NUCLEAR REGULATORY COMMISSION

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268TH GENERAL MEETING

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UNITED STATES NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

268TH GENERAL MEETING

Room 1406

1717 H Street, N.W.

Washington, D.C.

Thursday, August 12, 1982

The Committee met, pursuant to notice, at 8:40 a.m.

ACRS MEMBERS PRESENT:

P. SHEWMON, Chairman

J. C. MARK

M. PLESSET

C. SIESS

D. MOELLER

M. BENDER

W. KERR

H. ETHERINGTON

D. WARD

J. EBERSOLE

H. LEWIS

OKRENT

DESIGNATED FEDERAL EMPLOYEE:

R. FRALEY

MR. SHEWMON: The meeting will be in order. This is the first day of the 268th meeting of the Advisory Committee on Reactor Safeguards. During today's meeting the committee will hear reports on and discuss the following:

(1) Grand Gulf Nuclear Station

(2) Proposed revision of 10 CFR 50.46, Appendix K, ECCS evaluation models

(3) Proposed NRC nuclear plant severe accident research plan (NUREG-0900) and relating rulemaking

(4) Nuclear power plant control room habitability

(5) Proposed ACRS reports to NRC regarding Grand Gulf and Ginna Nuclear Power Plants

(6) Foreign LWR licensing practices

(7) Activities of ACRS members

The items scheduled for discussion on Friday and Saturday are listed in the schedule for this meeting which is posted on the bulletin board at the back door of this meeting room.

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

Portions of this meeting will be closed as necessary to discuss proprietary or otherwise privileged information.

Mr. Ray Fraley is the Designated Federal Employee for this portion of the meeting.

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

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

The first item on today's schedule is the ACRS Chairman's report.

(Not reported.)

MR. SHEWMON: We will now to the report from the Subcommittee on Grand Gulf.

MR. OKRENT: You have in front of you a sheet of paper which tells you what the agenda for the briefing is.

Let me refresh your recollection. You will recall that back in October, we did an interim review of Grand Gulf, which we completed action on most issues. At that time, the staff had not completed its review, either on a final or interim basis, of the new proposed system, and there was still at least one open question regarding hydro-dynamic loads on the structures and components above the suppression pool. We identified those two matters, things that we would review later, in our letter of last October.

Also in our letter of last October, we mentioned in particular the importance of adequate operating experience with the PWRs, and the necessary outside representation, and things of that sort.

Since that meeting, where we wrote that letter, two or three things of particular interest have occurred. One is that there was a letter or a report made available by a former employee of General Electric who raised several detailed kinds of questions concerning things that go in, on and around the suppression pool, and we will go back to that item. Also, the utility has lost the services of their system plant manager, thereby losing a large amount of PWR experience, and I think this is something that we want to think about.

At the subcommittee meeting, which was held yesterday, the matters we dealt with were hydrogen control, management structure, technical capability, and questions concerning single failure criteria that Jesse Ebersole was interested in. Since it was relevant to Grand Gulf, we discussed that in some detail. There was also some review of quality assurance and quality control, we had reports from I&E on that.

The matter of hydro-dynamic loads that was identified in our earlier review by the staff as not being finished, and the more recent questions related to suppression pool issues were reviewed at separate subcommittee meetings held by Dr. Plesset, and after I am finished, he is going to give us a report on these.

In summary, the feeling of the subcommittee is that these are in acceptable shape, and that is why you don't see a long discussion on these issues on the agenda. We have accepted the report of that subcommittee, and we structured the agenda that way.

Now, let's look at the agenda for a minute and see what is proposed in here. As I indicated, hydrogen control was an issue outstanding at the previous subcommittee meeting.

The staff has concluded in this regard that on an interim basis what is being proposed is okay. They expect to complete their final review in a year or so, they tell us. There have been separate reviews made of what is proposed by Mississippi Power and Light in this regard.

In addition to the staff, they have Sandia looking at this matter. Sandia has raised some questions concerning whether the number of ignitors might be improved. In fact, the utility engaged the services of an outside consultant to look at this, and this consultant suggested some ignitors at lower elevations. This matter is not closed, I think, either in the utility's mind or in the staff's mind. You can ask about it. I don't know that it is a vital issue on an interim basis, and it can be resolved on a final basis.

You also see some discussion proposed on the available PWR experience now in the operating organization, also within the utility itself, back at the ranch as it were. We thought we wanted to hear about this, so that is on the agenda.

It seemed to me that Jesse Ebersole was reasonably satisfied with the original question about the sensor system as was posed. However, as we all know, he had lots of questions.

In fact, he raised a question that is not new, a question that has been brought out many times in the past, which in fact the Applicant has been on notice might be brought up in the subcommittee meeting, where there was some discussion of it, and that is the following:

Within the drywell, you have pipes that might rupture, or have big leaks, or something, primary system pipes, maybe, and you also have some lines that run down to the control rod chambers, hydraulic lines for actuation of the control rods. There is a technical position that has provided an acceptable approach as far as the staff is concerned with regard to deciding where in the piping system you have to provide for jet effects, and so forth, at one of the highest stress calculated positions of piping.

My understanding is that the Applicant has met that position. There may be other positions where failure of the pipe in one manner or another could lead to a loss of function of several or many of these hydraulic lines.

It is not completely clear that enough lines are involved that you could expect to lose ground, but if you pick the right rods, it doesn't take too many of these to make it difficult to shut down the reactor, especially with cold water.

The concern that Mr. Ebersole has, if I understand it correctly, is there in principle the possibility that failure might lead to not only a LOCA, but the ability to shut down the reactor, and it goes critical. The boron doesn't help you very much because it is not designed for that kind of thing.

So, as I said, we had some discussion on this yesterday. We didn't try to conclude it. We thought it would be useful to try to have some additional discussion today, and we have asked the staff and the Applicant to give us a short summary. As it was pointed out yesterday, this is not a question unique for Grand Gulf.

On the other hand, it may apply to Grand Gulf, so it seemed relevant to at least talk about it, but I doubt that you will be able to talk about it in excrutiating details, at least not with the time I have shown on the agenda.

The only other item you see called out is something that arose in a paragraph in the SER which said that the Applicant had proposed venting at the design pressure of 15 psig. I, myself, at that time had not seen the letters from Grand Gulf, which discussed this.

I have just been handed these copies, and I guess there are copies for everybody. Anyway, I thought we ought to hear a little bit about it today, so that is why you see that agenda item.

That is all I propose as a subcommittee report, Mr. Chairman. Mr. Bender, Mr. Ebersole, Mr. Plesset, who is going to report in a minute, were there, as well as Mr. Mark.

Is there anything that the subcommittee members want to add?

MR. EBERSOLE: I might make a comment about this. When you hypothesize and situation like this, which has the potential for disaster, which is what this does, it is, at least in my view, undesirable to create a recognized complicated system of dependency which must be erected to keep the accident from cascading to its extreme consequences.

It is much better to reexamine the design and, in essence, go around the problem, not through the myriad of detail that you must dredge up and argue that all these things would work to keep from progressing to its final stage.

You will find in the discussion here that, in fact, there are quite a few details to show that this thing will not so progress. In own view, a better design is to create the geometry and configuration which prohibits this sort of argument to be erected in the first place.

It is not impossible to postulate what has been suggested, but the likelihood of a combination of things happening is quite low. The technical position is that where the stresses in the lines are low, the likelihood of rupture is small and will not necessary strain the pipes.

The kinds of breaks that might cause trouble are splits in high pressure piping that have to impose a very strong jet on the control rod drive hydraulic lines, which themselves are very strong, and the problem has to do with whether you might pinch some lines to the extent of straining the hydraulic fluid flow.

I think the combination of circumstances that is postulated is a very low probability event, much lower than things that we normally accept as being beyond the realm of probability. Consequently, I have been inclined not to want to push this issue.

If the committee feels that it is important to push it, then I think it is foolish to think about Grand Gulf as being the place to work on the problem. We had better go back and start with the first PWR and go through the whole gamut. I think that is an impractical circumstance, consequently I think we will have to look at the probability argument.

MR. EBERSOLE: May I comment on that. The reason that it comes up at Grand Gulf, it is the problem of impacting on the deck or the floor, which constrains the control rod drive units. Those are also elements that you could strain, which you must apply to the rods to get them to close. We are paying a great deal of attention to see that the decks or floors are not disturbed by the surge.

Looking at this, you could argue that this is also very unlikely, that sort of a disturbance. In my own view, looking at that is secondary to looking at the direct flow.

MR. OKRENT: We will have some discussion of this, as I indicated. The staff will tell us about how they judge that probabilities to be in fact low.

Any further comments.

MR. KERR: I have a request. If the Applicant has someone here who can tell us, I would be interested in knowing how they evaluate in their emergency procedures the source term that goes into the calculation of off-site doses.

One has to make these calculations to predict, in order to make measurements. I am just curious as to what their approach is to getting the source term that one puts into the calculation. If there is nobody nere to do it, I guess I can look it up sometime. If there is someone here who could comment on it later on at some appropriate time, I would appreciate it.

MR. OKRENT: If I understand correctly, you mean in an actual emergency.

MR. KERR: In an actual emergency, when one is supposedly making use of all the meteorology to get a dispersion relation, and the source term also has to go into this.

MR. OKRENT: We will put that under other.

MR. KERR: Do you understand the question I am raising?

MR. McGAUGHY: We understand the question.

MR. KERR: See if there is somebody here who can address it.

MR. McGAUGHY: We will try to get you that.

MR. OKRENT: I guess we will put that n the agenda under other. If there are no further comments, I suggest that we go to Mr. Plesset.

MR. PLESSET: Thank you.

In our subcommittee meetings, July 29 and 30, we considered a large number of concerns that we called out by a fellow from General Electric. We had Mr. Ebersole, Mr. Etherington, and Mr. Ray as members. We had consultants Busch, Catton, Garland, and Zudan, so we were well represented.

Most of these concerns are of a second order. The judgments were whether or not they were associated with a safety problem. You heard Mr. Ebersole concerns about the control lines for the control rods in the drywell. It was not discussed, but he did bring it up.

Now that is a little different concern from the hydraulic control unit. The question was, is it built to withstand this or not.

From the point of view of the load dynamics, there was agreement, both on the part of the staff and the Applicant, that it was quite conservative loads. At the meeting, on this question of the structural analysis, given the agreement on the loads, the staff has approved the structural analysis for the hydraulic control units. The question is at rest.

That is about all I need to report, Mr. Chairman. There are minutes available of the meeting. They are here if you would like to look at them. That is all, Mr. Chairman.

MR. OKRENT: Are there any questions?

MR. MOELLER: Looking at your minutes of July the 29 and 30, the draft minutes, on page 7, item (7), I wanted clarification, and I think that I see it. When I initially read the second paragraph in section (7), I apparently misinterpreted it.

The Grand Gulf plant, they have made changes in it to correct the deficiencies which led to the Browns Ferry failure to scram, is that correct?

MR. PLESSET: I think that is true, in general.

MR. MOELLER: So what you are referring to, or what the minutes refer to is that the review process, which should have found the Browns Ferry design defect, has not been improved. Is that what it is saying?

MR. PLESSET: I don't think so. We had a rather general discussion of the interface between the designer of the nuclear island and the architect engineerig, and the Applicant, and the item that they brought up as an example of this interface, GE as I understand it imposed certain requirements on the Applicant and the architect engineer, and this didn't work out the way it was supposed to.

MR. SIESS: Do you mean that the Browns Ferry didn't conform to some interface requirement that GE had set.

