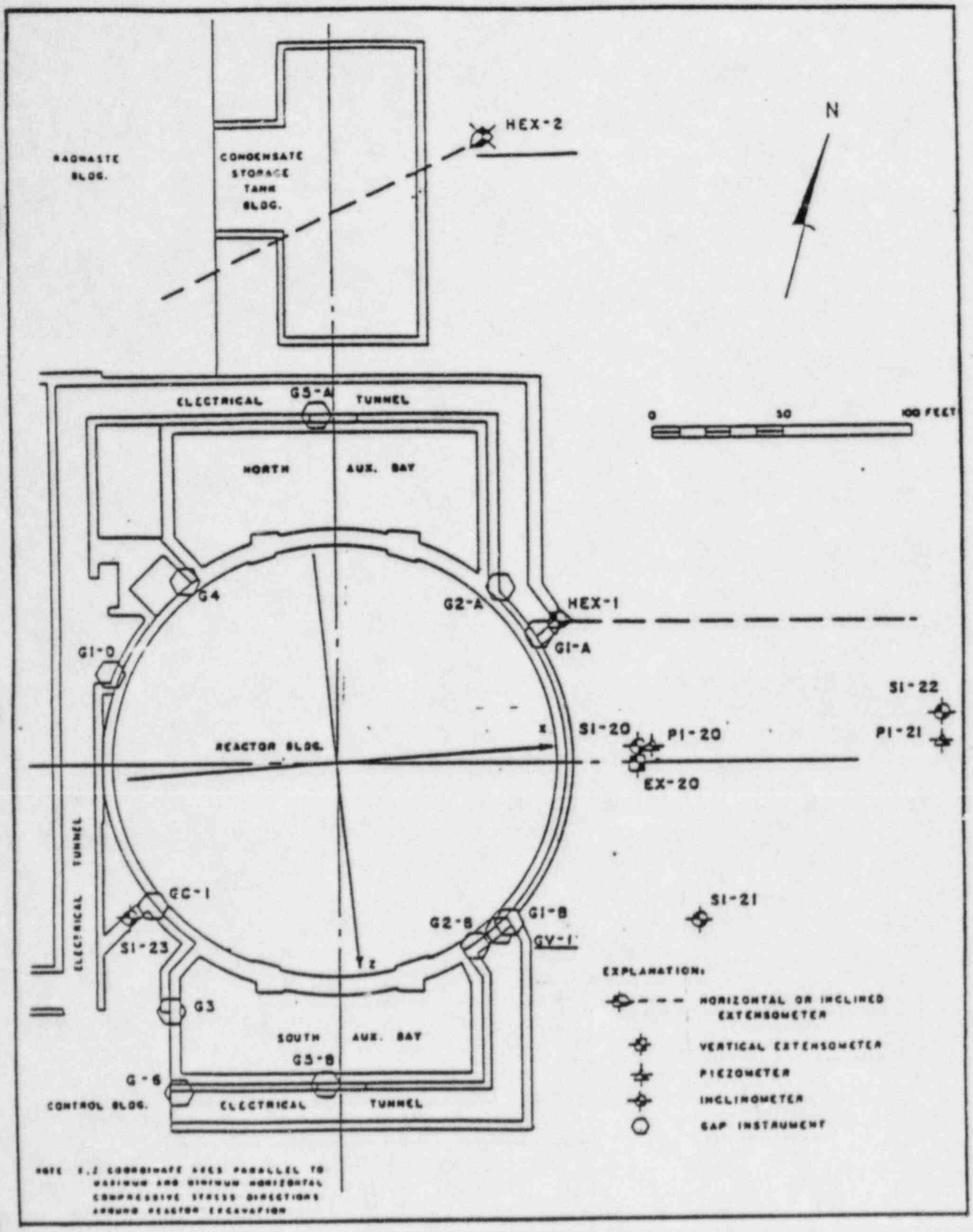
- EVENTS LEADING TO ITS FORMATION ARE NO LONGER EXISTENT

- A DYNAMIC ANALYSIS OF THE STABILITY OF THE RADWASTE THRUST STRUCTURE DURING A SEISMIC EVENT HAS BEEN MADE

- A CONSERVATIVE MOVEMENT OF ONE (1) INCH HAS BEEN USED FOR DESIGN OF STRUCTURES THAT COULD BE AFFECTED BY FUTURE MOVEMENT OF RADWASTE THRUST STRUCTURE

- A SERIES OF INSTRUMENTS WERE INSTALLED TO MONITOR RADWASTE THRUST STRUCTURE MOVEMENTS. THROUGH FOUR (4) YEARS OF MONITORING, MOVEMENTS HAVE BEEN VERY SMALL AND DUE TO CLIMATIC CONDITIONS

FIGURE 6



LINEAR DISPLACEMENT SENSORS

FIGURE 7

CONSERVATISMS IN SEISMIC ANALYSIS

- Damping Values
- Actual Vs. Artificial Earthquakes
- Peak Spreading

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COMPARISON OF DAMPING FACTORS

Type of Structure or Component	Damping Factors SSE Level	
	NMP2 Design Basis	Realistic Damping Values
Small Dia. Piping System Dia. < 12"	2	3
Equip. & Large Dia. Piping System	3	5
Welded Steel Structures	4	7
Bolted Steel Structures	7	15
Reinforced Concrete Structures	7	10

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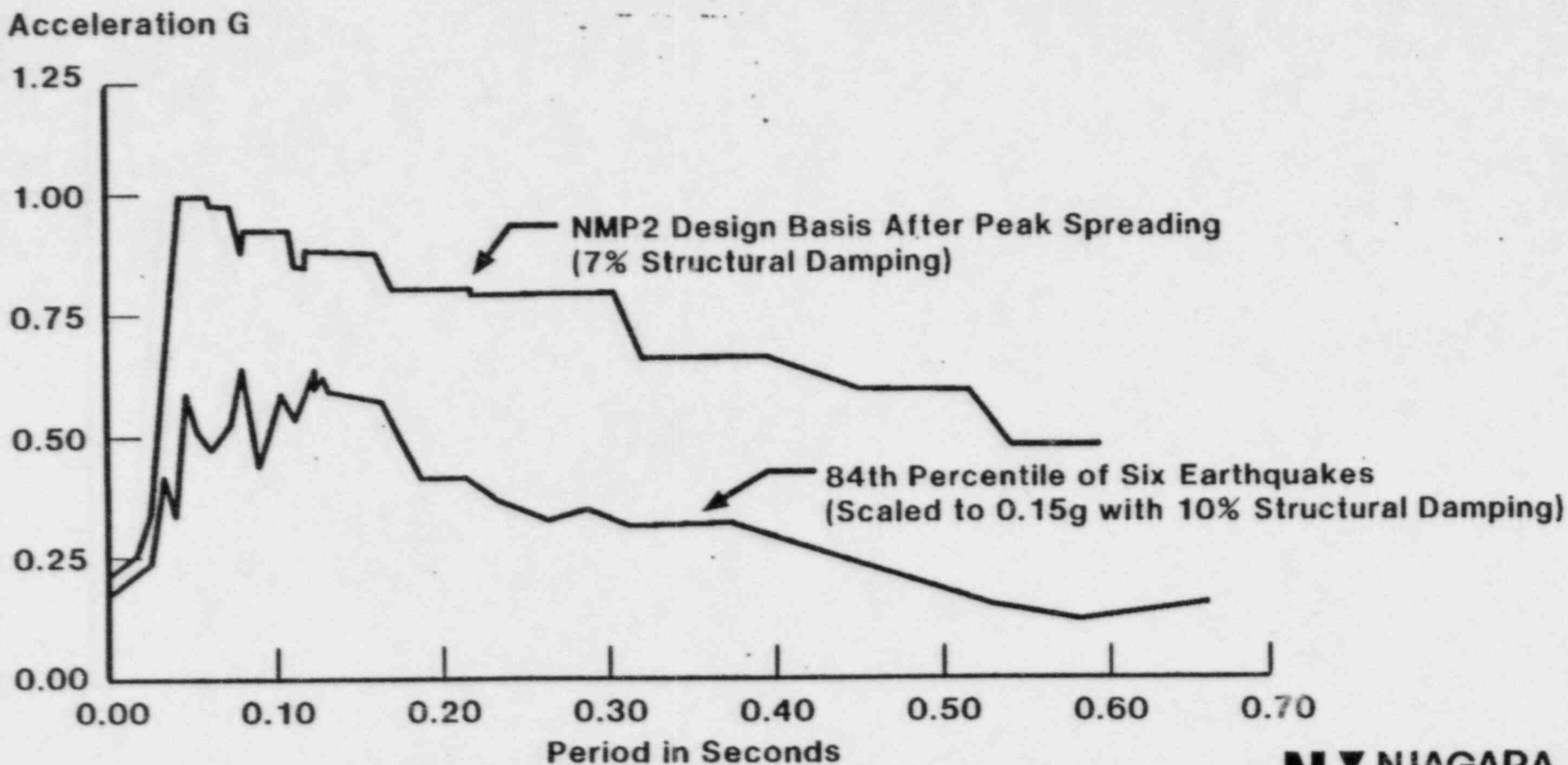
COMPARISON OF NMP2 DESIGN BASIS VS ACTUAL EARTHQUAKE



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DOUGLAS PIKE

Asst. Manager, Project Engineering
Systems Design

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EQUIPMENT QUALIFICATION

NMP2 Base Commitments

- NUREG 0588 Category II Plant
- IEEE 323-74
- 10CFR 50.49
- IEEE 344-75
- Regulatory Guide 1.100

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EQUIPMENT QUALIFICATION

Bases

- Plant Divided into Harsh and Mild Zones
- Normal, Abnormal, and Accident Environments Determined for Following:
  - Temperature
  - Pressure
  - Humidity
  - Radiation
- Submergence and Water Spray Considered Where Applicable

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EQUIPMENT QUALIFICATION

Mild Environment

- Located Outside of Primary and Secondary Containment
- Not Subject to Accident Environments Due to LOCA or Pipe Breaks

Program

- Safety Function and Environmental Conditions Specified in Procurement Specification
- Vendor Certifies Equipment will Satisfy Specification Requirements
- Vendor to Identify any Surveillance or Maintenance Requirements Necessary to Maintain Qualification

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EQUIPMENT QUALIFICATION

Harsh Environment

BOP Program

- Class 1E Safety-Related Electrical Equipment
- Nonsafety-Related Electrical Equipment whose Failure Could Prevent the Function of Safety-Related Electrical Equipment
- Post-Accident Monitoring Equipment - R.G. 1.97

Requirements Specified in Vendor Procurement Specification

- Vendor Performs Qualification
- Vendor Submits Test Plans & Reports for Acceptance

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EQUIPMENT QUALIFICATIONS

Qualification Methods

- Type Testing Identical Equipment
- Type Testing Similar Equipment with Analysis
- Experience with Identical or Similar Equipment

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EQUIPMENT QUALIFICATION

NSSS Program

- G.E. Generic Program
- NMPC Provides Technical Guidance Through Technical Review Committee

Program Bases

- NUREG 0588
- IEEE 323-74
- IEEE 344-75

Qualification Methods

- Testing is Preferred
- Partial Test with Analysis
- Operating Experience

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EQUIPMENT QUALIFICATION

Mechanical Equipment in Harsh Environments

- The Mechanical Equipment Qualification Program Establishes the Qualified Life of Safety Related Nonmetallic Components

Qualification Methods

- Identification of Safety Related Mechanical Equipment
- Develop Environmental Conditions
- Identification of Organic Materials
- Development of Component Thermal and Radiation Service Life
- Evaluate Environmental Conditions versus Capability
- Document Qualified Life

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EQUIPMENT QUALIFICATION

Seismic/Dynamic Qualification Program

- IEEE 344-1975
- Regulatory Guide 1.100

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EQUIPMENT QUALIFICATION

Scope

- Safety-Related Mechanical Equipment
- Safety Related Electrical Equipment
- Non-Safety Related Equipment Whose Failure Could Jeopardize Safety Related Functions

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EQUIPMENT QUALIFICATION

Qualification Methods

- Analysis
- Testing
- Combined Analysis/Testing

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DONALD PRACHT

Lead Mechanical Engineer

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INTERGRANULAR STRESS CORROSION CRACKING

NMP2 Approach to IGSCC

- NUREG 0313 Rev. 1
- Significant Upgrades
- Additional Considerations

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INTERGRANULAR STRESS CORROSION CRACKING

NMP-2 Approach to IGSCC

- Review of Plant Systems
  - Conformance to NUREG 0313 Rev. 1
- Control of Site Procedures & Practices
  - Interpass Temp
  - Chloride & Sulfur Free Expendables - Reg. Guide 1.37
  - Lay-Up - Either Dry or w/Deaerated Demineralized Water

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INTERGRANULAR STRESS CORROSION CRACKING

NMP-2 Approach to IGSCC

- Significant Upgrades (Cont'd.)
- Control Rod Drive
  - Insert & Withdraw Lines Changed from 304 to 304L

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INTERGRANULAR STRESS CORROSION CRACKING

- Significant Upgrades (Cont'd.)
- Recirc. Piping
  - Coincident W/Safe End Replacement Upgrade from 304 to 316 NG (Oct. 4, 1979)

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INTERGRANULAR STRESS CORROSION CRACKING

NMP-2 Approach to IGSCC

- Significant Upgrades -
  - Recirc. Inlet Safe Ends
    - Problems at Duane Arnold in 1978
    - Review of Later Designs of Brunswick & Lasalle 2
    - GE Offer to Replace NMP-2 Thermal Sleeve w/Forged 316L Tuning Fork Design (Committed Oct. 4, 1979)
    - Elimination of Tight Crevices & Residual Stresses on Fabricated Thermal Sleeve

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INTERGRANULAR STRESS CORROSION CRACKING

NMP-2 Approach to IGSCC

- Additional Considerations
- Plant Water Chemistry
  - Philosophy is to Meet EPRI Guidelines Less than .3 Micro MHO/CM
- Hydrogen Water Chemistry and IHSI
  - Would be Considered in Future if Conditions Warranted

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INTERGRANULAR STRESS CORROSION CRACKING

Conclusions

- BWR Owners Group on IGSCC - 1978
- Participation W/NRC, EPRI, GE
- In-House Engineering of Unit 1 Recirc. Piping Replacement
- NMPC Engineering Aware of Emerging Issues
- Monitoring of Mitigating Actions
  - Hydrogen Water Chemistry
  - Induction Heating Stress Improvement

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NORMAN L. RADEMACHER

Nuclear Design Coordinator

Nine Mile Point Unit 2

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PROBABILISTIC RISK ASSESSMENT (PRA)

Introduction

- A PRA is an Analysis of Adequacy for Core Melt Accidents
- Identifies Sequences which Contribute to Risk
- NMP2 PRA was Performed to Meet NRC Interim Rule in Federal Register (45CFR40101)
- PRA's may be Categorized as Full PRA or Mini PRA

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PROBABILISTIC RISK ASSESSMENT (PRA)

Nine Mile Point Unit 2 Mini PRA

- Identifies Accident Sequences Important to Core Melt Risk
- Includes Loss of Offsite Power as an Accident Initiator
- Does Not Include Other External Events such as Flooding, Fire, Seismic
- Includes Sequences of Plant Systems Needed to Mitigate Core Melt Sequences
- Includes Various Containment Failure Modes
- Includes Results of Risks, Socioeconomic Impacts

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PROBABILISTIC RISK ASSESSMENT (PRA)

PRA Inputs

- Accident Event Trees and Functional Success Criteria
- Site Specific Offsite Power Grid Reliability
- Plant Specific Fault Trees
- Generic Component Failure Rate Data
- One Year of Hourly Meteorological Site Data
- Plant Specific Midlife Population Data
- Plant Specific Topographical Data and Socio-economic Data
- Plant Specific Emergency Planning Information