MR. PLESSET: I wouldn't say that in that way. It did conform, but it had defects in continued performance.

MR. SIESS: Then why do you call it an interface problem? It seems to me that it was just a design problem, and it met the interface requirements.

MR. PLESSET: General requirements on the scram unit are imposed by GE, and the detailed installation was made by the Architect engineer.

MR. SIESS: Are you saying that if the

architect engineer had made the design strictly in accordance to meet GE's interface requirements, they would not have had the problem?

MR. PLESSET: No, that is not what I would say. I would say that if it had been designed and installed with reasonable understanding of the function of the unit, they would not have had the trouble. They had plugging of the lines, and inadequate indication in the installation. This was changed by the kind of scram discharge volume that was installed and in the raising of the line.

MR. SIESS: I was just trying to understand why it was an interface problem, rather than a design problem.

MR. PLESSET: GE, as I understand it, does not prescribe in detail the installation.

MR. MOELLER: The draft minutes say, "Extensive discussion disclosed that GE relies on A-E audits to catch such design problems, i.e., no substantive changes have been made in the process since the Browns Ferry incident."

MR. PLESSET: I think that is correct that they have to rely on the engineering capability of the architect engineer and the Applicant. GE imposes certain requirements, they do not design.

MR. MOELLER: All right.

MR. PLESSET: I think we have a representative of GE here, if you would like to get a more detailed discussion of it.

MR. MOELLER: That would help me whenever it is appropriate.

MR. OKRENT: Now.

MR. MOELLER: All right, could I hear a clarification.

MR. PLESSET: Why don't we let GE talk about the Browns Ferry type of scram discharge volume problem. Is it an interface problem, or is it a GE problem, or what?

MR. SMITH: My name is Alan Smith, I am from General Electric. I am the Project Manager for the Grand Gulf Project.

Could you please restate the question?

MR. PLESSET: I think what Dr. Moeller would like is a little better understanding of the relationship of GE and the architect engineer on this matter of the scram discharge discharge that led to the problems at Browns Ferry, and why we don't expect those to happen at Grand Gulf.

What does GE do, and what does the architect engineer do?

MR. SHEWMON: Let's take them one at a time.

MR. SMITH: To the best of my knowledge, in the Browns Ferry incident, there were General Electric criteria that were generally implemented in that design. As you pointed out, there were some specific items that perhaps were not implemented in accordance with our engineers' initial intent, but that is a detail of the implementation.

There were, as I understand, not good housekeeping procedures and not good operational procedures implemented in the course of operation, which General Electric would have preferred to have seen, but nevertheless were not overt criteria that General Electric would monitor as such.

So it is a combination of operational activities and housekeeping that perhaps were not as prudent as they should have been in this case, and to some extent some of the details of GE criteria not being as fully implemented as perhaps the design engineers would have intended. But nevertheless, it met the overall nature of the criteria.

With respect to other issues between General Electric and the architect engineer community, as I stated in San Jose, we do generate various levels of documentation all from overt mandatory requirements to lower level information or nice to know material to the architect engineer community for their use.

Many of our design specifications, and so forth, are absolute mandatory requirements, and we depend upon the utility and the architect engineer's quality assurance program to ensure that these mandatory requirements are implemented.

There are types of documentation that we submit, such as informational items that are really at the discretion of the utility and the architect engineer to implement as they see fit.

MR. MOELLER: I think that answers it.

MR. PLESSET: The problem of the Browns Ferry scram discharge system was the volume that the unit discharged on the drive to the control rods was not adequate.

If one did not get good drayage of the scram discharge volume, the rods wouldn't go in. It was something that should have been detected by level indicators, which were also not of the best design. This is true only of a couple of others, and easily modified. The later ones don't have this.

Also the Browns Ferry lines were very asymmetric, some were very long, and very low pitch, which did add to the problem. But these are details of an installation which everything works meets the GE requirements.

It is a question of housekeeping, perhaps, as much as anything, but it has to be such that you don't have to have good housekeeping.

MR. MOELLER: Thank you. What I am gathering is that GE does not audit the design. They depend upon the architect engineer and the utility to do that, but I am not sure that that is the best way.

MR. PLESSET: I am not familiar with the details of all the systems, but I am sure cannot audit all the engineering involved in the system.

MR. OKRENT: Anything more on this matter?

If not, I propose that we go on to the next agenda item, which will be a report on the status of review.

MR. HOUSTON: My name is Dean Houston. I am the Project Manager in the Division of Licensing for Grand Gulf. I apologize for those of you who were here last night because this will be almost a duplicate of what I said.

I want to talk briefly about the chronology, the status of the outstanding issues, and the issues which have surfaced since the last ACRS meeting.

First the chronology. We issued the safety evaluation report for the operating license on Grand Gulf early in September of last year. This was followed by the two-day ACRS subcommittee meeting in Jackson, Mississippi, and a full committee meeting here in Washington on October the 16th.

Following that meeting, the committee issued an interim report. The interim report approved the issuance of a lower power license and asked us to return after resolution of three outstanding issues. Those outstanding issues involved management staffing and capability, the LOCA loads on the HCU floor, and hydrogen control.

Since the issuance of the interim report, the review has continued. We have issued three supplements, in December, June, and July. We have issued the low power operating license concurrent with the issuance of the second supplement in the middle of June.

The second supplement supported low power licensing, with the exception of equipment qualification, presented the structural and containment resolution of the LOCA loads on the HCU floor, and presented the resolution of license conditions for the management capability concern.

The third supplement presented the resolution for the equipment on the HCU floor, and generally addressed the resolution of hydrogen control. We would anticipate, following this up with a supplement sometime in September, Supplement No. 4, and the issuance of the full power license either in late September, or sometime in October.

Just a brief look at the status of outstanding issues, these were issues that were identified in the SER published and issued in September of 1981. We see mostly that these have all been resolved to the staff's satisfaction. In a few cases there are some items pending confirmation, or with license conditions.

I might briefly say for issue (4) on the LOCA loads, the only things to be confirmed are the thermal couples on that floor level to be seismically qualified and some re-review of increased response spectra. There is sufficient margin in the initial calculation not to give the staff any problem.

The equipment qualification, the license condition here was continued seismic and environmental qualification of the equipment. For the operation of the plant, the two main things that were still hanging were the MSIV, the seismic qualification of the MSIVs and RHR heat exchangers, and these will be qualified by the 31st of August.

containment purge, the license condition addressed signaling on certain containment --

MR. KERR: Excuse me, what does qualify by August 31 mean? Do you mean that the paperwork will be in place, or some testing will be done?

MR. HOUSTON: The tests will be completed, and I believe the qualification package will be in-house for review.

For condition (9), the containment purge, the license condition addressed sealing off certain containment isolation valves that were not qualified at that time. The qualification package is now at NRC and under review.

Continuing on with the listing of outstanding items, issue (13) was resolved with a license condition. In the license, we have a condition defining the operating shift advisor, the advisor to the corporate management, the training instructors, and the duties of the corporate safety review group.

Emergency preparedness ---

MR. MOELLER: Before you leave that one, would you summarize briefly what kind of experience you specified they had to have somewhere along the line, practical BWR operating experience?

MR. HOUSTON: I think that will come up on a later line item. I would prefer to defer that.

MR. MOELLER: All right.

MR. HOUSTON: I think you will get into the deeper details in the discussion of capability.

Emergency preparedness has been resolved for low power. We needed a follow-up site audit, and a FEMA approval based on looking at the State of Louisiana or the parishes of Louisiana, at their emergency plans.

MR. MOELLER: Where does it stand for full power?

MR. HOUSTON: For full power, the site audit, I believe, has been completed. We don't have their evaluation as yet. The Louisiana parishes have indicated that they would have a plan submitted sometime in September, and FEMA would look at that with a turnaround sometime either in late September or early October.

Issue (17), where we say "Resolved for interim operation with license condition," we intend here full operation for a period of about a year or a year-and-a-half. The license condition addresses the completion of the comprehensive qualification test program to demonstrate that the ignitor assembly will remain functional in the post-accident environment.

Issue (20) came up since the SER had been published, and these are the Humphrey concerns on containment. We are looking for a submittal from MP&L next week, which will address the analysis of some of these and the resolution and justification for full power operation, without resolution of others. I believe that Al Schwencer has a word to say.

MR. SCHWENCER: Al Schwencer, NRC staff.

With respect to the emergency plans and FEMA, my latest understanding on that is that the staff has asked FEMA to provide their input to us by the end of August, with the intent of the staff being able to complete its review in September.

MR. HOUSTON: We have done that, however, Louisiana has indicated that their plan would not be available until September. So there may be a later word on that.

MR. KERR: In connection with your comment on the Humprhey concerns, how does the staff decide what to do with something like this. Do you say to the licensee, answer these questions, or do you look at them first and see if they are legitimate questions?

MR. HOUSTON: In this case, he met with Grand Gulf in Jackson, Mississippi, prior to coming to Bethesda. Shortly after the meeting in Bethesda, we had a meeting with Grand Gulf and General Electric, and Mr. Humphrey. We took a transcript of this meeting, and I believe at that time the staff took a stand on the nature of the concerns, and whether they felt there was a major or a minor concern, and how we should go about resolution.

MR. KERR: I think your answer is that you do try to decide whether the concern is legitimate before you ask the licensee to address it.

MR. HOUSTON: Yes. In telephone conversations with Mr. Humphrey before the meeting, it appeared that his concerns had a technical basis and some merit. It wasn't the kind of thing that one could dismiss from the initial conversation.

MR. KERR: Thank you.

MR. HOUSTON: If we go on to the issues introduced since the last ACRS meeting, the first of these, the LPCI modification, this is or was a generic problem with only the Mark III containment design. In this case, the LPCI system enters right at the top of the core in a horizontal plane.

In a foreign reactor, which has other problems with valves, was using this for a shutdown cooling mode and at a reduced flow had set up resonant frequency in an instrument tube, and the instrument tube had failed in fatigue in about 12 hours.

The modification proposed by General Electric to overcome this problem is to put a flow diverter on the shroud at the entrance of the LPCI system. Grand Gulf has modified their particular plant to incorporate that shroud, so we would not see any problem here.

The second item is the probable maximum precipitation flood analysis. Since the plant was analyzed and that analysis submitted for staff review, the Applicant had gone back and looked at obstructions in a drainage basin and redetermined that for that particular flood analysis, which involves a six-hour rainfall of 30.5 inches, that the flood level would increase from 133 feet to 133.5, and this five inches is above the doorway's elevation to safety structure.

We have a license condition at the present time to sandbag certain doors to a foot above that elevation. In the meantime, the Applicant is looking at a permanent fix to seal the doors or put curbs around doors, this type of thing. That review is on-going and should be finished fairly soon.

MR. MARK: On this point, could you help me. This 133 feet, is that the highest level that you could get if it were installed in Wyoming, or higher than that, or what?

MR. HOUSTON: No. This is only a drainage basin in a rainfall. I believe that the Mississippi flood level is 95 or 96 feet, or something like that.

MR. McGOUGHY: It is 103 feet.

MR. HOUSTON: It is what?

MR. McGOUGHY: It is 103.

MR. MARK: Is that the highest ever seen, or something like that?

MR. MCGOUGHY: When the river tops the banks on the other side of the river, the river would go over to Shreveport before it would come to us.

MR. SHEWMON: You have come to the end of your time. Would you through the rest of what you have?

MR. HOUSTON: Surely.

We have the Humphrey concerns, and you have heard of those.

The independent design verification was an outgrowth of the Diablo Canyon syndrome. An independent consultant looked at two areas of Grand Gulf and has issued a draft of the final report, and it has not uncovered any great problem.