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PROBABILISTIC RISK ASSESSMENT (PRA)

PRA Review

- Nuclear Engineering - 2 Nuclear Engineers
- Station Operations - Station Superintendent/SSS
- Project Engineering Manager - Lead Engineers (Structural, Electrical, Mechanical, I&C)
- Environmental Department - Manager and Environmental Engineer
- Radiation Management/Radio Chemistry - Superintendent

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PROBABILISTIC RISK ASSESSMENT (PRA)

Summary of Nine Mile Point Unit 2 PRA

- Fulfills NRC Requirements for Class 9 Accidents
- Provides an Estimate of Plant Risk
- Provides a Check on Emergency Effectiveness
- Provides an Indication of Risk Mitigating Features
- Provides an Indication of Dominant Risk Contributors

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PROBABILISTIC RISK ASSESSMENT (PRA)

PRA Conclusion

- Further PRA's Include: Unit 1
  Unit 2 (Upgrade)
- Training on the Uses of PRA to Corporate Site Staff
  - Mitigating Core Damage - Operations
  - Engineering Training

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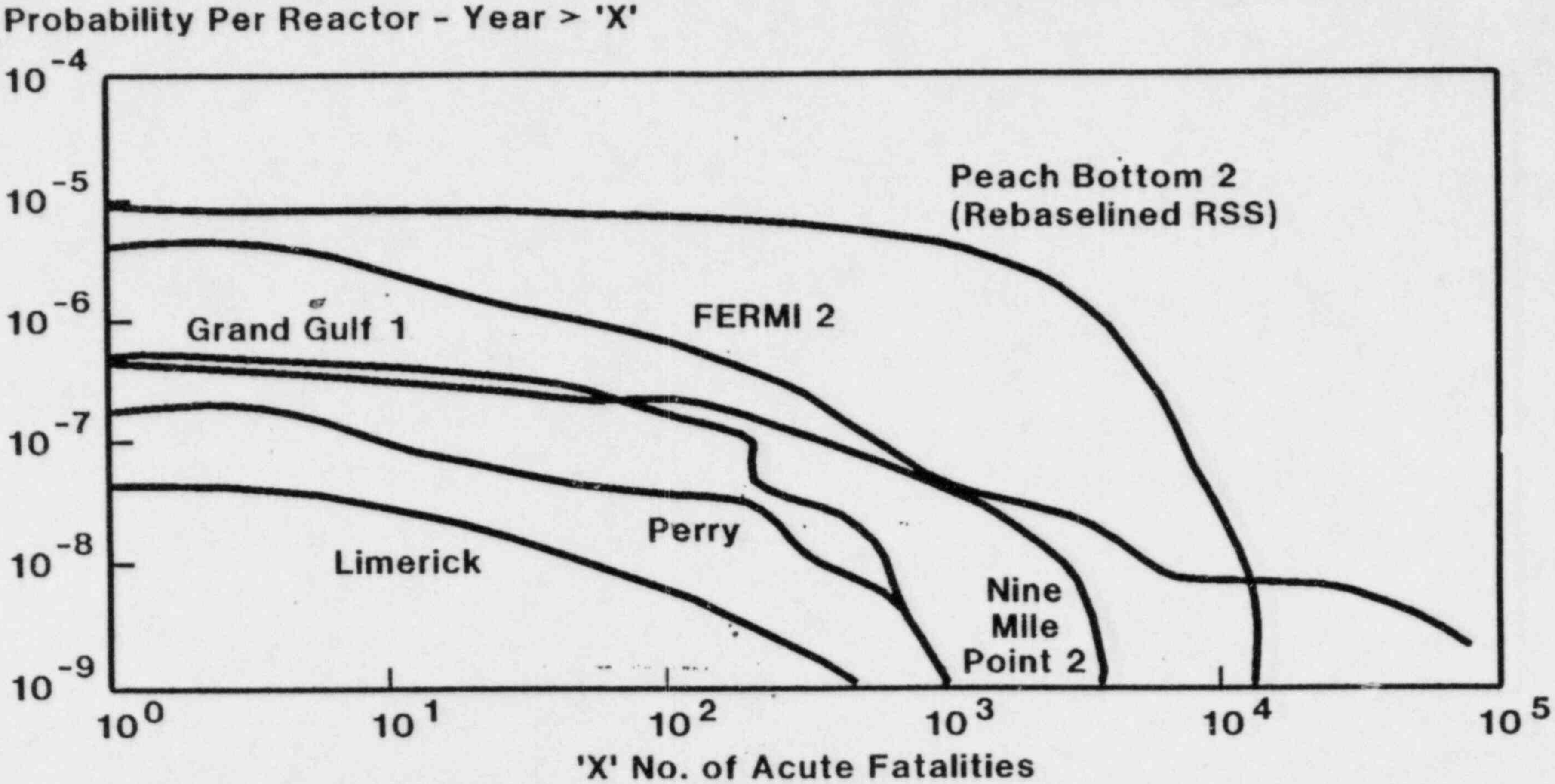
PROBABALISTIC RISK ASSESSMENT (PRA)

Acute Fatalities - BWR Comparison



The Y-Axis Units for the Limerick Curves are:
Frequency (Events/Year) > 'X'

NIAGARA MOHAWK

PASQUALE VOLZA

Supervisor of Radiological Support

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EMERGENCY PLANNING

Emergency Response Organizations

Staffing Level I -	Shift Personnel
Staffing Level II -	Shift Personnel Technical Support Center Operations Support Center
Staffing Level III -	Shift Personnel Technical Support Center Operations Support Center Emergency Operations Facility Joint News Center

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EMERGENCY PLANNING

Coordination with Participating Government Agencies

- State and Local - Lead Agencies
  - New York State Disaster Preparedness Commission
  - Oswego County Office of Emergency Preparedness
- Federal - Lead Agencies
  - USNRC
  - USDOE - Brookhaven Area Office
- International - Lead Agency
  - Canada - Ministry of the Solicitor General

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NMP2 EMERGENCY PLANNING

III. Coordination with Governmental Agencies

- NYS and Oswego County
- NRC & DOE
- Canadian Ministry of the Solicitor General

IV. Emergency Response Facilities

- Emergency Response Facilities Built and Staffed in Accordance with Sup. 1 NUREG 0737 and NUREG 0654

V. Exercise Experience

- NMPC has Successfully Conducted Three Exercises

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NMP2 EMERGENCY PLANNING

I. NMP2 Emergency Preparedness

- Incorporated into an Already Existing and Successful Emergency Preparedness Program Supporting NMP1
- Local and State Offsite Emergency Plans Approved by FEMA
- Public Alert and Notification System Approved by FEMA

II. Emergency Organizations

- Begins with the Minimum Shift Crew and Expands to Include Other Personnel
- Onsite Emergency Organization Augmented by

  Corporate Support
  INPO Support
  NSSS (GE) Support
  Local Services Support
  Nearby Nuclear Facilities Support

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SECOND PROJECTOR SLIDES

EMERGENCY PLANNING

Support from Other Nuclear Power Plants

- Assist with Equipment and Personnel in Monitoring and Evaluation of On-Site and Off-Site Radiological Situations
- Provide Personnel for Assistance at the Off-Site Emergency Center
- Assist in Communications Between Emergency Centers, the Control Room, and Off-Duty Plant Personnel of Outside Agencies
- Allow Use of Radiation Chemistry Laboratory
- Allow Use of Environmental Lab Facility
- Allow Use of Whole Body Counting Facility
- Allow Use of Geophysical Phenomena Monitoring Equipment
- Allow Use of Backup Meteorological Tower
- Provide Personnel for In-Plant Recovery Operations as Available
- Provide Personnel to Assist in the Joint Information Center, as Available

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EMERGENCY PLANNING

Emergency Response Facilities

Facility	Location
Control Room	306'EL., Control Bldg., NMP2
Technical Support Center	248'EL., Admin. Bldg., NMP
Operations Support Center	261' & 277'EL., Admin Bldg., NMP
Emer. Operations Facility	NMP Training Center, NMP
AEOF	Niagara Mohawk Service Center; Volney, N.Y.
Joint New Center	Naval Militia Bldg., Oswego, N.Y.
Corp. Emer. Operations Facility	Niagara Mohawk Corp., Headquarters: Syracuse, N.Y.

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EMERGENCY PLANNING

Nine Mile Point Exercise Experience

1981 - NMPC Conducted the First Successful Full Scale NRC/FEMA Observed Exercise in New York State

1982 - NRC Participated in a Successful Nine Mile Point Site Full Scale NRC/FEMA Observed Exercise with JAFNPP

1983 - NMPC Conducted its Second Successful Full Scale NRC/FEMA Observed Exercise

1984 - NMPC Conducted a Successful Small Scale NRC Observed Exercise

1985 - NMPC to Conduct its Third Full Scale NRC/FEMA Observed Exercise

NIAGARA MOHAWK

DOUGLAS PIKE

Asst. Manager, Project Engineering
Systems Design

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CONTROL ROOM HUMAN FACTORS DESIGN REVIEW

- Nine Mile Point Unit 2 is Conducting a Control Room Design Review in Accordance with the Guidance Provided in NUREG 0700 & 0737 Supplement 1

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CONTROL ROOM HUMAN FACTORS DESIGN REVIEW

History of Control Room Review

- PGCC Design Reviews in 1976-77 Determined Control Room Controls & Alarms
- Control Panel Mock-Up Review in 1977 Determined Location of Controls
- Final Design Reviews in 1978-1979 Repeated PGCC Reviews
- Reviews Performed by Experienced NMPC Engineering & Operations Personnel
- Mini-Human Factors Review of Actual Control Room Panels Conducted in 1982

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CONTROL ROOM HUMAN FACTORS DESIGN REVIEW

CRDR Phases

- Operating Experience Review
- Inventory Review
- Task Analysis
- Panel Survey
- Assessment & Improvement

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CONTROL ROOM HUMAN FACTORS DESIGN REVIEW

Additional Features

- SPDS Integrated into Review Process
- Remote Shutdown Panel Reviewed
- TSC and EOF Layout Reviewed

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SECOND PROJECTOR SLIDES

(PHOTOGRAPHS ARE NOT INCLUDED)

CONTROL ROOM HABITABILITY

- NUREG 0737 Requirements
  - Radiological Protection
  - Toxic Gas Protection
- Unit 2 Design
  - Shielding
  - HVAC System

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CONTROL ROOM HABITABILITY

Accident Scenarios for Radiological Habitability

- LOCA
- Main Steam Line Break Outside Containment
- Control Rod Drop Accident
- Fuel Handling Accident
- Results of Analyses Done in Accordance with Regulatory Guides & SRP (30 Day Limits)
- Radiological Analysis Results
  - All 30 Day Dose Limits are Satisfied

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CONTROL ROOM HABITABILITY

Toxic Gas Analysis Performed

- Nitrogen
- Sulfuric Acid
- Carbon Dioxide
- Propane
- Halon 1301
- Hychocloric Acid
- Chlorine
- Conclusion
  - No Potential for Operator Incapacitation

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CONTROL ROOM HABITABILITY
HABITABILITY DESIGN FEATURES

Shielding

- Walls & Roof 2 Ft. Thick Reinforced Concrete
- Interior Walls at Least 1 Ft. Thick
- Floor at Least 9" Thick

Leak Tightness

- Concrete Floors & Walls Leak Tight
- Leak Tight Access Doors - Self Closing
- Penetrations Sealed with Fire Retardant Material

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CONTROL ROOM HABITABILITY

Control Room Ventilating System

- Fully Redundant Safety Grade System
- Seismically & Environmentally Qualified
- Class 1E Instrumentation and Controls
- Two Separted, Redundant Missle and Tornado Protected Air Intakes
- Provides Control Room Boundary with 75°F, 50% Relative Humidity Conditions
- Positive Pressure in Control Room Boundary
- Normal & Emergency Modes of Operation

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CONTROL ROOM HABITABILITY

Additional Features

- Smoke Detectors in Air Intakes Alarm in Control Room
- Both Air Intakes Can be Isolated
- System has Smoke Removal Capability
- Seismic Fire Dampers at Control Room Boundary
- Breathing Air Available
- Lighting Assured for all Conditions

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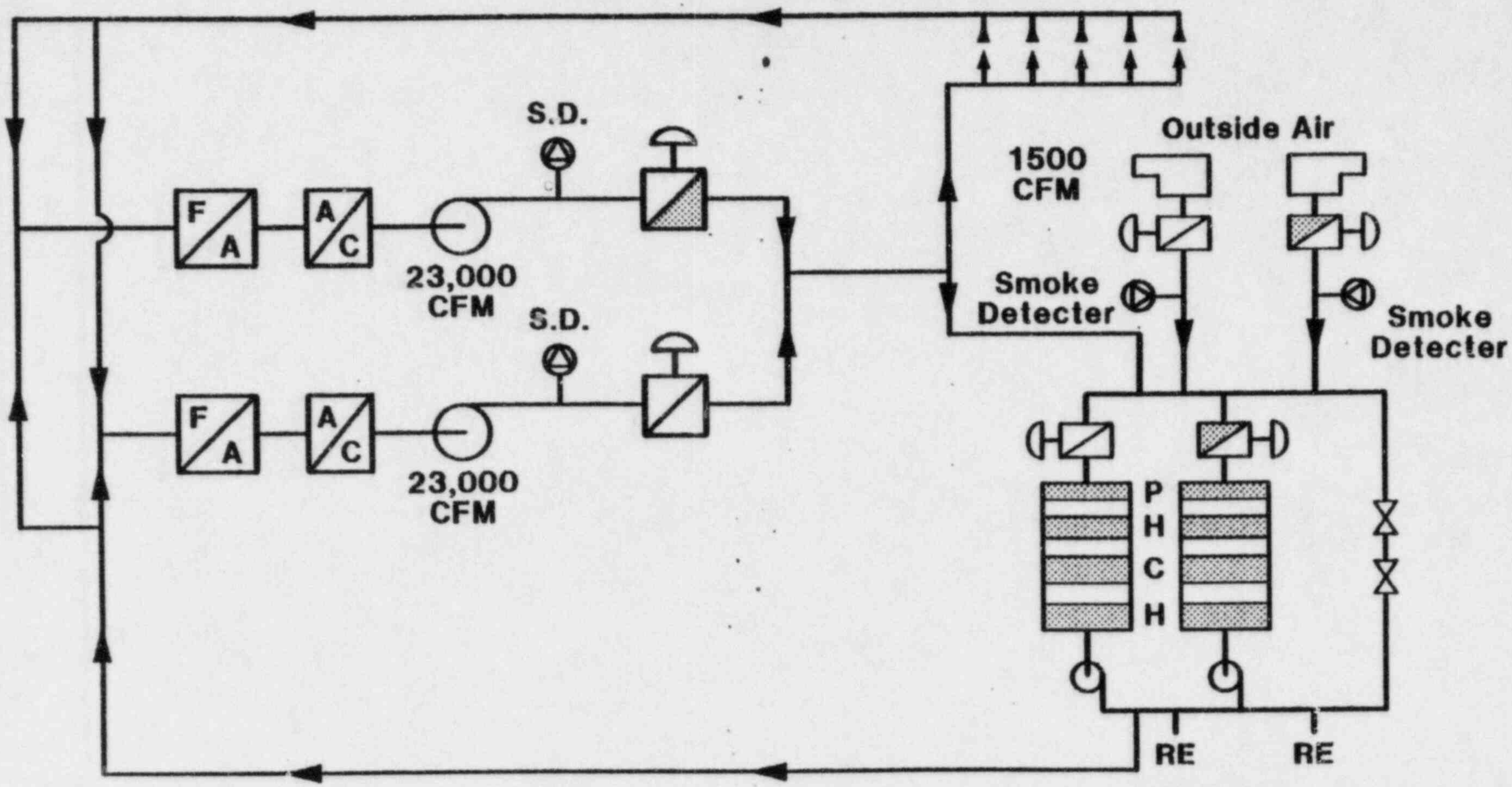
CONTROL ROOM HABITABILITY