The last one is the one that I think has given NRC the most concern, staffing changes in respect to the critical time that these happened, just before licensing. I believe someone said here that the plant manager was lost, but it was the assistant plant manager from the operating side.

Since the plant was licensed, two of the outside corporate safety review group consultants have resigned and have been replaced, and one of those represented extensive operating experience. So it was the loss of the operating experience, from both the plant operating side and the safety review group, that the staff has been wrestling with.

That concludes what I have to present.

MR. OKRENT: The next agenda item is a short presentation on the status of the plant.

MR. McGOUGHY: Briefly where we are, we have completed loading all 800 fuel bundles in to the vessel. We are in the process of installing the vibration monitoring system for the prototype core to do vibration monitoring, we are in the process of installing that. Then we would hope sometime this weekend to achieve the first criticality, and start our zero power testing.

MR. OKRENT: Why don't we go on to the license control items, and get the staff to provide a summary of how they see it.

MR. JERRYLSTEIN: I have just a couple of slides.

As we have stated before in meetings with the committee, our first action was to evaluate the Grand Gulf hydrogen ignition system to evaluate its adequate on an interim basis. Upon completion of the evaluation, we proposed license conditions that require demonstration of safety margins within approximately one year. The licensing conditions would require additional testing and analysis.

At the end of this period, we intend to perform a final evaluation.

The interim evaluation of the Grand Gulf hydrogen ignition system was performed to determine the effectiveness of the system in controlling consequences of hydrogen releases from a TMI-type degraded core accident.

MR. SHEWMON: Sir, would you move back, so that we can read these, since we don't have them in hand?

MR. TINKLER: This is in order to prevent breach of containment and allow safe shutdown.

MR. KERR: What is the significance of a TMI-type degraded core accident as contrasted to some other accident where you get hydrogen?

MR. TINKLER: For the interim evaluation, we considered the accident sequences chosen by MP&L for evaluation of the hydrogen ignition system, without consideration of sensitivities on hydrogen, steam release breaks, as well as consideration of other sequences such as net emissions.

So in that regard we have not considered all the possible hydrogen release rates that might result in degraded core accidents.

MR. MARK: What is the basis for thinking one knows the rate at TMI-II?

MR. TINKLER: I suppose it is more of a belief that they have some idea of the upper bounds of the hydrogen release, and that the small break LOCA and stuck open relief valve transient represent reasonable hydrogen and steam release rates that might be result.

MR. MARK: It comes, then, does it not, from a calculated picture of boil down.

MR. TINKLER: Yes.

MR. OKRENT: The question was raised, the system depends on AC power from on-site or off-site, so it does not pretend to be able to deal with all events, for example. Actually there are also some sprays and compressors that have to run, which need power.

MR. MOELLER: What is the signal that the utility would use to actuate the hydrogen ignition system?

MR. TINKLER: An indication that the water level has reached the top of the active fuel.

MR. MOELLER: You mean, has decreased or

lowered to that level, and then you turn on the system?

MR. TINKLER: Yes.

MR. MOELLER: How do you know whether the system is working?

MR. TINKLER: To my understanding, the actuation of the system is in the main control room. I am not aware of what connection they have.

MR. MOELLER: I will ask them when they respond.

MR. TINKLER: The basis for evaluating the hydrogen ignition system was the testing and analysis performed and referenced by MP&L, as augmented by staff's confirmatory analysis and testing.

Previous testing performed by the ice condensor owners group, Livermore and Sandia Laboratories referenced by MP&L to demonstrate igniter performance.

Part of the staff evaluation included an independent evaluation of the system by Sandia.

Also serving as a basis for the interim finding was the MP&L endorsement of the PWR hydrogen control owners group research program.

The conclusion was that the hydrogen ignition system was found adequate on an interim basis, conditional to the successful qualification of the igniter assemblies. The testing will be completed in August, and as far as I know, it is on schedule.

The topics that we expect to pursue for the final review include the investigation of combustion phenomena pertinent to a Mark III containment, verification of wetwell igniter performance, the consequences of combustion in the drywell, and the mixing characteristics of the Mark III containment.

We also expect to pursue CLASIX-3 code verification and containment analysis, the consideration of accident scenarios, some consideration of the design features of the hydrogen ignition system. We expect to continue our review of emergency procedures, particularly those that are related to containment purge and spray actuation for degraded core accidents.

We also expect to continue our review on equipment survivability.

MR. KERR: What is the significance of CLASIX-3 verification, is that an unverified code that has to be verified?

MR. TINKLER: It is not unverified in the sense that it is virtually identical to the CLASIX code which was used for the ice condensor. In this portion of the code, you calculate the consequences of hydrogen combustion.

MR. KERR: Does this statement mean that someone has to verify the code, and then it is going to be used for containment analysis?

MR. TINKLER: The statement means that the CLASIX-3 code has been used by MP&L. We will pursue some additional verification of the CLASIX-3 through both independent analysis of other models and then by questions to the Applicant.

MR. MARK: Does the verification involve two calculations, and if they look the same, then they are both right, is that what verification means?

MR. TINKLER: It is not limited merely to comparison against other codes which had been largely unverified. There is a continuing data base of experiments to draw upon, and then to validate codes against. We would expect to see that the various containment codes be used to calculate the results.

MR. MARK: But it is, is it not, a strenuously simplified model. It assumes uniform mixing in any space it wants to talk about, and things like that?

MR. TINKLER: We would expect that by necessity most of these codes would remain relatively simple. But, yes, it is a simplified model.

MR. OKRENT: On survivability, the question was raised whether you could get effect on equipment due to forces that arise, for rate of burn, in other words, not just static temperature and pressure.

MR. TINKLER: Pressurization rates were considered --

MR. OKRENT: Not the pressurization rate that I have heard mentioned, but as burning moves rapidly along a path, you may get more force. It is not my field, so I am asking the question.

MR. TINKLER: Other than gross pressurization rates and differential pressure effects between the containment drywell, I am unaware of any other pressure effects on equipment that need be considered.

MR. OKRENT: Dr. Mark, do you recall that question by Dr. Schott of the locally rapid burning progression, and this is related to local pressure differences, which is different than the global one.

MR. MARK: I certainly don't recall the exact way in which he raised that question, but he raised it several times, and of course it is also raised in the literature.

When we start a burning, it may perturb the distribution of material as it proceeds, so that you can get some compressed air and you can also get burning which goes a different way than that that you would assume from uniformly mixed stuff with atmospheric pressure.

MR. KERR: Dr. Schott, are they representing your views accurately?

MR. OKRENT: Here he is. I didn't know he was here.

Do I remember correctly your question?

MR. SCHOTT: I believe Dr. Mark has represented the sense of the concern. The point with respect to the Mark III containment, the observation that I would emphasize is that the absence of large or moving parts in the flow paths in this type of containment makes this containment receptive or vulnerable to these large displacement flows or flow speed effect in comparison with the ice condensor situation.

So while these effects of accelerating flames may not be fully modeled, it is less important that they be modeled in the Mark III water suppression pool design than in one in which direct comparison is possible and acceptable.

The most conspicuous feature of the Mark III to me is that modeling of the expected response of deliberate ignition is less capable of achieving a precise result, and I would worry about the future prospects of meeting the call that Charlie Tinkler has made for a ´ ·onstration of safety margins with very much precision in the course of a year or any reasonable about of time in which to apply analytic models.

MR. HOBBS: My name is Sam Dobbs. I am with Mississippi Power & Light. I am going to give a very brief presentation today. The first thing that I would like to do is to respond to a question that it was indicated I would get in a few moments on how you know that the hydrogen ignition system is on. The hydrogen gnition system is controlled by switches in the control room, and there are indicator lights that indicate when the breakers have in fact closed. That is the only direct indication that the hydrogen ignition system is on.

However, there are a number of other supplemental indications which will be of very substantial value in the event that the hydrogen ignition system is used.

There are hydrogen detectors which will indicate the presence of hydrogen, and there are temperature and pressure indicators in the containment which would indicate the kind of puff phenomena we would anticipate being able to see, an increase in temperature and some small pressure response in the event that there were hydrogen burns.

MR. MOELLER: These hydrogen indicators or hydrogen monitors, are they designed to survive the environment of a burn?

MR. HOBBS: Yes, those were included in our equipment survivability list, which I will discuss briefly in a few moments, as something that we wanted to have to be able to monitor during the course of an accident.

MR. MOELLER: Why I raise that question, and I have said that to others before, but we, myself working with one of the ACRS Fellows, have recently completed a review of the past four years of LERs at BWR installations.

We have found that one in 20 of all of the LERs reported over that four-year period was a failure in a hydrogen or oxygen monitor. I just wondered if you were aware of the seemingly high failure rate of these monitors and if, indeed, you have assured yourselves that your monitors are better than the ones in the existing plants.

MR. HOBBS: I cannot address that right now. We can try to get an answer to you either later today, or somewhat later. I was not aware that there was that frequency on BWRs of LERs relating to a hydrogen or oxygen monitor.

MR. MOELLER: I would appreciate the answer, and it could be later, if you prefer.

MR. MARK: On this same general area, what is the nature of the signal reliability and precision that the water has indeed gotten to the level where you want it turned on, the ignition?

MR. HOBBS: If I may, I would like to defer that question to our Manager of Safety and Licensing who is mire familiar with the details of that system. But it is a very highly reliable, redundant system.

John, can you answer the question. The question was, what is the nature of the signal for water level that we use to actuate the hydrogen ignition system?

MR. RICHARDSON: This is John Richardson from Mississippi Power & Light.

The water level signal is the standard BWR cold reference link vessel level monitoring system.

MR. MARK: It tells the water level within a foot, or something like that?

MR. RICHARDSON: I don't remember the accuracy of the device right off the top of my head. They are very accurate devices, but I don't remember specifically what the number is.

MR. MARK: It is the standard differential pressure and water level.

MR. RICHARDSON: Correct, and then there are four channels or four condensing pods for that indication.

MR. BENDER: Is there anything crucial about the time when that ignitor should be turned on as it relates to the water level?

MR. HOBBS: Basically, the ignitor system needs to be on prior to generation of substantial amounts of hydrogen. We assumed in our modeling that it would be turn on at 20 minutes into the accident. That was an easy assumption to make.

In fact, we have, I believe, a period of 30 minutes after that in the cases which we have analyzes, in which you could turn it on prior to having substantial hydrogen generation.

MR. BENDER: It is more a matter of knowing to turn it on, as I understand it. If you use the water level monitor as a device, I guess the real question is, is there anything that could malfunction in it that would result in your not getting the signal when you need it, at least?

MR. HOBBS: We believe we have handled that in the design. I think there were concerns with reference to things like boil-off which originated as a result of

Three Mile Island. We have addressed that concern in our design.

In addition, knowing when to turn it on, basically, the emergency procedure guidelines as you get into a situation where you are attempting to control the water level, and you don't have make-up flow, or to cool the core without make-up flow, you end up in a procedure which is entitled core cooling without level restoration.

If you, in fact, are unable to determine water level and you are also unable to put water into the core, then you enter that procedure and you would begin to carry out the actions required.

MR. MOELLER: Mike, there has been a lot of experience with these level indicators for BWRs. They can see it coming, they have low, low and low, and low and low and low, and they are redundant.

MR. BENDER: I am not trying to challenge their ability to know the level.

I was more trying to establish what there wasn't anything crucial about when they decided, as long as they got a signal that said the level is low, if the accident were to confuse the signal, and they might not do something. I don't propose to suggest more than that you take a look and be sure that you know that the level indicator will be functioning at the time when you are concerned about hydrogen.

MR. HOBBS: The level indicators, in the plural.

MR. MOELLER: I am sure this question has been asked many times before, but you have containment sprays, I gather.

MR. HOBBS: Yes, sir.

MR. MOELLER: If you have a hydrogen ignitor that is on and the containment spray turns on, maybe you will tell me that it can't spray and hit an ignitor, but assume it sprays, and the water hits the ignitor, what happens? Does that bother the ignitor at all?

MR. HOBBS: No. That was included in our qualification testing program for the ignitors, the direct spray of water on the ignitors.

MR. MOELLER: Thank you.

MR. OKRENT: You don't always have 20 minutes in some scenarios. I hope the operator doesn't think that he always has 20 minutes.

MR. HOBBS: No, in fact, the operator is not keyed to the times in our scenario analyses.

MR. OKRENT: Fine.

MR. HOBBS: We want to cover briefly today system design and qualification, our base case selection, equipment survivability, structural capability, and the testing program.

Evidently, I have misplaced a slide since yesterday, and if you will pardon me, I will come to this slide in a moment. I believe that it is in your book.

We have ignitors which located in 90 locations throughout the drywell, the wetwell region, and the containment. Eighteen of those ignitors are located in the drywell, 11 are located in the wetwell, and 61 are located in the upper containment.

Nominally, we have 30 feet of horizontal separation between ignitors. We have a two-train system, and with one train out, we have nominally a maximum of 60 feet of horizontal separation between ignitors.

We make use of the General Motors AC Division model 7G igniter which the industry has a great deal of experience with since Sequoyah and Duke have made use of this ignitor. The ignitor is attached to a welded metallic enclosure with a spray shield, and has access provision, and has a transformer for voltage stepdown.

The power supply is 120 Vac power. They are powered by two ESF divisions, and each division is separated into two breakered circuits, and operation is by manual switches in the control room.

We have a 1700 degree Fahrenheit minimum glow plug surface temperature.

MR. MARK: You say that there are 18 in the drywell. Are some of those near the air inlets from the vacuum breakers?

MR. HOBBS: I believe currently the closest one is between 25 and 30 feet. We are evaluating locations to put some in some air discharges.

Thie ignitor assemblies are seismically qualified, and they are environmentally qualified. In addition to the very stringent, but normal environmental qualifications, they are designed to survive environmental conditions which would result from successive hydrogen burns.

We have a testing program which is underway, of which only burn tests remain to be conducted, and those are scheduled for August the 26th. Up until this time, we have not had any problem with our testing.

During operation of our hydrogen ignition system, if we get into an event which has potential for generating excessive hydrogen, that is, if you get into a core cooling without level restoration situation, the water level falls to or below the top of active fuel, at that time we energize the hydrogen ignition system and we initiate operation of the combustible gas control systems, purge compressors and the containment sprays.

In evaluating scenarios to use, we went through a number of considerations, and within the basis of not having make up water to the core, and of having a severe accident in the first place, we attempt to use a reasonable initiating event, and a reasonable and realistic scenario as far as the actual operation went.

We evaluated a number of initiating events. We evaluated various kinds of recovery events, and we selected two base cases, in which one was a stuck open relief valve which discharges hydrogen and steam directly to the suppression pool, and one is a small break LOCA with discharges hydrogen and steam initially to the drywell, and then later, due to some recovery events, to both the drywell and the suppression pool.

In the base case of the stuck open relief valve, we have system transient, such as loss of feedwater and main steam isolation valve closure, followed by safety relief valve actuation as required, or an advertent valve opening and the safety relief valve sticking open.

The mitigating events are that at the time the water level drops to the top of active fuel, the operator begins a sequence of opening additional safety relief valves to provide steam flow across the core, and provide additional cooling, and goes through initiating the containment spray, energizing the hydrogen ignition system, and turning on the combustible gas control system.

MR. OKRENT: Mr. Dobbs, I am a little concerned that if we try to get through all of the viewgraphs here, we will not be able to fit it into the five or ten minutes shown on the agenda.

MR. HOBBS: I was trying to skip some of them.

MR. OKRENT: I am afraid that you are going to have to choose the most basic issues and summarize them in the next five minutes. If members of the committee have questions, they have the viewgraphs, and they can ask questions.

MR. HOBBS: I will, despite that, go over the very next slide, because I think that is an important one.

In doing the evaluation, we made use of the hydrogen release rates from the MARCH code. We assumed that the combustible gas control system and ignitors were initiated at 20 minutes, that we had an upper pool dum at 30 minutes, that our burn parameters were initiated at eight volume percent, and we had an 85 percent completion on burn up.

We assumed that the flame speed was six feet per second, and one spray train was initiated only after the first burn. We assumed that there was some spray carryover into the wetwell region from the main containment.

Because of the fact that in that base case we got very close to the parameters required to initiate a burn in the main containment, at the very end of the event, we ran a double case, and we did initiate a forced containment burn at the end of the base case.

The pressures and temperatures for this base case, which we will be referring to periodically, we saw no burns in the drywell and the containment, except for the one forced burn in the containment, and 59 in the wetwell. We saw relatively modest temperatures in the drywell and containment, around 1000 degrees in the wetwell, and peak pressures of around 9 psi in all three measured compartments.

For the forced burn case, we saw about the same temperature in the wetwell, about 681 degrees in the containment where the forced burn was, and pressures that were no more than 24 pounds for the three compartments.

The small break in the drywell was an

extremely similar event. We made us of the same hydrogen release rates. We assumed that part-way into the event, additional as safety relief valves were open, we had a distributed release of hydrogen, with some releases in the suppression pool, as well as into the drywell.

The results of that were very similar. We did see a higher peak temperature in the wetwell of about 2300 degrees, with temperatures in the drywell and the containment of around 700 degrees and 860 degrees respectively. We did see pressures in the wetwell and containment of a little over 32 pounds, and of about 16 pounds in the drywell.

The most severe thermal environment that resulted from these cases was the wetwell burn. The wetwell burns were used as the basis for an equipment survivability program for all components, regardless of where they were located.

Our equipment survivability program --

MR. OKRENT: Excuse me, but the 2295 wetwell temperature in the drywell break, that arises from burn where, in the wetwell?

MR. HOBBS: Yes.

MR. OKRENT: Thank you.

MR. HOBBS: The survivability program was twofold. It was, number one, to determine thermal response of potentially essential equipment exposed to hydrogen burn environments, and to determine the ability of that essentially equipment to withstand pressures resulting from hydrogen burns.

We generated, based on three parameters, an essential equipment list which we evaluated the equipment on that list. The three parameters were that we wanted to maintain the containment pressure boundary, and the containment integrity. We wanted to be able to recover and maintain the core, and mitigate accident consequences. Then, we wanted to be able to monitor the course of the event.

MR. EBERSOLE: Can I ask you a question about these extremely high temperatures like 2250. In this, does the flesh effect that you have from this impinging on such materials as insulation, etc., and the containment suggests that there may be ignition resulting, and internal fire problems?

MR. HOBBS: We evaluated that question, and I believe that there were two potential areas where we were concerned we might have secondary fires. The first one was insulation on cables, which might be exposed to the burn environment, and the second was all reservoirs on pieces of equipment which required lubrication.

The evaluation which was made on the thermal response indicated that we did not anticipate surface temperatures which would lead to a secondary fire for the cables. In the case of the oil reservoirs, in almost all cases, I believe in all cases, the oil reservoirs were enclosed and would not have been exposed directly to the burn environment.

Had there been a secondary fire resulting, it would either have been contained in a very small area due to clear up around the reservoir area, or was actually designed under a pressure retaining system, so that we would not expect any real issue.

MR. EBERSOLE: Thank you.

MR. HOBBS: Basically, our equipment was assumed to survive if our maximum external surface temperature was less than equipment qualification temperature. If we did not meet that criteria, then the maximum internal temperature limiting components was less than the equipment qualification temperature. In some few cases, we had limiting component data where we were able, based on post-accident, to demonstrate survivability directly in that regard.

MR. SHEWMON: Sir, would you go to your concluding slide now please.

MR. HOBBS: Basically, my conclusion was just a brief discussion of the testing program. I will go to that rather than summarize.

Mississippi Power & Light is active in the Hydrogen Control Owners Group. On a generic basis with the owners group, we are entering into a test program to confirm the analytical assumptions that have been made, and evaluating the performance of containment response resulting from burns from the hydrogen ignition system.

There are three basic areas of testing which are anticipated. The first one is the testing of flammability limits and hydrogen-rich steam environments. The second one is testing of burn phenomena above the suppression pool. The final area is some testing aimed to resolve some issues concerned with the mixing of hydrogen in various regions.

MR. MARK: You have assumed the same hydrogen release rate, I believe, in your base case and in the higher range. That was 60-odd pounds per minute, which is quite possibly a rather high rate. It is a higher rate than on any average basis that occurred at TMI.

Have you explored the possibility with the possibility or the differences in case the rate was, say, a half or a third of that?

MR. HOBBS: We did a range of sensitivity study, and I believe that we looked at one half of that rate. We found results that were not dramatically different. I can talk about that case.

MR. MARK: In that case, it is clear what would happen. The wetwell will come to 8 percent.

MR. HOBBS: Very similar.

MR. MARK: In a longer time.

MR. HOBBS: Right.

MR. MARK: When I get to 8 percent, it is the same, but in the meantime more hydrogen has gone upstairs. So you would be more likely to precipitate the possibility of an upper containment burn for hydrogen.

MR. BENDER: Can I go back to the level indicator signal for a moment, please.

If you didn't get the level indication at all, would there be anything to tell you to turn on the ignitors?

MR. HOBBS: I believe that our procedures are clear that if you do not have level indication, in that circumstances, you are supposed to take the actions that are indicated in our emergency procedures.

MR. BENDER: That is not an answer.

MR. HOBBS: Mr. Ken McCoy, our Plant Manager, has something to say.

MR. McCOY: Ken McCoy, Mississippi Power &

Light Company.

The answer to that is, yes, if there is any doubt about the accuracy of water level indication, we do turn on the ignitor system. The ignitor system is to be turned on in the event that water begins to uncover the core, and any indication that that is happening, including just loss of water indication, is sufficient action to do that.

MR. BENDER: Thank you.

MR. SHEWMON: Thank you very much. We will take a ten minute break now.

(Short recess.)

MR. SHEWMON: Let's resume.

MR. OKRENT: The next agenda item is related to the question of available BWR operating experience, and also experience in the technical support organization at Mississippi Power & Light.

MR. STAMPLEY: Mr. Chairman, I am Norris Stampley of Mississippi Power & Light.

We would like to address this question in three tiers, or three areas. We have with us today, as we did yesterday, Mr. Floyd Lewis, Chairman and President of Middle South Utilities, our parent company, and I am sure that many of the committee members know him for his activities, particularly those following

Three Mile Island.

Mr. Lewis will comment on the support, the overview, the peer review activities at the Middle South system level.

Following him will be Mr. Jim McGaughy, our System Vice President for Nuclear Production at Mississippi Power & Light. He will comment on the technical and managerial support within Mississippi Power & Light Company.

Then, Ken McCoy, our Plant Manager, will tell you about the organizational experience in BWR operations at the plant level.

MR. SHEWMON: All of that in 25 minutes.

MR. STAMPLEY: Yes, all in 25 minutes or less.

MR. SHEWMON: Including discussion.

MR. STAMPLEY: Including discussion. These are going to be very brief comments, and then we will entertain your questions.

Mr. Lewis will kick it off.