Control Room HVAC System



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SECOND PROJECTOR SLIDES

SAFETY PARAMETER DISPLAY SYSTEM

NRC Requirements - NUREG 0737

- Concise Display of Critical Plant Variables
- Aid Operator in Determining Safety Status of Plant
- Located in Main Control Room

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SAFETY PARAMETER DISPLAY SYSTEM

Unit 2 Design Features

- Utilize Quadrex/Honeywell System
- integrated with Radwaste Control & Generator Temperature Monitoring
- Unavailability Goal of .01 for Reactor Operating
- Unavailability Goal of .2 when Shutdown
- Two Redundant 4500 Central Processors with Auto Failover
- UPS Power Supply

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SAFETY PARAMETER DISPLAY SYSTEM

Displays

- Utilize All But One of BWR Owners Group Displays
- Data Base is Subset of R.G. 1.97 Parameters

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SAFETY PARAMETER DISPLAY SYSTEM

Level I Provides Overall Status

- Real Time Values
- Green/Red Color Coding
- Rate/Trend Information
- Level II Status
- Mode Switch Position

Level II Displays for:

- Reactivity Control
- Core Cooling
- Coolant System Integrity
- Containment Integrity

Level II Displays Provide:

- Time History of Past 6 Minutes
- Real Time Values
- Trend/Rate Information
- Status of Level II
- Time, Date, Title
- Mode Switch Position

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SAFETY PARAMETER DISPLAY SYSTEM

Emergency Response Facility System

- R.G. 1.97 Data Base of 552 Analog & Digital Points
- Functions Available in Control Room, TSC, & EOF
    - Graphic Displays (180)
    - Bar Charts
    - Trending of Variables (X-Y Plots)
    - Alarm Summary
    - Group Logs

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SAFETY PARAMETER DISPLAY SYSTEM

ERF Historical Features

- Historical Event Retrieval
    - Trigger Signals
    - 2 Hours Pre-Event/12 Hours Post-Event
    - Recorded in 1, 5 & 30 Second Groups
    - 15 Minute Periods for Up to 14 Days Following 12 Hr. Period
- Historical & Event Trigger Data Accessible from TSC & EOF

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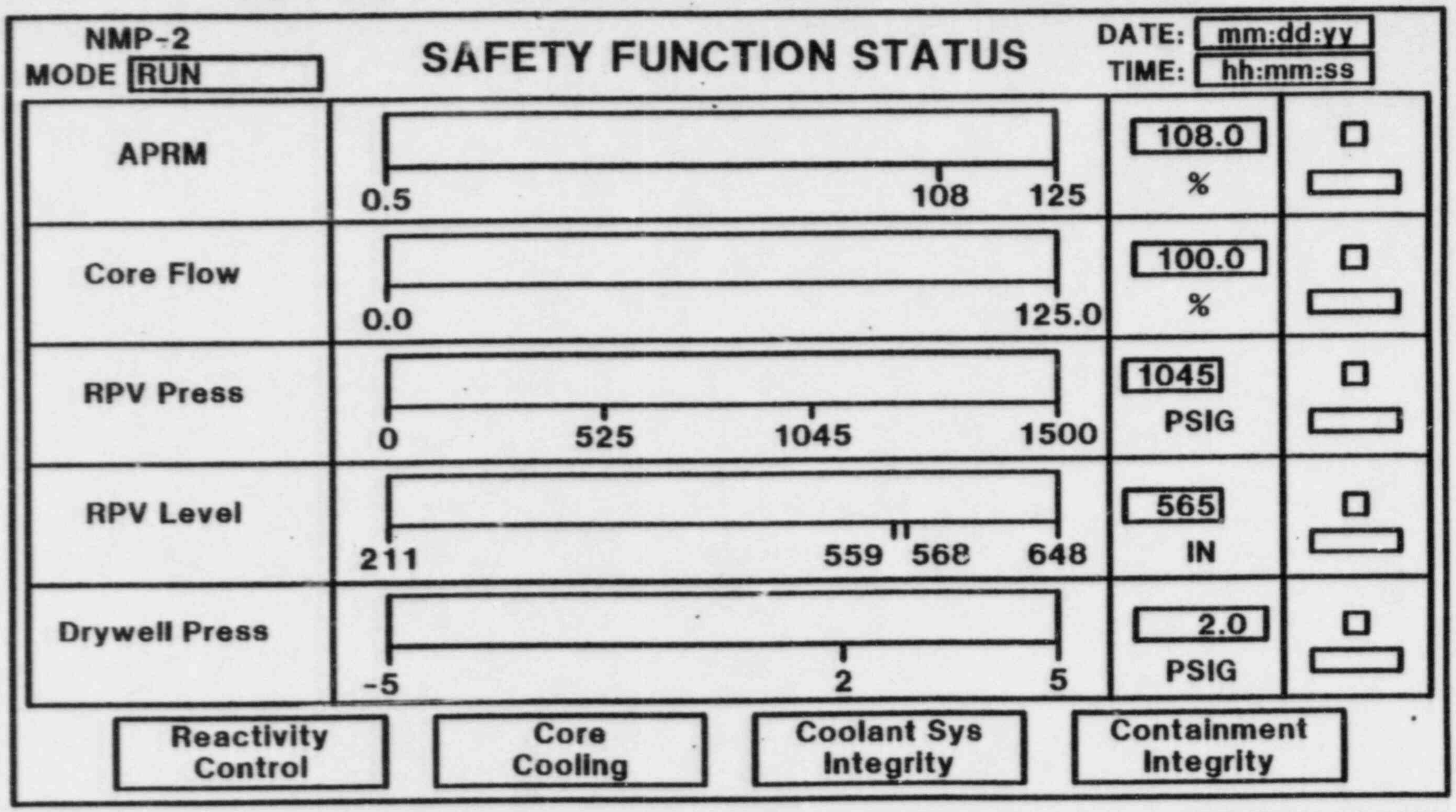
SECOND PROJECTOR SLIDES

SAFETY PARAMETER DISPLAY SYSTEM



Safety Function Status Display

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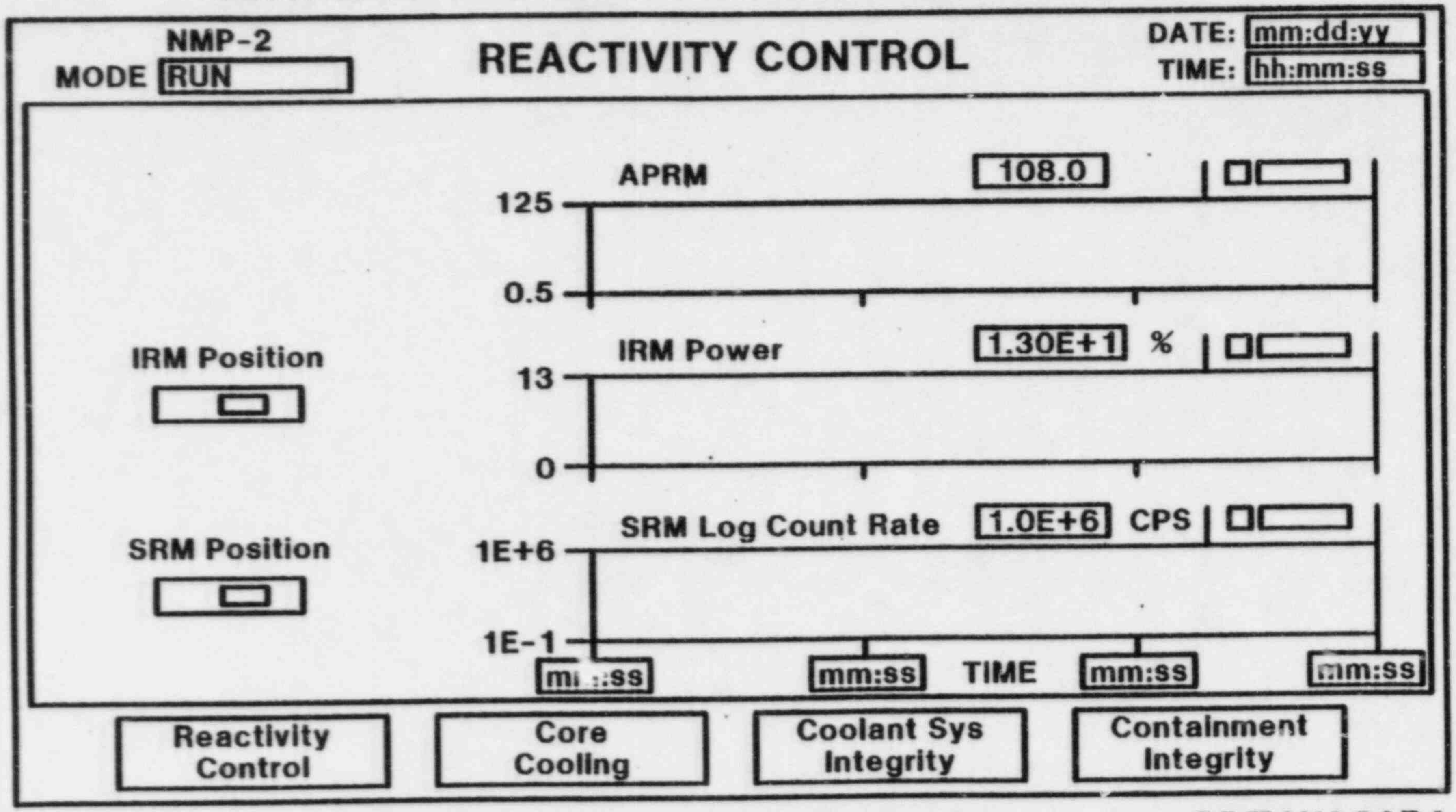
SAFETY PARAMETER DISPLAY SYSTEM



Reactivity Control Display

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REMOTE SHUTDOWN CAPABILITY

Nine Mile Point #2 Design

- Remote Shutdown Panels Provided to Meet Requirements
- Remote Panels Fulfills All Appendix R Requirements When Coupled with Some Local Operations

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REMOTE SHUTDOWN CAPABILITY

Panel Design Features

- Designed to Same Standards as Main Control Room
- Seismically Qualified
- Class 1E Redundant Instruments and Controls
- Divisional Separation per Regulatory Guide 1.75
- Redundant Controls Separated by Three Hour Fire Barriers
- Positive Pressure Maintained in Rooms

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REMOTE SHUTDOWN CAPABILITY

Physical Independence

- Two Elevations Below Main Control Room
- Electrically Separated by Keylocked Transfer Switches
- Two Safety Related HVAC Systems Physically Separated from Main Control Room HVAC System

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REMOTE SHUTDOWN CAPABILITY

Panel Layout and Systems Provided

- Service Water
- Reactor Core Isolation Cooling
- RHR-Shutdown and Suppression Pool Cooling
- SRVs - Four ADS Valves
- Remote Shutdown Room HVAC

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REMOTE SHUTDOWN CAPABILITY

Panel Layout and Systems Provided - Indication

- Totally Independent From the Control Room
- System Indication for Operability
- Critical Redundant Indication for:
  - RHR Heat Exchanger In/Out Temperature
  - Reactor Shell Flange and Bottom Head Temperature
  - Reactor Level
  - Reactor Pressure
  - Suppression Pool Water Level
  - Suppression Pool Temperature

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REMOTE SHUTDOWN CAPABILITY

Additional Features

- Plant Computer CRT/Keyboard
- Plant Wide Communications System
- Lighting Provided for All Conditions
- Panel will be Included in CRDR for Human Factors Input

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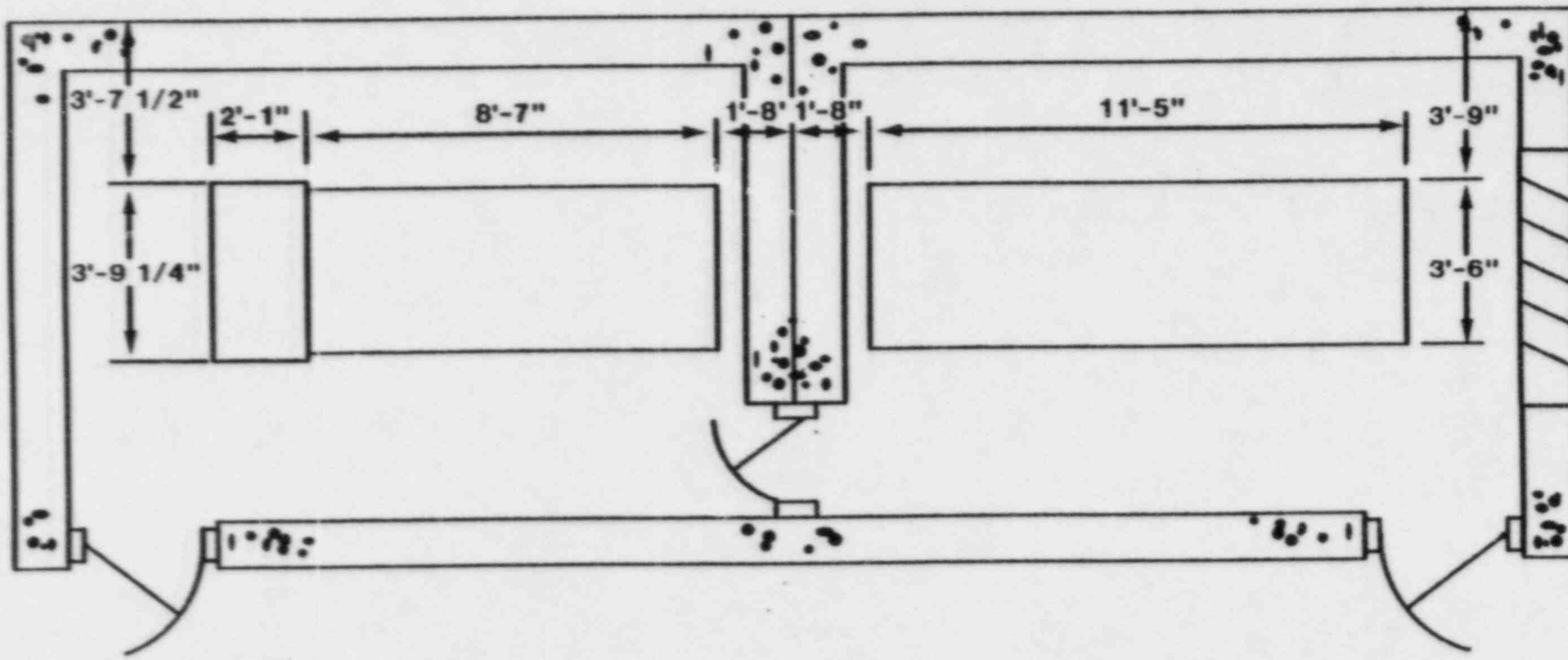
SECOND PROJECTOR SLIDES

REMOTE SHUTDOWN CAPABILITY

Remote Control Room



NIAGARA MOHAWK

NORMAN L. RADEMACHER

Nuclear Design Coordinator
Nine Mile Point Unit 2

NIAGARA MOHAWK

ROBERT RAYMOND

Supervisor
Fire Protection
Nuclear

NIAGARA MOHAWK

FIRE PROTECTION

- Fire Protection Program
- Defense in Depth
- Responsibility of Fire Protection Program
- Personnel Qualifications

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FIRE PROTECTION

Administration Controls

- Fire Brigade
- Quality Assurance

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FIRE PROTECTION

Fire Hazards Analysis

- Fire Hazards/Loading
- Safe Shutdown Analysis
- Associated Circuits

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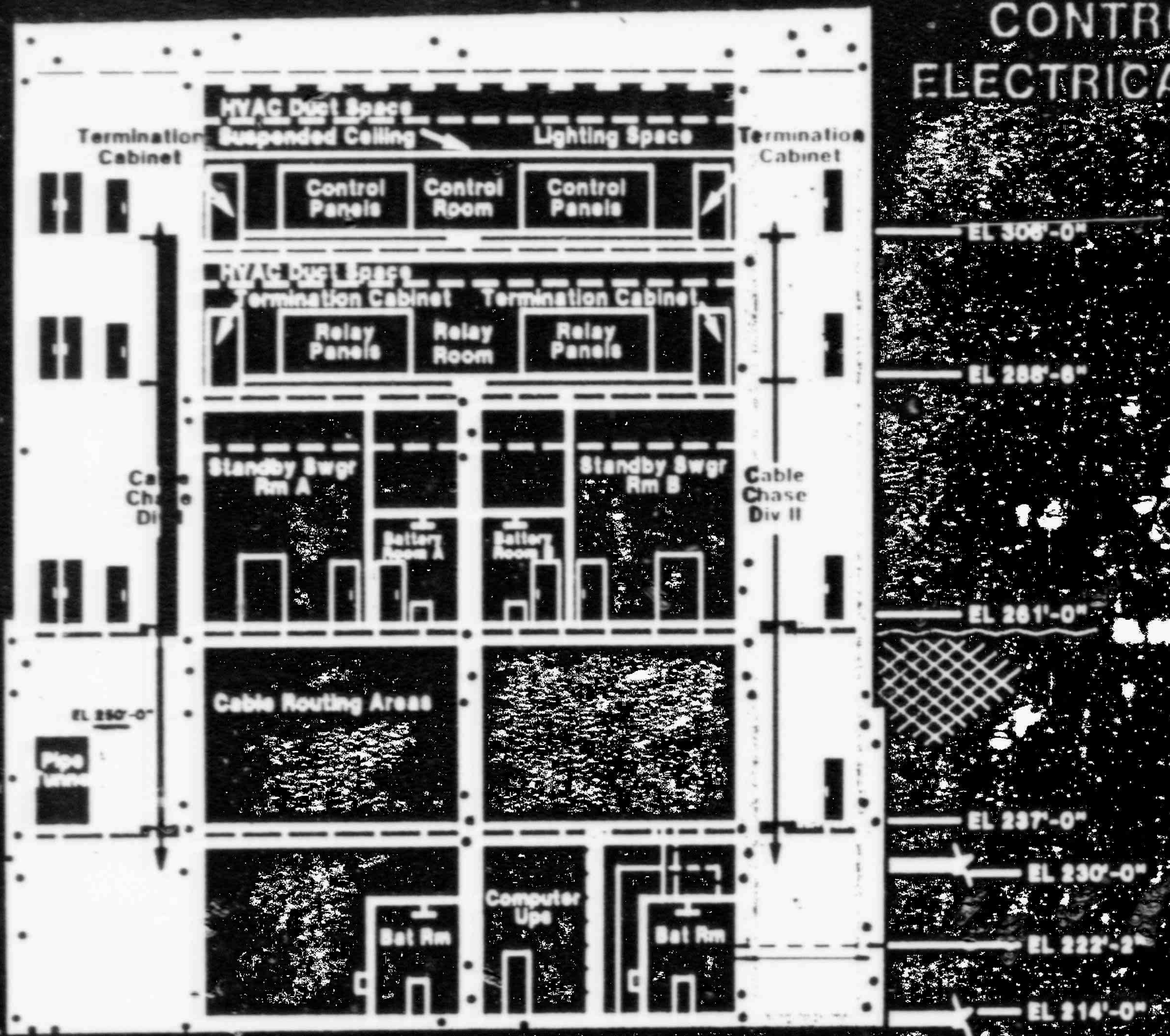
CONTROL BUILDING
ELECTRICAL SEPARATION



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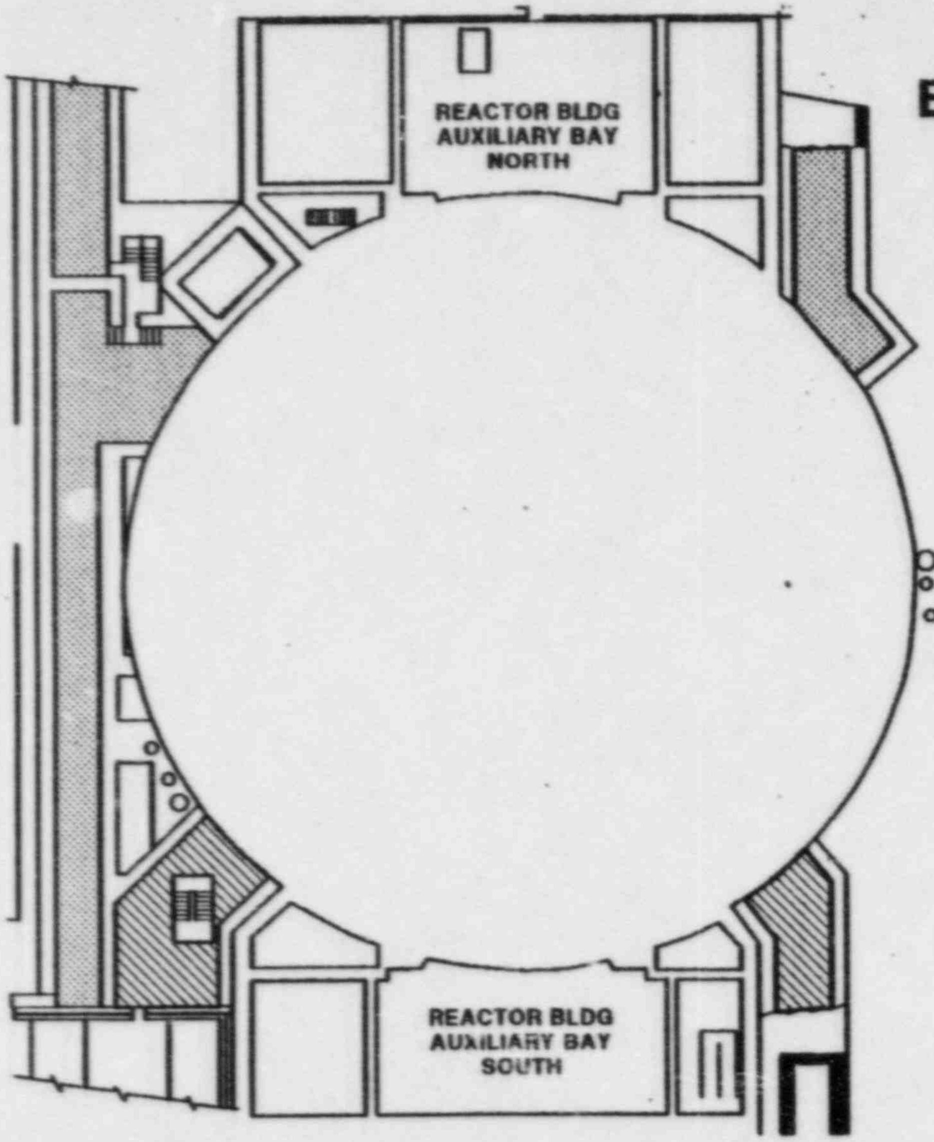
REACTOR BUILDING ELECTRICAL SEPARATION



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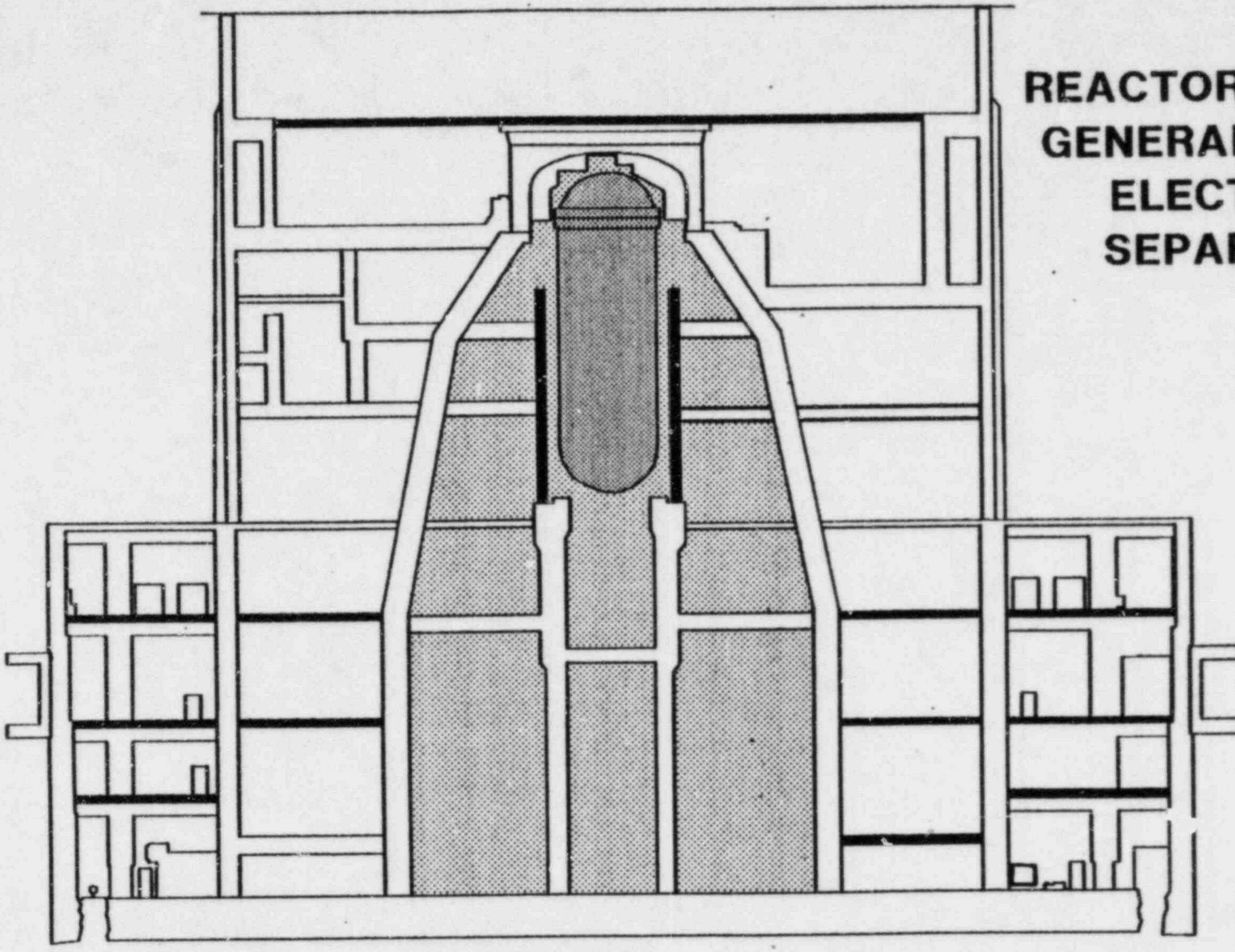