MR. LEWIS: I am Floyd Lewis of Middle South Utilities.

In our system, when we complete the consolidation of two of our operating companies, which we expect to occur in the months immediate ahead, every corporate unit in our system will have some direct involvement with nuclear. Three of our companies will be licensees for nuclear power plants.

Our energy company is involved with Grand Gulf, about which we are talking today. Our service company provides nuclear services with people dedicated to technical support of the nuclear units. And our fuels company is involved with nuclear fuel procurement.

We were concerned to know that we were employing our resources -- basically people and experience -- in a way that would be most productive for the system until we retained a nationally recognized consultant to give us a recommendation as to the optimum organization of our system nuclear resources for achieving our goal of safe and efficient operation of nuclear power plants.

Out of that has come the decision to create a system nuclear oversight committee which has now been created. It is composed of the highest level officer of each of the operating companies with nuclear power plants and our service company -- that is, the highest level officer with professional nuclear background. We will add to that three outside, independent members with the highest nuclear qualifications that we are able to secure.

Now this oversight committee is intended to perform a role in establishing, through peer review and exchange of information, improvement in the flow of information as between the separate units and the system that to this point in time pretty well stands alone in terms of the licensing and operation of their plants. We have at Arkansas experience in operating a nuclear plant that goes back to 1974 and we want to make use of that in a greater way than we have so far.

We also intend for this peer review, which the committee will report directly to me to assure compliance with all safety standards and to set standards for our system which will be as high as we possibly can make them. We believe that this will enable us to enhance the use of our nuclear experience to improve the flow of information between our various units in the system and will also enhance the professional opportunities for those in our system to provide for movement between the various units at times in the future.

I would like to conclude by saying that while the management of the Middle South system is very concerned with the financial consequences of outages on the nuclear units in the system -- and we have had some rather sad experience. Back in September 1980 we lost $18 million in one month because of Asian plans clogging up the fueling system -- the service water cooling system at both units at our Arkansas plant.

This concerns us very much, but management is firmly of the opinion that the worst thing we could do in terms of protecting the interests of the stockholders who have their money invested in our system would be to permit the unsafe operation of any of our nuclear units.

(Slide.)

MR. MC GAUGHY: My name is Jim McGaughy, Mississippi Power and Light. I want to briefly describe our organization, our support organization from the plant that supports the plant. We talked about this the last time we were here. I want to point out one or two changes since that time.

The manager of quality assurance reports directly to the senior vice president, Mr. Standless, so you have a project manager for unit two. You have me responsible for operation of the unit one, and then we have the manager of quality assurance, who looks over both of these projects.

Reporting to me is the nuclear plant manager, who will talk to you in just a second, Jim McCoy, our manager of nuclear plant engineering, who heads up our design group, and our manager of nuclear services, Larry Dale, who heads up our nuclear services efforts, which includes safety and licensing and includes our corporate health physicist who is now a certified health physicist.

(Slide.)

MR. MC GAUGHY: I will show you this slide, which shows all of the people who are not a part of the operating and maintenance organization in the plant but were part of the support organization, which includes all of the groups that I showed you.

We had some total of 118 professional people in Mississippi Power and Light Company and including the people in the support company it includes a total of 178 people, 1,333 years of professional experience, and 806 years nuclear experience, with an average of six years per man of nuclear experience.

Now, to update that for you since that time, we have gone from the 170 figure there to 218 total professional people who are supporting the plant, not part of the operation and maintenance organization. The experience is a total of 2,300, nuclear experience, 1,449, and nuclear experience 6.6. So the point I am trying to make is not only have we increased the number of people we have, we have also increased the average years of experience since we talked to you last fall.

And with that I would like to turn it over to Mr. McCoy.

MR. SHEWMON: On that table, how many of the people you have were not there last fall, so what kind of turnover has there been, even though there has been an increase?

MR. MC GAUGHY: I do not have that figure. Mike, correct me if I am wrong. The only man in our organization that we lost that had significant experience was our assistant plant manager, and Mr. McCoy will talk about that.

MR. SHEWMON: So that is basically new people, then, that you are talking about here?

MR. MC GAUGHY: These are new people, yes, and we have not lost very many at all, except for this man.

We mentioned two members of our safety review committee. We replaced one of them with Joe Hendrie, and the other one we replaced with Dr. Wayne Jones, who is head of the Center for Nuclear Studies at Memphis State University.

MR. SHEWMON: At Mississippi or Memphis?

MR. MC GAUGHY: At Memphis State University.

MR. SHEWMON: They are the outfit that runs the training program?

MR. MC GAUGHY: They run our training program and they do our screening for us and they do test analysis.

(Slide.)

MR. MC COY: I am Jim McCoy, plant manager at Grand Gulf. I would like to address the operating experience issue directly. I think that is the issue on the agenda.

To start with, I would like to point out our organization and the critical jobs and give you a brief summary of the experience of the people in those jobs. This is the plant organization. The critical jobs are the assistant plant manager, the nuclear support manager. Those two jobs are equal qualification and they serve as duty managers along with myself in the event of emergencies, et cetera.

In addition, the operations superintendent, maintenance superintendent, and the chemistry and radiation control superintendent are three of the key members in our staff. We also have an outage manager position which is a backup position for management at the top level.

At the present time, I myself have 18 years of lightwater reactor experience. Eight years of that have been at Grand Gulf throughout the design and licensing and startup of this unit. The assistant plant manager presently -- and let me first address, since this is one of the issues that was raised, we did have a turnover in this job in June of this year. The man who was in that job went back to his previous utility with a significant promotion and we were sorry to leave him or to have him leave us. But we have been able to make accommodations for that and we feel in a satisfactory manner.

What we did was we promoted the man who had been the nuclear support manager into this job. That man's name is Dick Embersino. He has twelve years of BWR commercial experience as a startup engineer, a maintenance superintendent, both at Peach Bottom and as a technical operations consultant to the General Electric Company assigned as site operations manager for the Dwane Arnold Energy Center, also for one of the Italiam GE reactors and a Japanese GE reactor.

We have in the interval since we talked to you last hired a man as outage manager who had 20 years, approximately, of reactor experience. He was licensed at the SRO level on a commercial BWR in 1973 and was associated with the startup business with General Electric. And his latest job had been a corporate training manager with a utility that had both BWRs and PWRs. He had previously served as an outage manager for General Electric Company.

MR. OKRENT: Did either of those individuals have experience as a senior reactor operator on the boiling water reactor?

MR. MC COY: Yes, this gentleman did have a license and had been a senior reactor operator on a Dresden reactor.

MR. OKRENT: For a lengthy period of time?

MR. MC COY: His experience was as an on-shift General Electric employee. He was at the simulator for General Electric, training students on the Dresden reactor for a period of about two years, as I recall.

MR. SHEWMON: There are three Dresden reactors. I assume you are talking about two and three.

MR. MC COY: That is correct.

MR. OKRENT: And the first individual has not had direct experience as a senior reactor operator?

MR. MC COY: That is correct. He was

certified by General Electric Company at the SRO level at their simulator.

The other individual, I would point out to you, our operations superintendent, has twenty years lightwater reactor experience. All of that is as a licensed reactor operator or senior reactor operation. Fifteen years of that is commercial BWR experience, including four different BWRs. He was a shift supervisor for the startup of Vermont Yankee and came to us from Shoreham.

The maintenance superintendent has 22 years of lightwater reactor experience. He was SRO-licensed on the Oyster Creek reactor. He has 16 years of commercial BWR experience.

Our chemistry and radiation control superintendent has 14 years of lightwater reactor experience, including being a reactor operator on a research reactor at a university, being responsible for the health physics program at that university, and then being a health physics supervisor at a commercial-2 unit reactor station, the Point Beach station, for four years prior to joining our staff.

MR. OKRENT: I am sorry. Who was that again that you said had the operating experience on BWRs?

MR. MC COY: The operations superintendent has approximately 20 years of experience in operating lightwater reactors. Fifteen of that is direct operations license on-shift experience in BWRs.

This is the operations chain that we have at the plant. We have at the supervision level the shift supervisors, a total of 150 years of reactor operating experience and 40 years of commercial BWR experience. One of our shift superintendents was a shift superintendent at the Quad Cities plant prior to joining us. He is also a nuclear engineer and was a reactor engineer at that site. Another was a reactor operator at Fitzpatrick.

(Slide.)

Another question that has come up in our operation is that since we talked to you last we had some conversations about whether we had adequate staffing in certain areas. We have made some changes in the areas that we feel are critical to the operation of the plant and the authorized level of people has gone from, in the operations department, from 60 to 81, in the chemistry and radiation protection from 36 to 58, and in the technical support area from 46 to 60, and in the training department from 10 to 21, and in the instrument and control area from 37 to 62.

So you can see we have made some significant commitments to increase the capability in-house. We have also made significant progress in getting to those levels. The present levels are in the parentheses here on the side and, as you can see, in all areas we are well along to getting to our authorized level.

The objectives we are trying to achieve are to reduce the use of overtime, to reduce the reliance on contractors and to reduce turnover but be prepared to handle attrition.

MR. MOELLER: You have not reduced some other group in order to fill these slots?

MR. MC COY: That is correct and I might add that these are not shown in the FSAR. These are just things that we have done on our own. We increased the total staff from 439 to 510 at this station.

MR. MOELLER: Thank you.

MR. MC COY: One other area that we are concentrating on at the plant that has been a subject of concern both to us and to the region inspectors is the area of procedure adherence. We are taking an active role in trying to improve our performance in this area. We have management commitment, including letters directly to our employees from the president of the company and from the line management all the way down and stressing the importance to our overall operation of adherence to procedures.

We have conducted training sessions for all employees in this area. We are making improvements in the procedures to make the procedures easier to adhere to and easier to follow and where necessary we are taking disciplinary actions in the event that procedures are not followed.

I might add that I do not think that we are unique in having this problem. The industry is going through a stage at the present time where the volume of procedures is increasing drastically. To give you some idea, we had some difficulty in getting our surveillance program kicked off at the plant as we went into operation and we said our surveillance operators could visit several older plants operating and we found that they had about 200 surveillance procedures and we had 1,000 to comply with, and the complexity of those surveillance requirements has increased also.

We also found that the procedures at the other plants ran in the neighborhood of 2,000 procedures, where ours ran in the neighborhood of 7,000 procedures. The difference is that we have a much stronger procedural control of the maintenance work that goes on at the plant, both preventative and corrective maintenance. So we think that we are making significant progress in the area of doing things in a formal, controlled manner.

And at the same time we do have some jeopardy there due to the fact that with this vast amount of procedures we are having difficulty in ensuring that those are all followed and that the people are all familiar with it.

MR. BENDER: The violations that are talked about here, are they more in the nature of approvals to do work or are they violations of tagging actions or things of that sort? What type of problems are being observed?

MR. MC COY: They were primarily of an administrative nature -- people failing to either be aware of or to follow some detailed procedure, for instance proper donning of clothing going into a radiation area or following a detailed procedure for conduct in different areas -- things of that nature.

I think it is both a training problem and due to the fact that we do have very complex procedures.

MR. BENDER: It is almost frightening that you have to remember 7,000 procedures.

MR. MC COY: Yes. And, as I said, we are tackling that problem, trying to make them simpler and to have things that key people in to when to go and pull the procedure rather than trying to remember.

(Slide.)

The last thing I would like to point out is that we are taking steps at the plant to improve our own management. We feel that this plant needs a very dedicated, well-managed operation.

Some of the things that we have in progress are we have a management development program that is endorsed by the company. We have conducted team-building sessions for both our management and we have in progress team-building being conducted for the supervisory level all the way down to the first line supervisors.