REACTOR BUILDING GENERAL LAYOUT ELECTRICAL SEPARATION

NIAGARA MOHAWK

EDWARD R. KLEIN

Assistant Manager
Project Engineering

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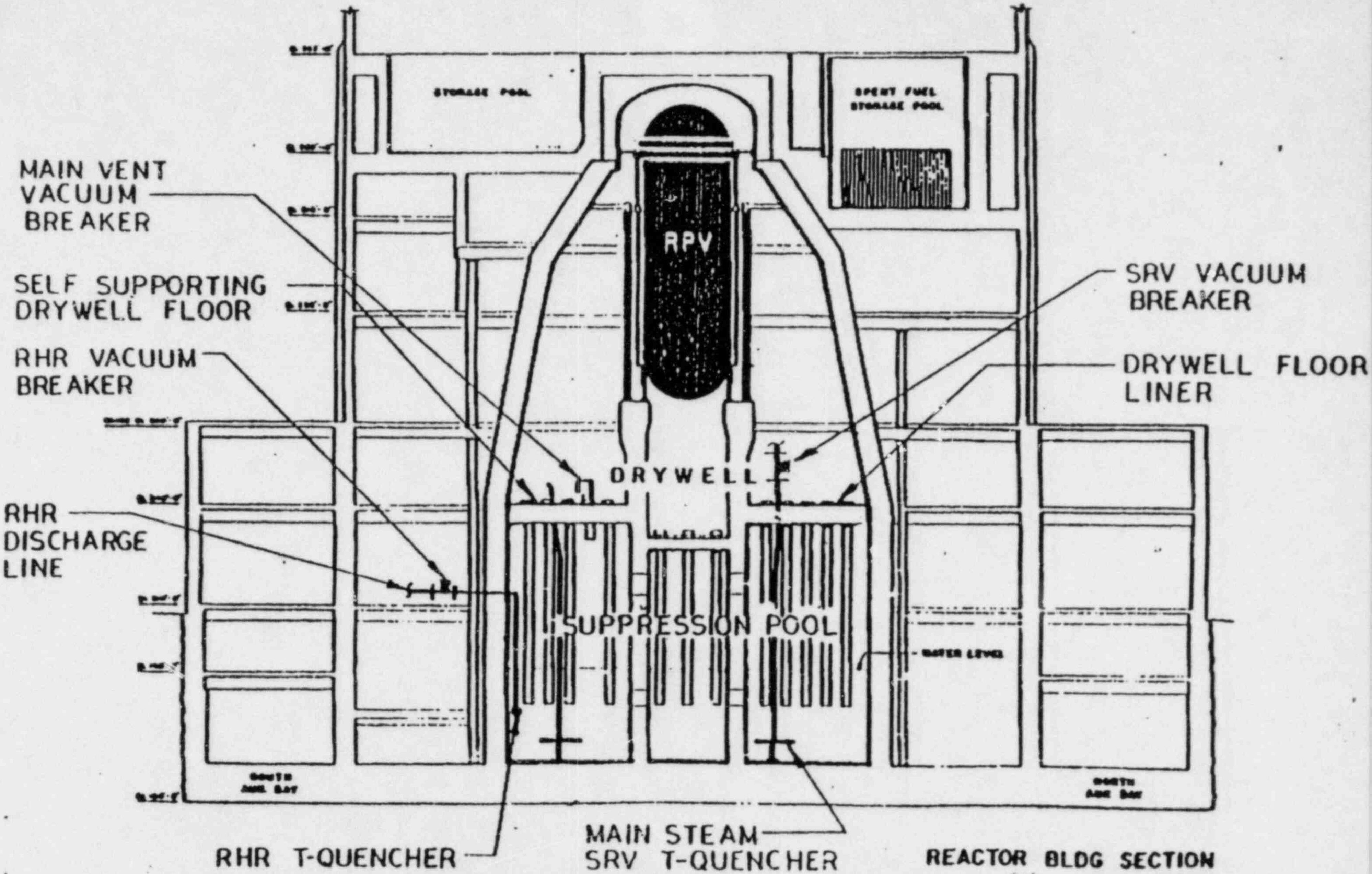
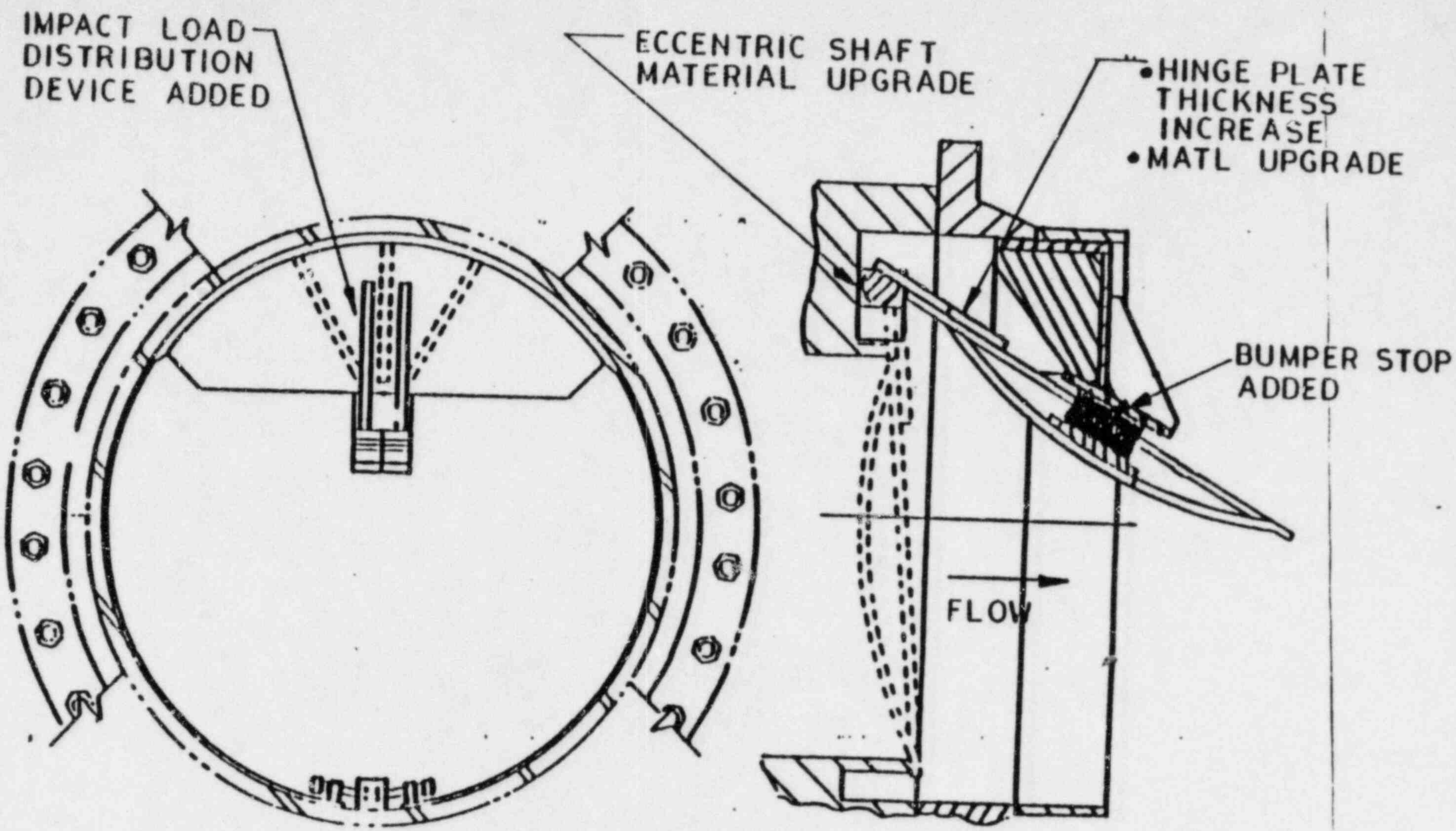
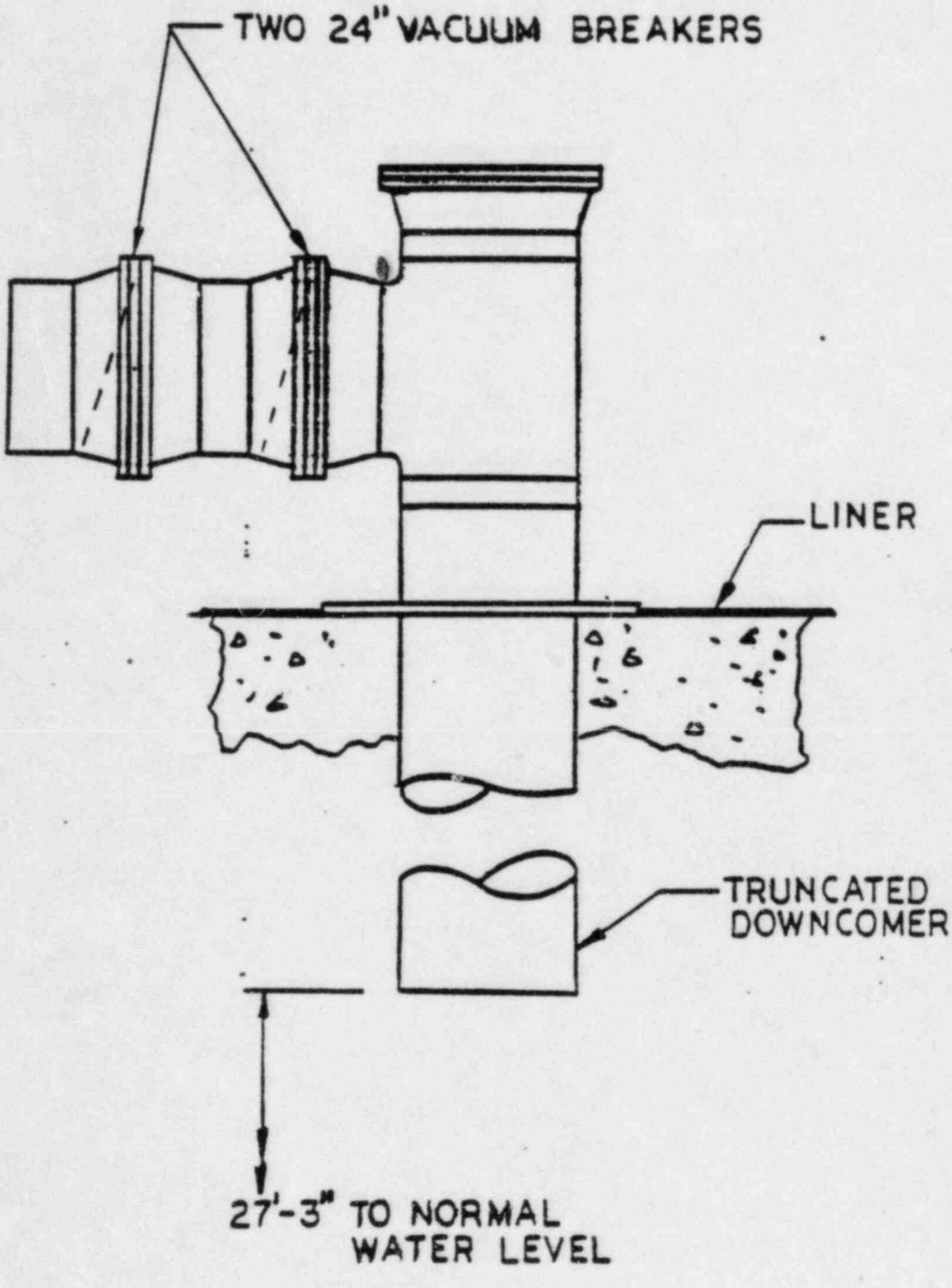


REACTOR BLDG SECTION
(LOOKING WEST)



VACUUM BREAKER MODIFICATIONS



VACUUM BREAKERS

NORMAN L. RADEMACHER

Nuclear Design Coordinator

Nine Mile Point Unit 2

NIAGARA
MOHAWK

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

- 10CFR50.62 - Alternate Rod Insertion, Recirculation Pump Trip, Automatic Standby Liquid Control
- Federal Register Volume 49 No. 124 June 26, 1984
- Reactor Trip System Reliability Assurance
- Challenges to Safety Systems

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ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Compliance to 10CFR50.62

- Unit 2 is Installing an Alternate Rod Insertion Subsystem
- Unit 2 is Installing Recirculation Pump Trip Subsystem
- Unit 2 is Installing an Automatically Initiated Standby Liquid Control System Which Provides 86 GPM Equivalent of 13 Weight Percent Boron
- Unit 2 is in Conformance with 10CFR50.62

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ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Nine Mile Point Unit 2 Conformance With Federal Register Volume 49 No. 124 June 26, 1984

Topic	Guidance	Unit 2 Design
Safety Related	Not Required Plus Existing System Cannot be Degraded	Mostly Safety Related
Redundancy	Not Required	Redundant
Diversity	To Extent Practical	Complies
Electrical Independence From RPS	Independent and Isolated	Complies
Separation From Reactor Trip System	Not Required	Complies
Environmental Qualification	For ATWS Events	Complies

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ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Unit 2 is in Conformance with NRC Guidance

Topic	Guidance	Unit 2 Design
Seismic Qualifications	Not Required	Seismic
QA	Non CAT I	CAT I
Safety Related Power Supply	Not Required	Safety Grade Power Supply
Testable at Power	Required	Complies
Inadvertent Actuator	Minimize Frequency of Inadvertent Reactor Trips and Challenges to Safety Systems	Complies

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NINE MILE POINT UNIT 2
CHALLENGES TO SAFETY SYSTEMS

A RELIABILITY ANALYSIS* FOR ATWS 3A WAS PERFORMED. THE RESULTS ARE:

- INADVERTENT RRCS SIGNALS RESULTING FROM RANDOM MODE SENSOR AND LOGIC FAILURES IS LESS THAN 1×10^{-4}/YEAR
- INADVERTENT FEEDWATER RUNBACK SIGNALS RESULTING FROM RANDOM MODE FAILURES IN SENSOR AND LOGIC FAILURES IS LESS THAN 1×10^{-4}/YEAR
- IMPACT ON OTHER SAFETY SYSTEMS RESULTING FROM 3A MODIFICATIONS IS NOT SIGNIFICANT

* NEDE 22157

ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

SCRAM Reduction

- Operating Personnel Experience
- Improved Equipment and System Design to Reduce SCRAMS
- Enhanced Quality Programs on Nonsafety-Related Equipment

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ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

CRD Hydraulic System Changes

- Materials Upgraded to Low Carbon Stainless Steel to Mitigate IGSCC Concern
- Insert and Withdraw Lines Designed Using "Fast SCRAM" Hydrodynamic Loads
- HCU Pilot SCRAM Solenoid Valves Refurbished

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ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Improvements in Component/System Design Reduce SCRAMS

- Three 50% Capacity Feedwater Strings
  - One Pump in Standby
  - Alternate Pump "C" Feeds
- Use of Rosemount Analog Trip System
- Recirc. Pump Runback on Partial Loss of Feedwater

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ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Operating Personnel Experience

- A Prime Contributor to Unnecessary SCRAMS is Inadvertent Actions by Personnel During Surveillance Testing
  - Carefully Evaluated in Preparing Surveillance Test Procedures
  - Experienced Operating Personnel
  - NMP1 Has Not Experienced a SCRAM Caused by Surveillance Test Since 1974
- NMP1 Overall Experience is 5.4 SCRAMS Per Year
  - Industry Average is 6.4/Yr/Plant
  - NMP1 Average Between 1972 and 1984 (Nov.) is 3.25 SCRAMS/Yr
  - NMP1 Average for Past Five Years is 1.4 SCRAMS/Yr

NIAGARA MOHAWK

DOUGLAS PIKE

Asst. Manager, Project Engineering
Systems Design

NIAGARA MOHAWK

INSTRUMENTATION TO DETECT INADEQUATE CORE COOLING

NRC Position

- Licensees shall have Instrumentation that Provides Unambiguous, Easy to Interpret Indication of Inadequate Core Cooling

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INSTRUMENTATION TO DETECT INADEQUATE CORE COOLING

BWR Owner' Group Activities

- Two In-Depth Studies Performed
  - Evaluation of Present BWR Level Instrumentation (SLI-8211)
  - Evaluation of ICC and Additional ICC Instrumentation (SLI-8218)
- Conclusions
  - Water Level is Conclusive Indication of ICC
  - Some Problems with Existing Systems that are Plant Specific
  - Recommendations for Improving Existing Systems
  - PRA Performed on Generic Plant Model to Put Problems and Potential Improvements Into Perspective

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INSTRUMENTATION TO DETECT INADEQUATE CORE COOLING

Unit 2 Level Measurement System

- BWR 5 Plant Specific Design
- Measure Diff. Pressure Utilizing a Filled Reference Leg and a Variable Leg
- Rosemount Analogue Transmitter and Trip System Utilized
- Redundancy and Diversity Built-In
- Jet Impingement Study Ensures HELB does not Disable Both Redundant Protective Functions
- Worst-Case Reference Leg Failure Coupled with Additional Single Instrument Failure Shows Core Remains Covered without Operator Action

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INSTRUMENTATION TO DETECT INADEQUATE CORE COOLING

Control Room Level Indication

Device	Panel	Range	Power Supply
Meter	P601	Fuel Zone	Division I
Recorder	P601	Fuel Zone	Division II
Recorder	P601	Wide Range	Division I
Recorder	P601	Wide Range	Division II
Meter	P603	Wide Range	RPS/UPS
Meter	P851	Shutdown Range	Black UPS
Recorder	P603	Upset Range	Black UPS (FWC)
Meter	P603	Narrow Range	Black UPS (FWC)
Meter	P603	Narrow Range	Black UPS (FWC)
Meter	P603	Narrow Range	Black UPS (FWC)

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INSTRUMENTATION TO DETECT INADEQUATE CORE COOLING

Instrument Ranges

- 5 Ranges Provide Level Indication from Below Core to Above Reactor Head Flange
- All Ranges Referenced to a Common 0 (Top of Upper Core Support Plate)
- Safety-Related Fuel Zone and Wide Range Indication Fed to SPDS Displays for Trending and Invalid Data Indication

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INSTRUMENTATION TO DETECT INADEQUATE CORE COOLING

Reference Leg Heating

- Uniform Heating of Reference Legs
  - Increasing Drywell Temperatures
  - Not a Concern Due to Same Vertical Drop in Reference and Variable Legs for Narrow, Wide and Fuel Zone Ranges
- Flashing and Boil-Off
  - Occurs when Vessel Pressure Drops Below Saturation Temperature of Water in Sensing Lines
  - For Pipe Breaks Inside Containment Protective Action Occurs Before Level Trips are Adversely Affected
- Emergency Procedures Provide for the Following:
  - Instructs Operator in Avoiding Situation
  - Tells Operator How to React if Situation Occurs
  - Safety-Related Drywell Temperature Monitoring System Available
  - Operator Can Utilize this System, Coupled With Reactor Pressure Indication, to Determine when Flashing and Boil-Off Conditions are Probable

NIAGARA MOHAWK

CARL D. TERRY

Manager - Nuclear Engineering

NIAGARA MOHAWK

DECAY HEAT REMOVAL

- Decay Heat Removal Systems
- NMP2 Design Enhancements

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DECAY HEAT REMOVAL

Systems

- Reactor Core Isolation Cooling (RCIC)
- Residual Heat Removal (RHR) - Multiple Modes of Operation
- High Pressure Core Spray (HPCS)
- Low Pressure Core Spray (LPCS)
- Automatic Depressurization System (ADS)

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DECAY HEAT REMOVAL

- NMP2 Provides Normal and Emergency DHR Systems Similar to Other GE BWR/5s
- Enhancements Relating to Reactor Building Design and Equipment Location
- Specified 1.15 Service Factors for RHR and LPCS Pump Motors Increase Reliability

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DECAY HEAT REMOVAL

Enlargement of Reactor Building and Primary Containment

- NMP2 Containment Design is an Enhancement in Traditional Mark II Design
- Representative of NMPC Design Philosophy
- Reactor Building Enlarged by Providing North and South Auxiliary Bays
- Relieves Congestion Typical Inside Most Facilities
- Auxiliary Bays Enhance Operability and Reliability of RHR and ECCS Equipment

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DECAY HEAT REMOVAL

Essential Core Cooling Equipment Location

- North Auxiliary Bay
  - LPCS Pump
  - RHR Pump and Heat Exchanger (Loop A)
- South Auxiliary Bay
  - RHR Pump and Heat Exchanger (Loop B)
  - RHR Pump (Loop C)
- Reactor Building
  - HPCS Pump
  - RCIC Turbine and Pump

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DECAY HEAT REMOVAL

Capability for Leak Detection Improved

- Design Prevents Loss of NPSH Due to Lowering Suppression Pool Level
- Flood Troughs Segregate Suction Line Leakage
- Leakage Collected Into a Sump
- Maximum Credible Crack Leakage Detection Possible Within 2 Minutes
- Assuming Leak Isolation Takes One and a Half Hours, Resultant Water Loss Represents Only 7 Inches of Suppression Pool Level

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DECAY HEAT REMOVAL

Enhanced Pool and Primary Containment

- Enlarged
- Improved Cleanliness of Water
- Precluded Long Term Pool Degradation

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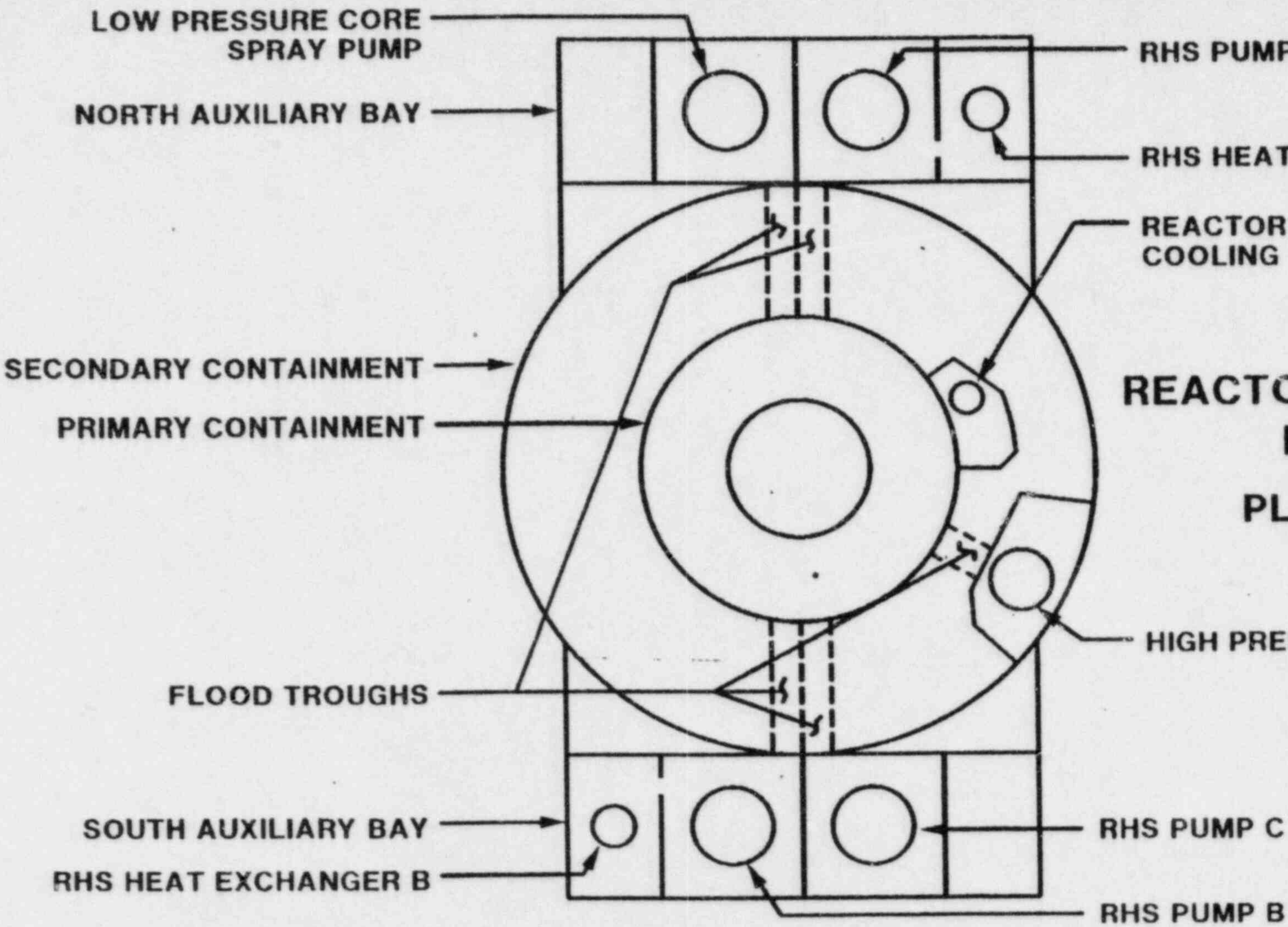
PHYSICAL SEPARATION CRITERIA FOR DECAY HEAT REMOVAL SYSTEMS
NINE MILE POINT - UNIT 2



REACTOR BUILDING - EL 175

PLAN VIEW

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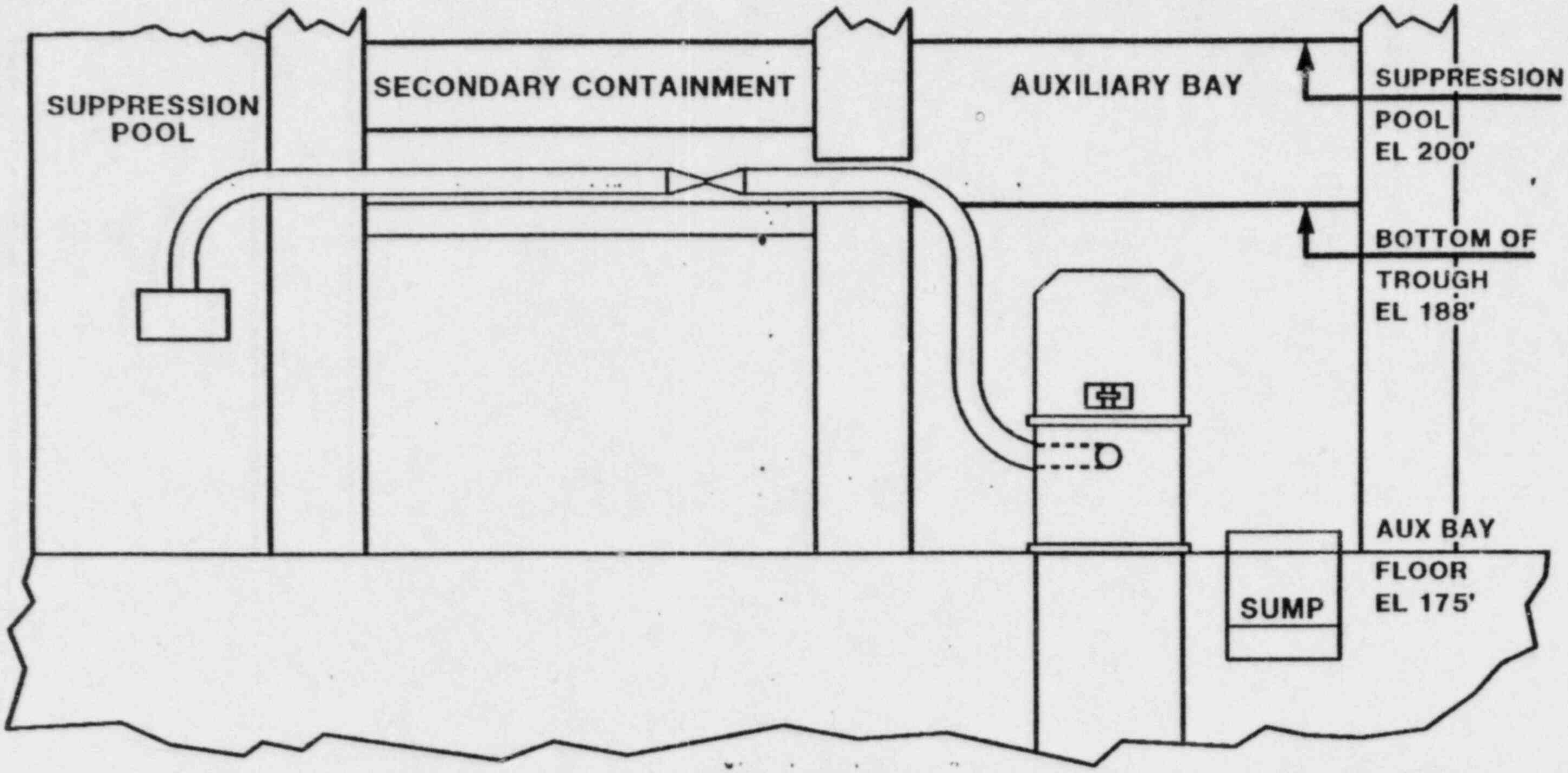
FLOOD TROUGH ELEVATION



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CARL D. TERRY

Manager - Nuclear Engineering

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SYSTEMS INTERACTION

Overview

- SI has Not Been Evaluated in Formal Study, But Considered in Virtually All Aspects of NMP2 Design
- Fundamental to Precluding SI Problems is the Established Principle of Defense in Depth
- Additional Design Features Provided for Protection Against Pipe Ruptures, Missiles, Seismic Events, Fires and Flooding
- Numerous Analyses Provide Additional Assurance that SI is Evaluated

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SYSTEMS INTERACTION

Functional Interactions

- Limited Probablistic Risk Assessment
- Failure Modes and Effects Analysis
- Evaluation of Control Systems Failures
- Fire Hazard and Safe Shutdown Analysis - Appendix R
- Excluded Equipment List
- Imposition of More Stringent Quality Programs for Procurement of Non-Safety Related Equipment - QA Categories IIA, IIB and III
- Piping Analyses where Transients are Caused by Non-Safety Related Systems Failures

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SYSTEMS INTERACTION

Types of System Interactions

- Functional Interactions Involving Interconnected Systems
- Spacial Interactions
- Human Interaction Including Man/Machine Interface and Information Interpretation

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SYSTEMS INTERACTION

Spacial Interactions

- High Energy Line Break (HELB) Evaluations (Damage Due to Pipe Whip and Spray Impact)
- Moderate Energy Line Break - Equipment Impacts Due to Exposure to Spray and Flooding
- Control System Failure Due to HELB
- Heavy Loads Evaluation
- Equipment Qualification
- Evaluation of Internally and Externally Generated Missiles
- Electrical Separation-Regulatory Guide 1.75
- Seismic Category II Over Category I

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SYSTEMS INTERACTION

Human Interactions

- Involvement of Operating Plant Personnel in Review of Design Layouts
- Human Factors Control Room Design Review
- Control Room Panel Mock Ups During Conceptual Design

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SYSTEMS INTERACTION

Implementation and Verification

- Design Review Testing and Inspection Programs
- Multi-Discipline Review of Design Documents and Independent Design Review
- Preoperational Testing
- Thermal Growth/Vibration Monitoring
- Walkdowns
- Ongoing Reviews/Evaluations
- A/E Engineering Assurance Program

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DOUGLAS PIKE

Asst. Manager, Project Engineering
Systems Design

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OFFSITE POWER SUPPLY SYSTEM

Nine Mile Design

- Unit 2 Provided with 2 Independent 115KV Power Sources

Grid System

- Tied to New York State Grid System
- State Grid System Tied Via 26 Interties to:
    - New England Grid
    - PJM Grid
    - Canada

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OFFSITE POWER SUPPLY SYSTEM

Scriba Substation

- 3000 Feet South of Station
- Fed from 5 Separate 345 KV Lines
  - Any One Feed Can Power All Station Loads
- Breaker and a Half Scheme for Reliability
- Two 115KV Feeds through Separate Transformers and Circuit Breakers

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OFFSITE POWER SUPPLY SYSTEM

115KV Transmission Lines and Switchyard

- Lines Separately Routed to the 115KV Switchyard
- Lines Fault Protected by Primary and Back-Up Schemes Fed From Separate Station Batteries
- Switchyard is Segregated by MODs and Circuit Switchers to Maintain Separation of Feeds
- MODs and Circuit Switchers Interlocked to Prevent Paralleling of Offsite Sources
- Offsite Source a Normally Feeds RSS Transformer A and the Auxiliary Boiler Transformer
- Offsite Source B Feeds RSS Transformer B

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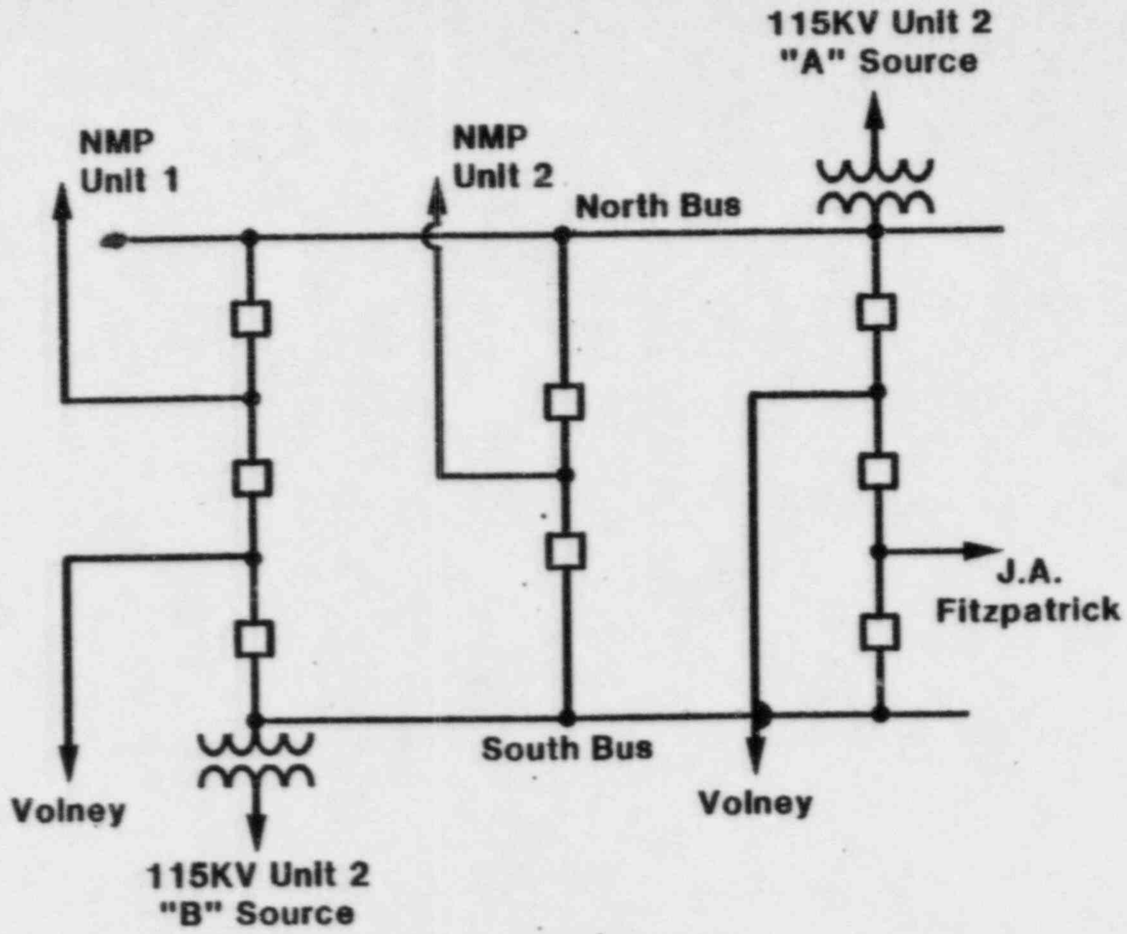
SECOND PROJECTOR SLIDES