We think this is particularly important in that we have staffed up rapidly and since this is our first nuclear unit many of the people come from various backgrounds and it is important for us to get all of these people pulling together with a sense of identity, of belonging to the unit.

We also have experienced consultants working both with myself and on shifts where we have shift superintendents that do not have commercial BWR experience, and we are taking efforts to increase the number of SROs that we have available to management. Specifically, we have taken our startup manager as he completed the pre-op test program and sent him to the SRO certification school, which is in progress at the present time.

And we also have created two positions as technical assistants to the plant manager where we can bring in experienced people and put them through the SRO training program to be able to supplement management.

That is all I have.

MR. SHEWMON: Thank you. Any questions?

MR. OKRENT: We might have a few minutes from the Staff on this.

MR. SHEWMON: Before we do that, I have a few questions for Mr. Lewis. It would seem to me that a year ago one might conclude that you were not as well equipped corporately to handle recruiting across the country as you are now, yet you did not mention and nobody subsequently has mentioned what you have done or any consultants there.

Would you comment a little on changes there, or has this been more trying harder with the old procedures?

MR. LEWIS: Well, I think it includes a great deal of trying harder. In some of our situations it also involves finally getting response to pressures to upgrade the salary levels for various nuclear type personnel.

This is not specifically Mississippi Power and Light Company, but in a number of our companies I have had to continually talk with chief executives, who have had great difficulty accepting that the fact that he was just going to have to say nuclear is in shorter supply -- nuclear qualifications and experience -- and you have got to up the price and pay more there than you do for an equivalent engineer in a fossil fueled power plant. Finally that bullet was bitten, and the results have been quite helpful.

MR. SHEWMON: When you interview for young engineers, do you do it outside of the states you have reactors in?

MR. LEWIS: Yes, sir. We do interview in other locations, although the principal effort, I think, is on engineers is with the schools in our general area.

MR. SHEWMON: Thank you.

Okay, does the Staff have something?

MR. BENEDICT: Are there questions, per se? We did no have a prepared presentation.

MR. OKRENT: What is your view on the adequacy of BWR operating experience?

MR. BENEDICT: This is Robert Benedict of the

Licensing Qualifications Branch.

Certainly the operations department, the superintendent, the shift superintendents involved meet all of the requirements that are necessary for an operation for licensing.

MR. SHEWMON: Are you squeezing that thing?

MR. BENEDICT: Yes. I will squeeze a little harder. Is that better?

(Laughter.)

We do not have any particular concern there and never really have in the manning of the shifts. Our earlier problems have been concerned more with the operating expertise in middle and upper management levels, and I think our concerns have been assuaged by the consultants and the contractors that MP&L have brought on board.

We were sorry to see the assistant plant manager leave because he represented a major proportion of the BWR operating experience that was represented in the plant operations department -- the plant staffing total.

MR. SHEWMON: Fine. Is that it, then?

(No response.)

MR. SHEWMON: All right. Thank you. Are we ready to go on, then, to the LOCA on hydraulic line effects?

MR. OKRENT: If there are no further questions by the Committee in this area.

MR. SHEWMON: Are we satisfied with their staffing levels, gentlemen?

(No response.)

MR. SHEWMON: We seem to be.

MR. OKRENT: I think I would propose that on the next item we have the NRC Staff tell us how they are addressing this matter and whether they have any residual questions or whatever, and then Mississippi Power and Light can comment with regard to what they have done.

VOICE: Jim Bremmer will give a presentation.

MR. TERAO: My name is David Terao and I am with the Mechanical Engineering Branch. I do not have any prepared slides or statements to make. I just wanted to briefly summarize the Staff review to date on a few of pipe breaks on the CRD piping bundle.

Apparently the ACRS discussed this topic yesterday and I did not have the benefit of hearing the discussion, so I do not know the ACRS concerns. I hope I do not understate the problem too much.

As I understand it, at the previous ACRS subcommittee meeting it was noted by the ACRS staff that a CRD piping bundle was routed very closely to a high energy reactor recirculating piping and the concern was that if you had a pipe break in the recirc piping that pipe with loads may impair the CRD function, the scram function.

At the time of my discussions with the Applicant, Mississippi Power and Light was looking into the problem. They were aware of the close routing and were doing an analysis at the time to determine what exactly would be the effect of these loads and they were going to provide a fix, if required, prior to fuel load.

Apparently the problem was that if you had a longitudinal break in the recirc piping that the jet impingement and pipe width loads would affect the CRD piping. So what Grand Gulf was proposing was to reduce the conservatisms to reduce the fatigue and the high stresses in the piping to eliminate the break from a high stress point of view -- that is, using the branch technical position, NED 3-1.

They were also proposing to look at modifying the pipe width experiment, I believe reducing the gap from about four inches to two inches, and possibly stiffening the rods in order to reduce the deflection of the pipe width from about 17 inches to 12 inches, and therefore the pipe would not impact the CRD piping.

And as a result of their analyses, Mississippi Power and Light submitted a letter on April 27, 1982 which stated that as a result of their analyses, there was no need to provide any further shields to protect the CRD bundle or the support. So based upon their analyses, we felt the issue was then closed.

MR. OKRENT: Could I ask a question? Does the staff have some kind of an evaluation of an accident where you have a rupture of this particular pipe and you lost a sufficient number of lines that you were not able to shut down the plant if you reflooded it, in other words. And if you do, is it a higher or a lower release category kind of accident?

Where does it fall in the spectrum of events, in your opinion?

MR. TERAO: Well, our criteria, as stated in the Standard Review Plan 3.6.2, defines the methodology for postulating breaks and determining the --

MR. OKRENT: You are answering a different question.

MR. NOVAK: Dr. Okrent, the logic that we have followed is that if there is sufficient time for operator action to take place, then the likelihood of an accident progressing to where you are unable to drive in a number of rods to shut down is low enough that it need not be specifically analyzed. In other words, we have not gone through the actual analysis to identify offsite releases; and what we have relied on is to establish that there is enough information available to the operator and there is enough time that it is reasonable to assume that the proper actions would be taken such that the continuation of that scenario could be aborted early in the event.

MR. OKRENT: Now, what you've said is the operator can do something manually if he loses these lines. Is this correct? I don't know. I'm just asking.

MR. NOVAK: That is my understanding, that they could actually go to local stations and perform actions locally in order to shut the system down.

MR. OKRENT: And these would be accessible?

MR. NOVAK: That is part of the review, that they would have to be accessible, and my recollection is that they would have up to 30 minutes to perform that function.

MR. OKRENT: I'm surprised at the 30 minutes for a large LOCA, which I think is what you're talking about. Maybe you are right.

MR. NOVAK: This is where the conservatism would have to come in. In other words, I recall the generic study. I go back to the original concerns of Mr. Michelson where following his work the staff did look at the specific problem. And the conclusion that we drew was that we were satisfied that if in fact the design criteria were checked on a plant-by-plant specific basis -- in other words, the material and the actual construction would have to be checked plant by plant -- and we were satisfied that the likelihood of a pipe break was sufficiently small. However, we also looked at, in the event certain pipes did fail -- and I would have to qualify exactly the size -- but the review did look at the amount of time that the operating staff would have in order to put people at the right local stations to accomplish the actual shutdown.

MR. OKRENT: Again, I just wanted to make sure we're talking about the same event. I thought Mr. Ebersole was interested in a large pipe rupturing in some way, meaning medium to large LOCA in my mind, and that you would not have enough rods in to shut down when you reflooded.

MR. EBERSOLE: That is correct.

MR. OKRENT: If you're telling me that there's 30 minutes, you've analyzed it and the guy can get there, fine. But I'm just wondering do you really have 30 minutes?

MR. NOVAK: I don't know that we are talking about the same pipe.

MR. EBERSOLE: I think you're both talking about entirely different things. As a matter of fact, Tom, I don't know how you could move a rod at all without the drive tube supply and the drive tube exhaust.

MR. RICHARDSON: Mr. Ebersole, if I may add something. I was going to discuss it during my presentation, and I might mention it now.

If you sever all of the CRD insertable withdrawal lines and insert reactor pressure, you will insert the control rods within three to four seconds.

MR. EBERSOLE: That depends upon the characteristics of the LOCA. What you're counting on is the reactor pressure.

MR. RICHARDSON: That's correct, but the tail-off of the pressure of the LOCA is such that you're at above 1,000 pounds for the first five seconds, and you are above 600 pounds for a considerably longer length of time.

MR. EBERSOLE: I'm aware of that phenomenon, but unfortunately, what you didn't do to invoke it was go to the limit which was a total severance and then invoke the presence of reactor pressure to execute the upper thrust to get it in with a degrading pressure. As a matter of fact, what would happen is you would get some sort of random effect, and you would not get this -- you would have some rods effectively going and others not going in and the reactor pressure decaying, and some of them would still have accumulated pressure to help them and others would not. It's a very complicated business.

MR. SHEWMON: Jess, we aren't talking about a big LOCA now because we have taken care of that.

MR. EBERSOLE: Yes, we are.

MR. SHEWMON: I'm sorry. If we're talking about a big LOCA, then what he says is true, isn't it?

MR. EBERSOLE: No. A big LOCA may not necessarily completely sever all of those.

MR. RICHARDSON: That is correct. If they are not severed, as long as the flow area is greater than 35 percent.

MR. EBERSOLE: May I pursue what Tom was talking about?

Tom, I don't know about this station that an operator can go to. I am learning something indeed. I thought in order to move a rod you had to apply hydraulic pressure to it, and to apply the pressure you had to have pipes to get it there, and these pipes are presently about that size to be gone, and I don't know how you can get the rods to move.

MR. NOVAK: As I said, we may be talking about two different problems.

MR. EBERSOLE: I believe we are.

MR. NOVAK: But in going back to what I thought was the thrust and the concern that Mr. Michelson had raised, it must be over a year ago now, a large amount of staff effort did go into tracking down the likelihood of the inability to scram.

MR. EBERSOLE: The probability under other circumstances, yes. I think what Dave was purusing was had you looked at the consequential effect of this and determined what the ultimate consequence would be. My present knowledge of this is that you have not because it is too dirty a thing to look at. It is like an unmitigated ATWS.

MR. SHEWMON: Dave, do we have written down so that anybody other than you or Jess and possibly Hugh know what accident is being considered here?

MR. OKRENT: Well, I would prefer that Jesse state it since he is the one that posed the question. I have been trying to understand it, and I think my understanding is correct that he is talking about a medium to large LOCA affecting these hydraulic lines, not necessarily rupturing, maybe crimping something closed or whatever.

MR. SHEWMON: And this happens on both sides of the plant or one side?

MR. OKRENT: No. You only have to lose a certain number.

MR. SHEWMON: So where it takes out one side is okay?

MR. EBERSOLE: Not one side, Paul. Just a few tubes. Not many, four or five.

MR. SHEWMON: A few rods don't go in, and that stops things somehow?

MR. EBERSOLE: Right. It stops the ability to scram.

MR. SHEWMON: But we've designed this thing so that if half of them go in, that must do something constructive.

MR. EBERSOLE: It only takes, I'm going to argue, maybe a couple of these rods, control rods, not to go in to cause a problem when you reflood with reflooding water. The plant will then go critical.

MR. SHEWMON: So if there's only two rods that aren't in, then we probably do have some time but not 30 minutes. Is that the question?

MR. EBERSOLE: No. I say if as little as two fail to go in, we are in trouble. There are approximately 300 and -- well, it is twice 185 or thereabouts.

MR. SHEWMON: If we don't have two rods that go in, what kind of trouble are we in?

MR. EBERSOLE: I believe I could be corrected on this. It might take three.

MR. RICHARDSON: You are in no trouble with two rods. In the worst case situation, which is a group of rods clustered together, the highest order rods at the worst time of the cycle, it would take five clustered rods together.

MR. EBERSOLE: Five clustered rods together.

MR. RICHARDSON: And with five rods out you can achieve hot standby zero percent power.

MR. EBERSOLE: Wait a minute. Let's stop for a minute. How can you achieve hot standby? You have an open reactor which is going to be reflooded due to the LOCA.

MR. RICHARDSON: It is just the terminology; that temperature and pressure when you reflood, okay, you are at zero power; you are subcritical with those five rods out.

Now, with a random selection of rods you could have up to 50 percent of the rods completely out and still sustain a hot shutdown at zero power.

MR. EBERSOLE: With 50 percent of the rods out you're saying you can open the core and reduce the primary coolant to containment pressure, in other words, reflood it, and take away the decay energy and any fission energy that may be present as a result of these rods out? Is that so?

MR. RICHARDSON: With 50 percent of the rods out in a random selection now.

MR. EBERSOLE: Well, have you randomly organized these tubes so that a local region doesn't have a peculiar set?

MR. RICHARDSON: The pattern is generally established by making sure you can get all of these things underneath the vessel. I'm not sure if they took a look at where the particular bundles and rods in the core are located, but they are located in a manner that you've got four sets of bundles, one in each quadrant, so you are only talking about 25 percent of the insertable withdrawal lines in any particular quadrant.

MR. EBERSOLE: If they are localized so these are in one quadrant, that makes the matter worse because then you are affecting the rods in only one quadrant.

MR. RICHARDSON: Just because the bundles are in a quadrant doesn't mean that goes to a group of rods in the core of the quadrant.

MR. EBERSOLE: Well, let's summarize this, and maybe we can quickly get rid of it. How many rods can you have stuck completely out in a LOCA, assuming a reasonable burnout level, and take care of the subsequent reflooding and any power beyond? Decay energies may be present with the RHR system. Are you going to tell me half the rods out?

MR. RICHARDSON: I'm going to tell you that with 50, like in a random pattern, with 50 percent of the rods out, then you can reach a hot --

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

MR. RICHARDSON: Well, it means at normal temperature, 500 degrees or whatever.

MR. EBERSOLE: You're not telling me anything.

MR. RICHARDSON: Well, obviously at low temperature you have the amount of reactivity associated with the temperature.

MR. EBERSOLE: You would have a feed pressure at most in the containment?

MR. KERR: Suppose the water were only halfway up on the rods. Would the reactor be critical?

MR. RICHARDSON: I'm not sure. I would have to take a look. Obviously, with less water you've got less chance with that same rod pattern to reach a critical configuration.

MR. EBERSOLE: Well, I think the point is I haven't got an answer. You tell me hot standby. We have a pressurized reactor which will be at containment pressure.

MR. BENDER: There is no reason to try to show that you can operate with half the rods out. We are really thinking about what kind of accident can occur that could cause some fraction of the rods to stay out while the rest of them go in.

Now, the old accident was based upon the premise that at least five rods could be out. That number has been known for I don't know how long, for a number of years. GE has done that analysis a hundred times. How many more than that could be out depends a little bit on the burnup, on the circumstance under which the reactor is operating, and the temperature of the water. The temperature of the water will be to some degree dependent upon the size of the break and how fast the pressure is decaying. If it stays up, then the argument that is being made that the reactor will stay subcritical is true for a large number of rods being out.

Now, you can put all of these circumstances together and come to a conclusion that the likelihood of the event occurring is small, or you can make the argument that I don't care how small it is, it could happen. I don't argue that it can't happen, but I think the probabilities, when you put all of these things together, are going to be low enough so it is not the sort of thing that we have to be concerned about.

That is a position I have taken any number of times. I think if we are going to make a case for it in this particular event, in this particular reactor, then we have to make a case for it in every BWR that exists. I don't see why we don't table it for the time being and deal with it in a more general way when somebody who has the time to do the probabilistic analysis can sit down and put all of the uncertainties together and see what it means.

MR. OKRENT: Mr. Bender took the words out of my mouth. After this discussion was finished, and I hoped we would not exceed the agenda time, I was going to propose that we develop some kind of a memorandum to the staff asking have they looked at this probabilistically, and if not, could they, and at some time in the future examine it. So he has, in effect, made the recommendation that I was going to make.

MR. BENDER: Well, two members of the subcommittee agree for a change.

(Laughter.)

MR. SHEWMON: Are there any other questions on this issue then, or have we come to a satisfactory conclusion?

(No response.)

MR. SHEWMON: All right, fine. Thank you.

MR. OKRENT: Okay. We don't need a presentation by MP&L unless they are desperate to give one.

(Laughter.)

Let's go to the next agenda item. This is the proposed venting of the containment in the event of the buildup of pressure as a result of some postulated severe accident, which my understanding is MP&L in a letter in June mentioned as a tentative portion of emergency procedures.

Could you tell us a little bit about what you had in mind?

MR. HOBBS: My name is Sam Hobbs from MP&L.

(Slide.)

I wanted to go through the chronology very quickly.

Basically, the emergency procedure guidelines which were developed largely prior to containment ultimate capacity analyses, which were done as a result of concerns on hydrogen, included the option to allow containment venting and design pressure. Our emergency procedures at 15 pounds would allow the operator and the shift technical adviser and the shift supervisor the option under the appropriate circumstances of venting at pressures in excess of 15 pounds.

Because of the fact that this was an option and because of the fact that this had not been analyzed as a part of our sensitivity studies on hydrogen, the NRC was concerned about the consequences of any interactions with the hydrogen ignition system and the effects on the containment, whether or not that was the appropriate thing to do.

Our judgment was that living under those circumstances would in fact mitigate the event. However, in order to resolve the concern expeditiously, because we had a very detailed analysis in hand that indicated that we had an ultimate capacity that was essentially above the peak pressure that would be reached during any hydrogen burn events, we committed to raise the mid-pressure to 50 psig, and this involved the peak hydrogen burn pressure below the ultimate capacity.

At that point the future concern was raised that would the events in fact be operable under those pressure conditions. MP&L is currently working with the TMI BWR owners group regarding emergency procedure guidelines in this area.

(Slide.)

Now, the concern, as we understand it, is that it is related only to degraded core severe accident type scenarios; that it is not related to any kind of design basis accident. The object of venting would be to provide pressure relief and containment protection in the event that you were in a very bad situation and you wanted to have the option of having some releases and having some control over when you had it instead of violating containment integrity and perhaps having a very large, uncontrolled release.

The current containment vent purge system is nonsafety grade with the exception of the isolation valves and the radiological or radiological system and pressure concerns. The effluent is filtered with the ductwork, the filter trains, and all are not intended for pressures of this magnitude.

(Slide.)

Now, what MP&L was proposing to resolve this issue is that we will pursue the issue generically with the BWR owners group. We have initiated correspondence with the owners group, and we have apprised the Emergency Procedure Guideline Committee about the problem; and it is our understanding that they are going to take this matter into consideration. We have initiated feasibility-desirability type studies on system requirements and the operational consequences of venting. Those studies are not completed at this time. I think the key point is that venting is not needed for containment integrity protection for degraded core hydrogen concerns and is really only needed for accidents and scenarios more severe than have been fully considered and analyzed at this time.

Are there questions?

MR. MOELLER: In the feasibility-desirability studies with whom are you doing these?

MR. HOBBS: We are primarily working with our architect-engineer, Bechtel.

MR. MOELLER: Well, I don't know if it is directly applicable; I believe it is. I just wanted to ask if you are familiar with the studies at the Sandia National Laboratory on cost-benefit considerations for filtered vented containment systems?

One of their conclusions was that based upon man-rem averted for total accident cost, containment venting appears to be potentially cost effective for the BWRs that they evaluated. And one of the BWRs they evaluated was Grand Gulf, although not in detail.

MR. HOBBS: We are generally familiar with that. We have not taken that under advisement as an active part of our design considerations at this time. Basically, we have been somewhat on hold and actively pursuing this because of the severe accident rulemaking and wanting to take a unified approach to handling that issue.

MR. MOELLER: I believe this study that I quoted is part of the severe accident rulemaking procedure.

MR. HOBBS: Yes, sir.

MR. MOELLER: Well, I would encourage you to keep up with that and keep abreast of what they are doing.

MR. OKRENT: Is it your thinking that were you to do this venting that the existing filtration system would be adequate for the amount of fission products you would have? I don't mean the noble gases but the other things.

MR. HOBBS: Our evaluation is really not complete. Obviously, having any kind of charcoal filter is better than not having filtration, but we are not prepared to answer that.

MR. OKRENT: So if I understand correctly then, this is a matter which is under study at this time?

MR. HOBBS: That is correct.

MR. SHEWMON: Is that all for that item?

MR. OKRENT: Unless there are other questions from the committee.

MR. SHEWMON: I don't see any.

MR. MARK: Does the staff have anything?

MR. OKRENT: You see, I first became conscious of the fact that there was some correspondence looking at the SER. There was a paragraph that said the staff had received two letters in June -- and I didn't know if they existed because I hadn't seen them; I might have missed them -- but that this was something they were still evaluating. And so at the subcommittee meeting we added it to the agenda, but in fact we didn't get their correspondence until this morning.

I thought we had better hear about it today just to have at least an exposure to it. And when you see the draft letter, you will see that I expressed a continuing interest on behalf of the committee in the subject; but it is not something I'm assuming we will try to resolve at this time.

I think then we are on the area of the agenda called "Other," and Mr. Kerr had a question. Do you want to state it again, Bill?

MR. KERR: My question was the methods used during the course of an accident to determine the source term which one must use to predict possible offsite doses.

MR. MC COY: Ken McCoy. I'm the plant manager at Grand Gulf. We have an emergency procedure called dosive assessment which, if I understand your question correctly, this is the procedure that we have operators at the plant use to determine the releases and to make our recommendations to state and local governments and take protective actions.

Is that what you address as the source term?

MR. KERR: Yes, sir.

MR. MC COY: The preferred source term is an accident monitoring system that actually reads the gas release rate and the iodine release rate in the standby gas treatment system discharge, and that is our first preference. If that is not available, we use a source term based on the study in the FSAR which assumed 100 percent release of the noble gases and 25 percent release of the iodine in the core in the worst case with a .37 percent in volume release per day from the containment. And the first 100 seconds of that is not processed through the standby gas treatment, and then after that it is. So what we do is we take that worst case source term and use that initially until we have better information.

MR. KERR: You have instruments which you think are capable of measuring the releases of noble gases and iodine if the release is through the standby waste treatment system?

MR. MC COY: That is correct. That is the only release point from the auxiliary building which surrounds the containment and the enclosure building.

MR. KERR: But if one had a very serious accident in which there was a leak say from the containment above the normal, whatever, .1 percent per day or whatever one has established, you don't have a method for measuring that, or do you?

MR. MC COY: Yes. It still goes through the same stack. It is just not processed by the filter train.

MR. KERR: Well, I guess I don't see how if you have a leak from a containment to outside you are certain that it goes through a stack. I'm missing something.

MR. MC GAUGHY: The containment is completely surrounded by this. There is no path from the containment directly to the outside.

MR. KERR: I had assumed that this treatment of accidents perhaps took into consideration the possibility that one might have an accident serious enough so that there would be leakage through both buildings. That is not the case, I guess. Your assumption is that leakage always occurs through the gas, the waste gas treatment system?