(PHOTOGRAPHS ARE NOT INCLUDED)

OFFSITE POWER SUPPLY SYSTEM



Scriba Substation
115KV Offsite Supply Unit 2

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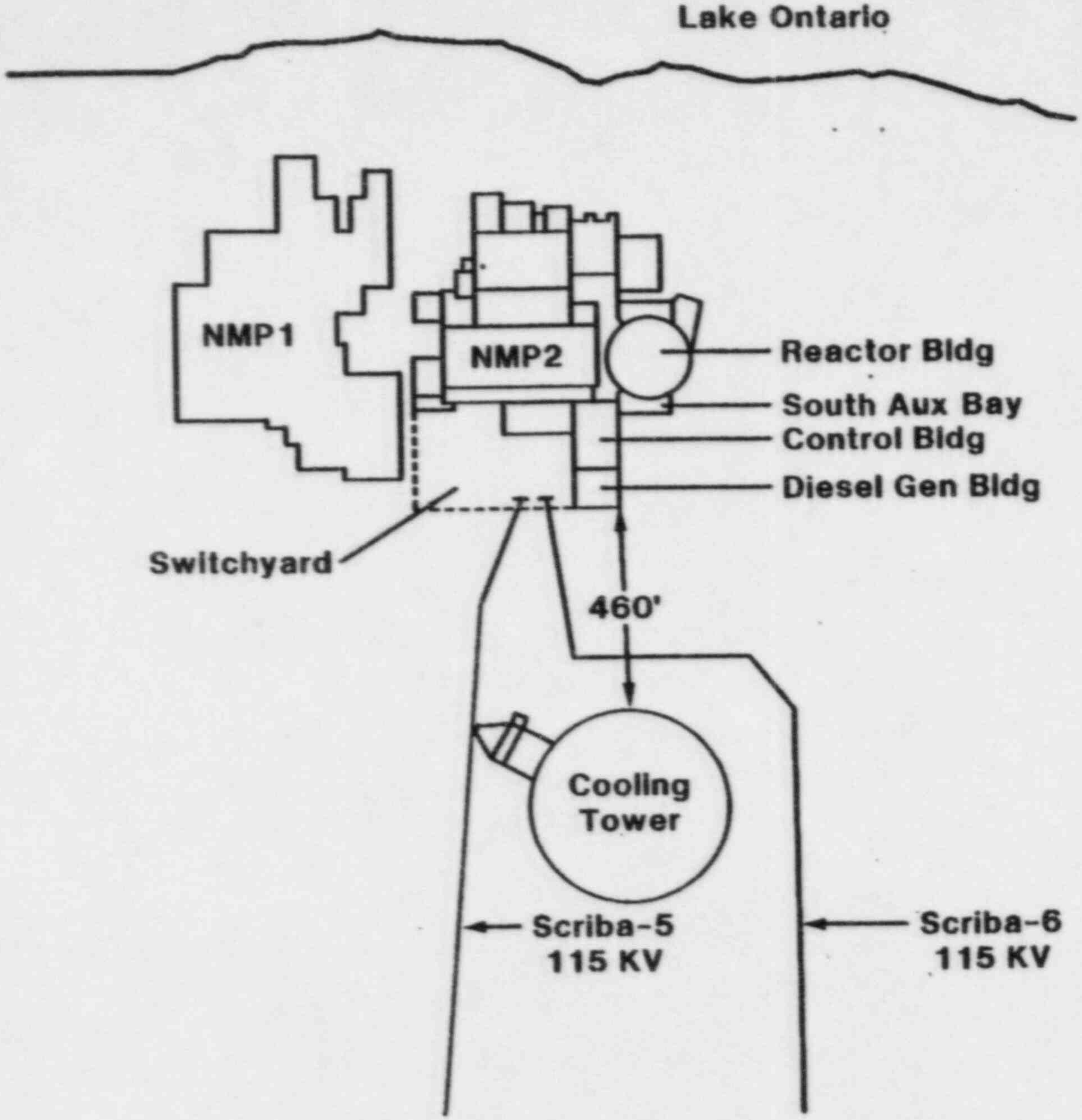
OVERALL POWER SUPPLY SYSTEM



115KV Offsite
Power Lines

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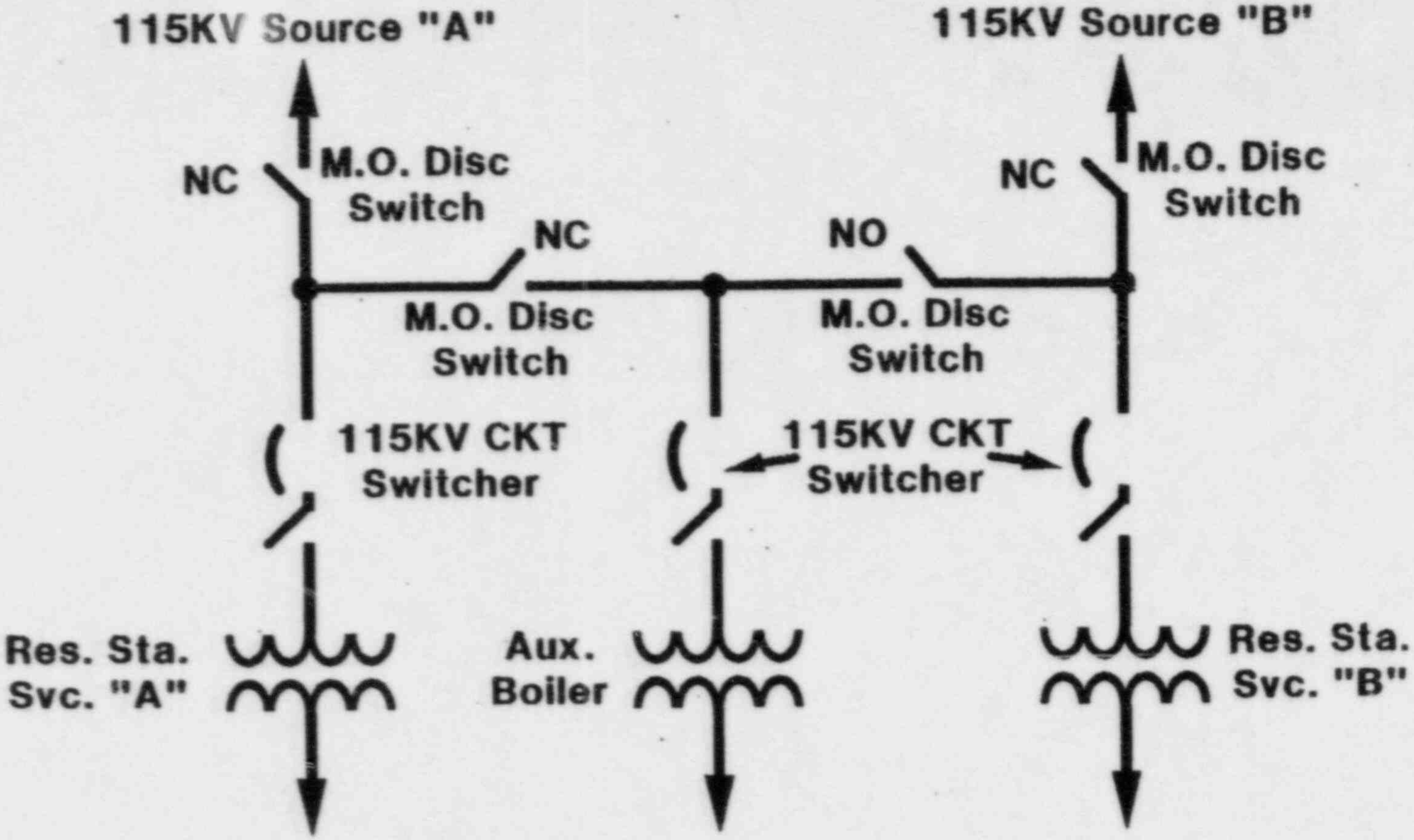
OFFSITE POWER SUPPLY SYSTEM



One Line Diagram
115KV Switchyard

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ONSITE AC POWER SYSTEMS

Non-Safety Related Systems

- Offsite Power Source A Feeds One 13.8KV Bus and Auxiliary Boiler Bus
- Offsite Power Source B Feeds the Other 13.8KV Bus
- Each 13.8KV Bus Can be Fed from Either Offsite Source
- Auxiliary Boiler Bus Can be Fed from Either Offsite Source
- 13.8KV Busses Fed from Unit Generator During Station Operation
  - Fast Transfer to Offsite Supply
- Power Distributed from 13.8KV Busses to:
  - 4160V Switchgear
  - 600V Load Centers
  - 600V Motor Control Centers

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ONSITE DC POWER SYSTEMS

- Non-Safety Related Power Supplies
  - Four Batteries and Battery Chargers for 24V DC Neutron Monitors
  - Two Batteries and Battery Chargers for 125V DC Station Loads
  - One Battery and Battery Charger to Feed Main Plant Computer Ups
- Battery Charges Can Feed All Non-Ups Loads and Recharge Batteries within 24 Hours
- Batteries Can Supply Load Profiles for 2 Hours with Loss of Chargers

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ONSITE DC POWER SYSTEMS

- Safety-Related Power Supplies
  - Three Divisions of DC Power Corresponding to AC Power Divisions
    - Class IE Equipment
    - Seismically and Environmentally Qualified
    - Physically and Electrically Separated
- Each Division has its Own Battery and Two 100% Redundant Battery Chargers
- Each Battery Charger Can Supply All Non-Ups Loads and Recharge Battery Within 24 Hours
- Each Battery Can Supply Worst-Case DC Load Profile for 2 Hours with Loss of Battery Chargers

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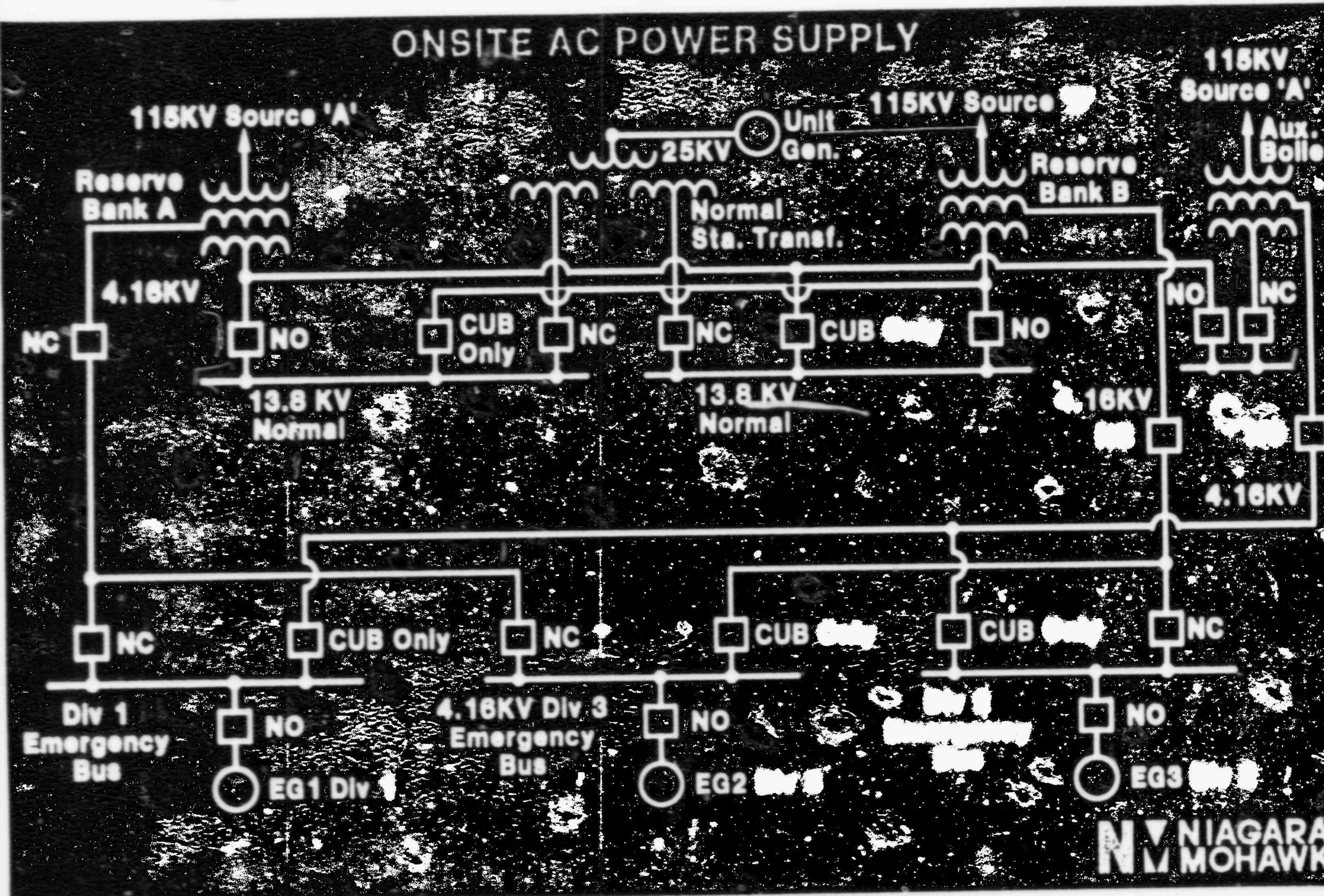
SECOND PROJECTOR SLIDES


ONSITE AC POWER SUPPLY
115KV Source 'A'
Reserve Bank A
4.16KV
NC
NO
13.8 KV Normal
CUB Only
NC
25KV
Unit Gen.
Normal Sta. Transf.
NC
CUB
NO
13.8 KV Normal
115KV Source
Reserve Bank B
115KV Source 'A'
Aux. Boiler
NO
NC
16KV
4.16KV
NC
CUB Only
NC
CUB
CUB
NC
Div 1 Emergency Bus
NO
EG1 Div
4.16KV Div 3 Emergency Bus
NO
EG2
NO
EG3
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ONSITE AC POWER SYSTEMS

Safety Related AC Power Systems (Continued)

- Divisions I, II and III 4160V Busses Feed Additional 600V and 120V Distribution Systems
- Division I and II Load Centers Can be Supplied Through Two Redundant 100% Capacity Feeders
- Divisions I and II Uninterruptible 120V Power Supplies Provided for Critical Instrument, Control and Lighting Circuits
    - Fed from Two AC and One DC Source
- Division I and II 4160V Busses Can Feed Non-Safety Related Stub Busses if No LOCA Signal is Present

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ONSITE AC POWER SYSTEMS

Safety Related AC Power Systems

- Three Independent Divisions of Safety Related Power
- Each Division Has Dedicated 4160V Bus
  - Offsite Source A Normally Feeds Division I and Division III
  - Offsite Source B Normally Feeds Division II
  - Offsite Source B Can Provide Back-up Feed to Division III
  - Auxiliary Boiler Transformer Can Provide Back-up Feed to Division I or Division II
- Each Division Has Diesel Generator to Provide Power Under Loss of Offsite Power or Degraded Voltage Conditions

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ONSITE AC POWER SYSTEMS

Reliability Features

- Main and Tie Breaker Control Circuits Fed from One Station DC Battery
- Feeder Breaker Control Circuits Fed from Separate Station Battery
- Two Station Battery Feeds Can be Interchanged through Manual Switching
- Most Busses Sectionalized so that Either Source Can Feed the Bus
- Seven Interruptible 120V AC Power Supplies
- UPS Power Feeds from Three Separate Sources

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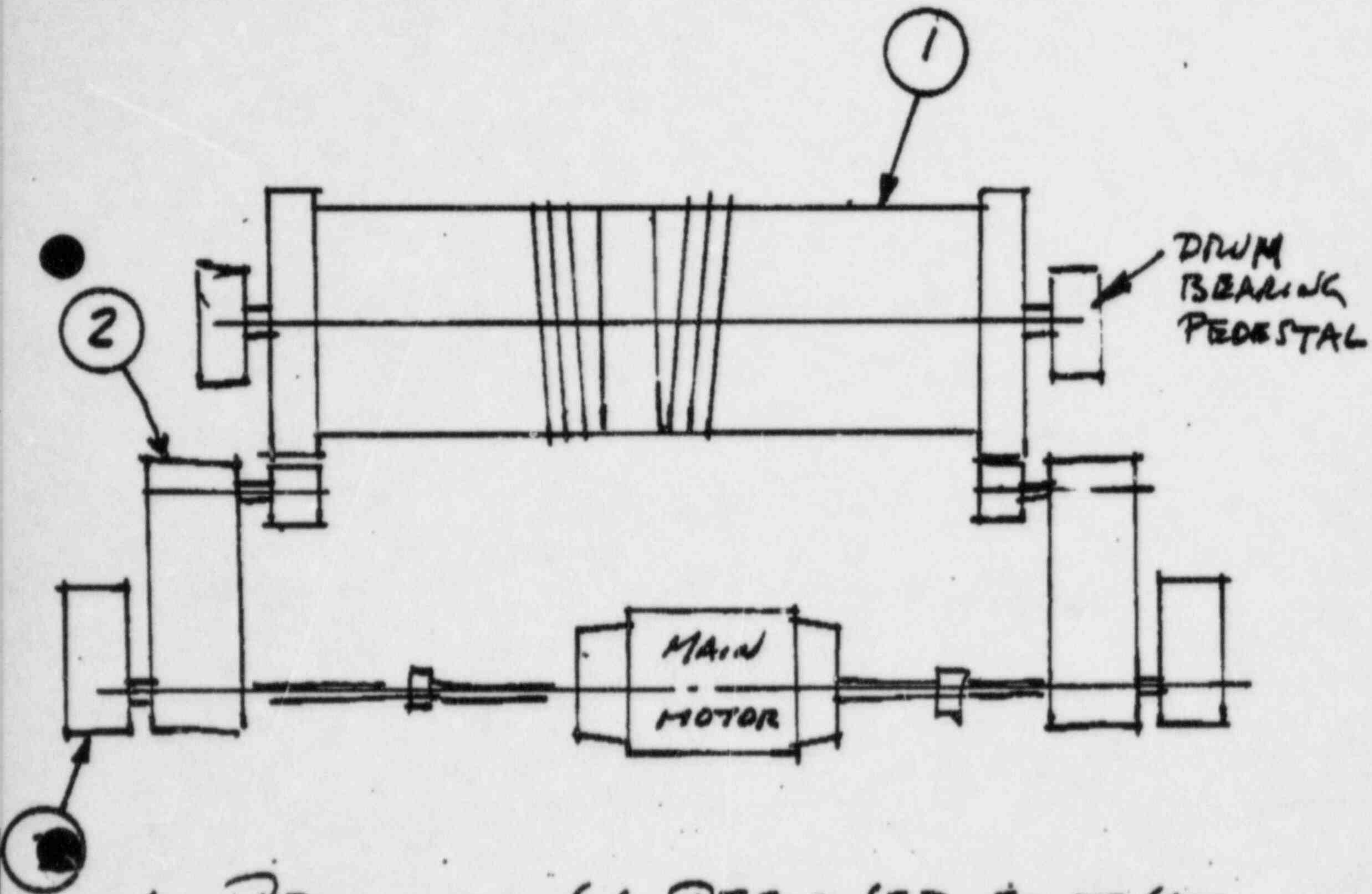
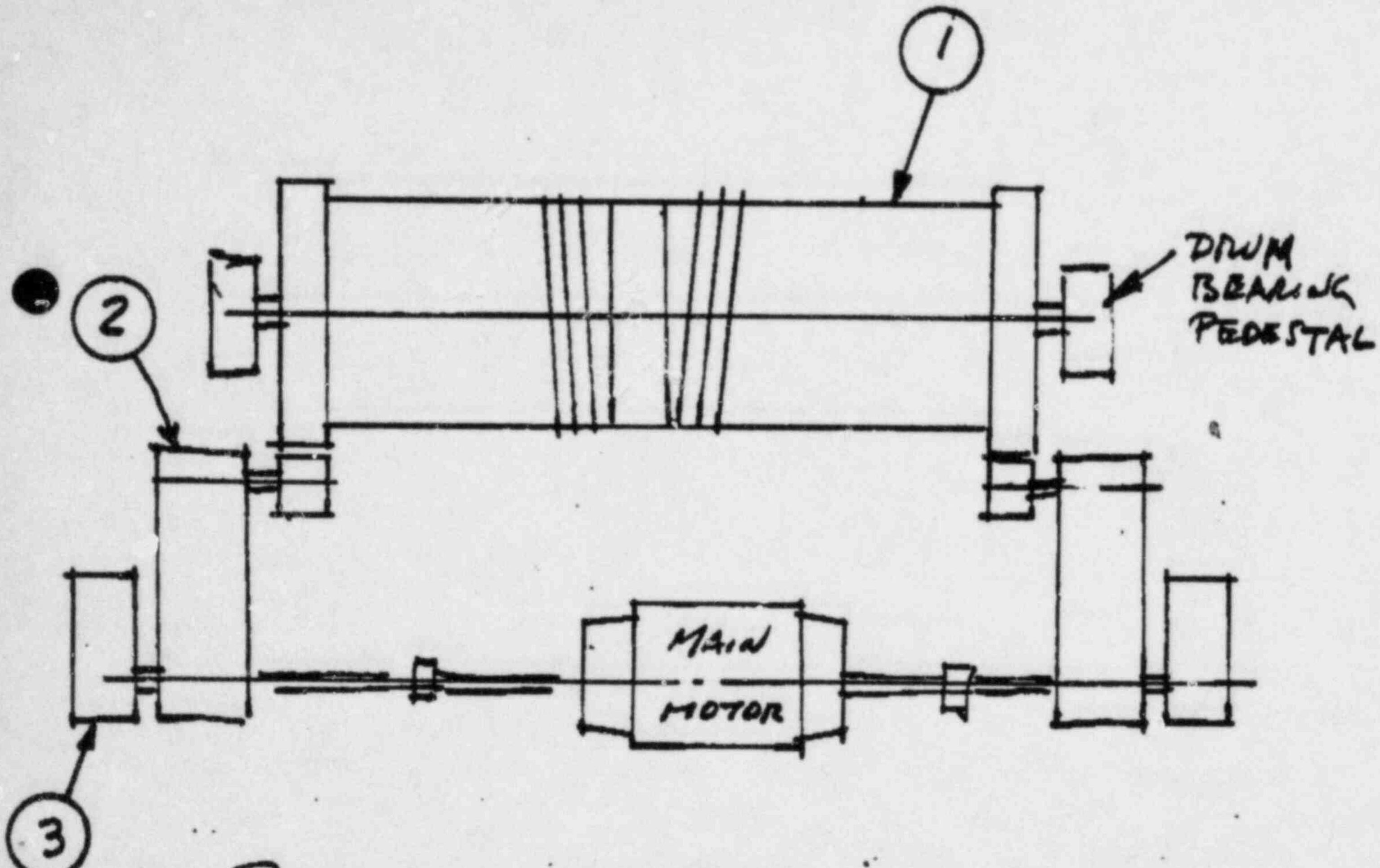


1. DRUM, MAIN: DESIGNED SO THAT UPON FAILURE OF DRUM SHAFT OR BEARING THE DRUM GEAR WILL REMAIN ENGAGED

2. REDUNDANT GEAR CASES
3. HOLDING BRAKES
4. REEVING SYSTEM
5. LOAD BLOCK
6. HOOK
7 LIFTING RIG



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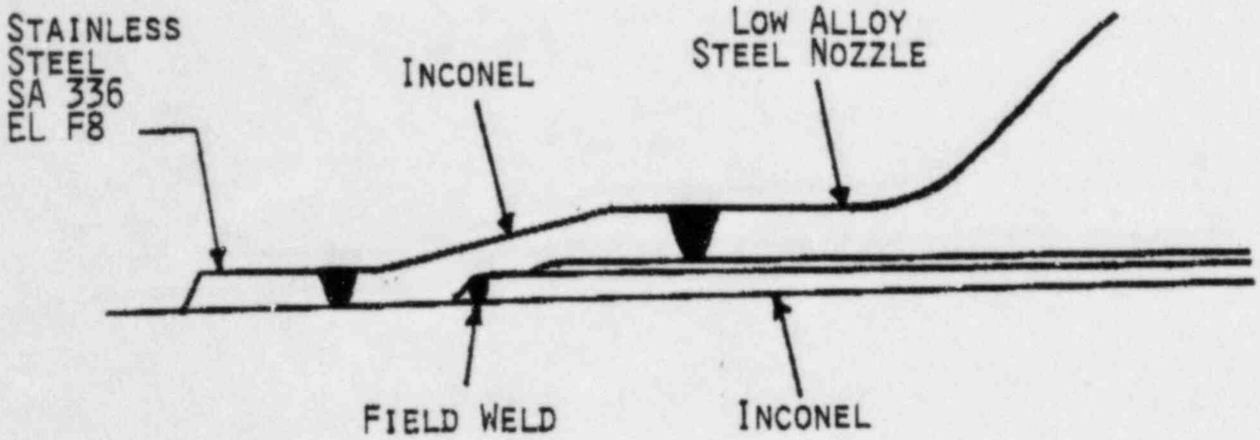
RECIRC INLET NOZZLES



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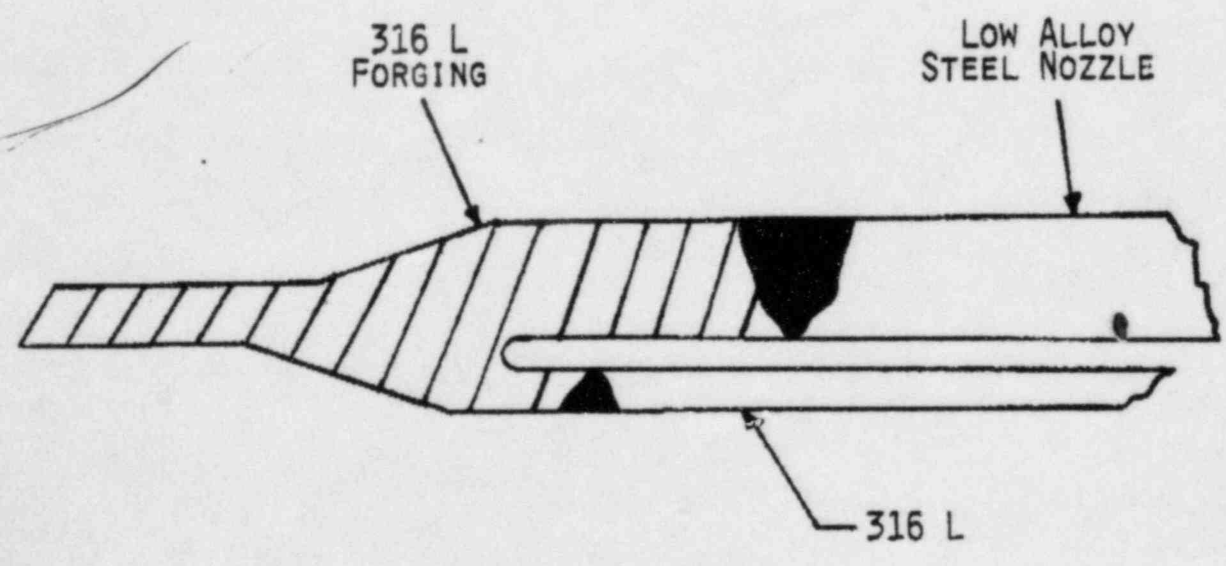

REPLACEMENT DESIGN

FOR NINE MILE - UNIT 2

Ws. ST. ~~8~~1.20 PM. a Simons

End 5 PM.

Begin Hand to RF NMPC only For Slides

February 19,1985

PROPOSED SCHEDULE FOR THE FEBRUARY 20-21, 1985
SITE VISIT AND MEETING OF THE ACRS SUBCOMMITTEE ON
NINE MILE POINT NUCLEAR STATION, UNIT NO.2
FOR OPERATING LICENSE REVIEW

Wednesday: February 20, 1985

6:15am		Buffet Breakfast	
		<u>Site Visit - Plant Tour</u>	
7:00	1.	Leave Hotel Syracuse for Plant Site a. Introduction and Orientation (on bus) b. Plant description (Principal Design Features)	60 min
8:00	2.	Plant Tour (form 2 groups)	180 min
11:00	3.	Leave Plant Site for Hotel Syracuse	60 min
12:00		- Lunch -	75 min
		<u>Subcommittee Meeting - OL Review</u>	
1:15pm		Opening Statements.	15 min
		NRC Staff Presentation	
1:30	4.	Open Items a. In Particular - Snow Loads Separation Criteria Safe and Alternate Shutdown Essential Lighting Air Start System	60 min
2:30	5.	Regional Evaluation of Construction Issues	15 min
2:45		- Break -	15 min
		Utility Staff Presentation	
3:00	6.	Management Philosophy	15 min
3:15	7.	Organization and Management a. Corporate Organization b. Nuclear Quality Assurance Organization	45 min

4:00	8.	Safety Review Committees a. Open Item - Operations Management	25 min
4:25	9.	Industry Interactions	5 min
4:30	10.	Operations Staffing and Training a. Simulator b. Experience with Fitzpatrick and NMP-1	30 min
5:00		- Recess -	

Thursday: February 21, 1985

Subcommittee Meeting - OL Review

8:30am	11.	Emergency Operating procedures	30 min
9:00	12.	Seismic Design and Geology a. Seismic Margins b. Liquefaction of Dikes	15 min
9:15	13.	AC/DC Power Systems Reliability a. On-site AC and DC Power Systems b. Off-site Power Supply System	20 min
9:35	14.	Systems Interactions	30 min
10:05		- Break -	15 min
10:20	15.	Decay Heat Removal	40 min
11:00	16.	Mark II Containment - Unique Features a. Open Items - Steam Bypass of Suppression Pool Secondary Containment Bypass Leakage Containment Isolation Containment Leak Testing	40 min
11:40	17.	Instrumentation for Detecting Inadequate Core Cooling	20 min
12:00	18.	Anticipated Transient Without Scram a. Scram Systems b. Scram Reduction Efforts	30 min
12:30		- Lunch -	60 min
1:30pm	19.	Fire Protection	15 min

1:45	20. Control Room a. Control Room Design Review - Description of Power Generation Control Complex b. Safety Parameter Display System c. Control Room Habitability d. Remote Shutdown Capability	30 min
2:15	21. Emergency Planning	15 min
2:30	- Break -	15 min
2:45	22. NMP-2 Probabilistic Risk Assessment a. Scope of Study b. What was learned c. How will the PRA be used	30 min
3:15	23. Intergranular Stress Corrosion Cracking	15 min
3:30	24. Environmental Qualification of Equipment	15 min
3:45	25. Radiation Protection Program	15 min
4:00	26. Plant Security (<u>this portion of the meeting may be closed</u>) a. Overview of Physical Security Plan b. Design Features Incorporated to Prevent Sabotage	45 min
4:45	Summary Comments: C.P. Siess a. Outline NMP-2 Schedule for the March ACRS Meeting	15 min
5:00	- Adjourn -	

(Sw) MARY Hawee
-AUGHEY
Bob GRA[illegible]M
Sam Collins
SLIDES) get set)
CAT (~~CAD~~ Inspection
TOM MURLEY