But I'm not trying to put words in your mouth. I'm trying to understand the process, because from reading the NRC documents I don't understand them. They should probably put more time on this, and I thought maybe you could help me.

MR. RICHARDSON: Normally, after an accident the containment is completely enclosed by the containment building and the auxiliary building, and that volume is then filtered through the standby gas treatment system. All of the other release points like the normal ventilation containment are also monitored with a high rate monitoring system.

MR. KERR: What do you mean by normal release point?

MR. RICHARDSON: Well, post-accident each release point was isolated. We are talking about during normal operation of other ventilation systems which exhaust the air and the environment in the buildings, which may be potentially --

MR. KERR: And if you have got a pressure buildup in the containment that was big enough so that you've got containment rupture and leakage, it would still go through the waste gas treatment system?

MR. RICHARDSON: The sequence would be that the containment isolates and that you postulate .35 percent of the volume per day leaves through the concrete in the building, and then it is taken -- it is then in the enclosure building, auxiliary building, and then it is removed through the standby gas treatment system where it's filtered.

MR. MC COY: But if there was a rupture in the containment, it would still be inside that enclosure volume, which would still be processed. We also have, as I am sure you are aware, two redundant trains completely independent; and we have not addressed the case, as I think you are asking, of if the normal ventilation systems were isolated and both independent trains of the standby gas treatment system were not operable.

MR. MC GAUGHY: But all that notwithstanding, we also have a radiation monitoring team out taking samples around the plant and in the plume to see that all of this checks out.

MR. KERR: I guess I trust that methodology more than all of the elaborate calculations I see. I was just trying to understand what one used to put into the calculations, because it seemed to me fairly important that one know something about the source term even if one does have very accurate meteorological dispersion available.

But I am with you. I think if I could I would get up there with a meter and do that to see what's going on.

MR. MC COY: Our procedures call for that, and also, we do have a much more accurate source term now because we do have high range accident monitoring in that exhaust which our earlier plans did not have.

MR. KERR: Thank you.

MR. MC GAUGHY: We also have an answer to Dr. Moeller's question at this time.

MR. RICHARDSON: Dr. Moeller, I haven't looked at all of the LERs to see what the staff has done, to see what the specific failures were. If I'm not mistaken, the number of LERs and failures are more for the type of hydrogen monitoring systems that I think that BWRs have had in the aux feed system, which is a kind of complicated process, chemical process.

A monitor that came up in the discussion of hydrogen control is a much simpler, much more reliable device. It is a thermal conductivity type of device, and to my knowledge there haven't been a lot of failures, and they are a very reliable device.

MR. MOELLER: Thank you. That is helpful, because the LERs do not -- have not made that distinction. Could the staff comment and confirm that for me?

Is the difference in the type of monitor the result of the staff's pressures or simply observations by utilities?

MR. SCHWENCER: We don't have an answer for that, Dr. Moeller.

MR. MOELLER: Could you let me know sometime?

MR. SCHWENCER: Yes, I could do that.

MR. SHEWMON: What was your question?

MR. MOELLER: I simply referred to the fact that there had been an increasing number of LERs that cited failures in hydrogen and oxygen monitors within BWR installations. The Mississippi group is pointing out to me --

MR. SHEWMON: I understood what you said. Your question for the staff, though, is?

MR. MOELLER: I was curious whether the fact that they are using a different kind of a monitoring instrument for hydrogen monitoring within containment than they do for the filter systems and recombiners and so forth, I wondered if that was the result of staff pressure or simply observation on the part of the utilities.

MR. SHEWMON: Whether there has been more failures due to staff pressure?

(Laughter.)

MR. MOELLER: No. Whether the use of a different kind of instrument -- why are they using a different type of instrument.

MR. SHEWMON: I see.

MR. MOELLER: And maybe someone has learned something.

MR. KERR: I would guess it might have to do with the concentration of hydrogen.

MR. MOELLER: I would like to know. It may be that the others are not -- that the sensitivity required here is not as great as in these others. Maybe the Grand Gulf people could help me with this. I mean I'm always in favor of progress.

MR. MARK: I wonder if they could let us know what is the readout time for the concentrations of hydrogen in the containment. Do the signals come electronically, or do you have some chemistry, or do you have to integrate them over half an hour or what?

MR. MC COY: Those detectors are direct reading, and the only thing you have is the response time of the instrument itself, which is relatively fast, an order of seconds.

MR. MARK: So you know the concentration of real time, where the detector is.

MR. MC COY: That is correct. And that instrumentation reads out in the control room.

MR. MOELLER: And why is it you are able to use a simpler, more reliable instrument here, or why don't they use them everywhere?

MR. RICHARDSON: The only response I could give you, Dr. Moeller, is that we evaluated the systems. We felt that this type of device was more reliable, and therefore, it would be better for use in the containment for accident monitoring. And I don't think it was any staff pressure or anything. There is a little bit of history, I guess, involved in that, and that is the best answer I can give you.

MR. SHEWMON: Dr. Plesset.

MR. PLESSET: Well, we were told that Dr. Tony Hurt was going to make an analysis for the applicant of the effect of intrusion in the air space above the wet well when you get the bubble rising and breaking through, and all I wanted was to ask if they could send me a copy of this calculation. Dr. Butler is not here, and he might send it along, but it might take a lot longer.

MR. SHEWMON: Okay.

Jess, did you have some questions?

MR. EBERSOLE: Yes. I had a few.

To go back into earlier meetings, gentlemen, you of course are aware that you can build pumps that you can use as dredges that can handle sand and mud, and the seals are designed for that purpose. On the other side of the spectrum you can have pumps which cannot handle any kind of fluid except clean fluid.

And I don't know what side of the spectrum you are on with the RHR spray pumps in this design. I know you have concluded enough not to use ordinary-type installation in this dry well, but that doesn't eliminate the consideration of paint and degraded concrete and other contaminants in the post-accident fluid stream that you have to look at and be sure are compatible; that is, that the level of contamination is compatible with the designs of the pumps and seals that you've used in the RHR core spray pumps.

Early on in an earlier meeting we asked you to go back and look at the cleanliness specifications for the fluid conditions that your pumps, especially the seals and bearings, were going to demand, and what you had done to ensure that you were going to have that sort of degree of cleanliness against the sump contamination level that you might have.

Have you done that yet, and can you tell us a little bit about how you have assured your pumps and seals will run on and on for say three months while you're handling a post-accident cooldown situation?

MR. TOWNSEND: My name is Hal Townsend from General Electric.

The RHR pumps, as we have told you, Dr. Ebersole, are the filters and their discharges to filter the flow into the seals. These are hydrocoat type filters that take out the large particles that might be present in the RHR flows. The RHR pumps themselves are deep well submersible-type pumps that are normally designed for irrigation-type service, so they have a long experience of being used with grit and sand-type flows in those pumps. And we think we have enough assurance that these will continue to run.

MR. EBERSOLE: What is the intake and discharge from the hydroflow-type pumps? We're not getting at the crux of the problem.

MR. TOWNSEND: Well, the intake to the hydro pump is the discharge of the pump, and that is fed back into the discharge.

MR. EBERSOLE: So what is presumed to go into the intake of the hydroflow, and what comes out, and how did you manage to ascertain that the contaminant was heavier than water so you could spin it out rather than probably lighter than water which I once found out it could be, which actually causes the hydroflow to feed contaminants?

How did you reach that rationale? In short, what is your contaminant list?

MR. TOWNSEND: I'm afraid I can't answer that.

MR. EBERSOLE: All right. We don't have a contaminant list before and at the strainers. We don't have presented any reliability evidence as to what the seals and journals can take.

My impression is that these designs are in the form of what one might call a final or ultimate filter wherein they act as collectors in the terminal context of whatever contaminants may be in that stream. I'm not so sure but what the hydroflows don't feed contaminants to these things. It would be true if the contaminants have a specific gravity of less than one.

I personally went through a little

experimentation to find out the sort of contaminants that are in this water could be in any portion of the section. It could be floating, semi-floating, or sinking. But it is a detail, and it is a critical detail in the flow pump operation which I think we ought to look at, and we evidently haven't.

MR. RICHARDSON: Mr. Ebersole, I might add one thing; that there is a list of post-accident contaminants that you might expect in the suppression pool, and any FSAR can evaluate that. I don't remember the specifics, but it is in there.

MR. EBERSOLE: Fine. And the next thing is what do you do with it? Where does it go and how do you keep them out of where they shouldn't go? It's just a question of logical evolution, and I don't see it completed.

What does the staff do about this?

MR. NOVAK: Tom Novak again. I think the concern you had we have probably addressed case by case. I can recall the North Anna application again, and it was a similar application where you had an extended shaft, and you had bearings, and there was a number of experiments that the licensee had to perform and modifications to the pump which went through an extended period of time until he convinced us that he had an acceptable design.

MR. EBERSOLE: Is that the case here?

MR. NOVAK: I can't speak to Grand Gulf, but it is handled, I would say, more on a case-by-case application, and we don't have a generic solution.

MR. EBERSOLE: Well, this is a case.

MR. NOVAK: I guess the thrust of it is that you look for the inlet to be a forgiving situation on the BWR-type containment. On the wet well I recall on some of the older designs that after you come down, you pull up, so you are not really dragging off the floor, or you are not really as worried about it as you are in a dry containment. There are some redeeming features in a BWR recirc mode that one would look at.

MR. EBERSOLE: On the other hand, the PWR design is, in general, the pumps have been designed to accept gross contaminant levels just like you would expect in a dredge of the cooling medium for the journals and virtually no influence on the journal function. It really gets down to an examination of whether you have materialized in detail what you need.

MR. BENDER: Why don't we suggest that it be looked into?

MR. EBERSOLE: Fine. Could we get formal evidence that you fixed this thing or looked into it?

MR. NOVAK: I'm sure the committee is aware of the generic work we are doing on large, dry containments. There is some work that we have done with the Alban Labs, primarily looking at debris, debris which originates from insulation material. We have not, to my knowledge, factored in the BWR consideration, but if the committee identifies this concern, I will make sure that we go back and relook at it. It is an unresolved safety issue.

MR. EBERSOLE: Well, one thing about the hydroflow built into this concept is the thesis that any contaminant has a specific gravity greater than one, and it is not necessarily true. A deep bed filter would, in my view, be a hell of a lot better; but that can be part of your investigation.

I would ask another little question. In your 1E, in particular, 1E DC systems, do you have any automatic electrical transfers which ultimately challenge the last critical supply source?

MR. MC GAUGHY: Could you restate that, please?

MR. EBERSOLE: Are there any automatic electrical transfers which ultimately challenge the last critical supply source that you are working with, and in particular, would you might have this true in the 1E DC systems?

MR. RICHARDSON: I may not understand your question completely.

MR. SHEWMON: I think he wants to give you an answer.

MR. RICHARDSON: In the DC system there's no transfer-type devices like from one supply to the other. It is just batteries and supplying its loads.

MR. EBERSOLE: Do your major electrical boards have multiple DC buses inside them? If the staff would look at this -- you know, these things have been condemned because of the potential for cascading to a terminal failure of all of the DC systems. This is ongoing in a more generic DC study. These transfers have a threatened viability at the last source.

MR. SHEWMON: Is that your final question?

MR. EBERSOLE: I see a little conference. Otherwise I'm done, Mr. Chairman.

MR. RICHARDSON: Mr. Ebersole, is your question the separation between divisions?

MR. EBERSOLE: No. It's whether you execute an electrical transfer in a progressive way.

MR. RICHARDSON: No.

MR. EBERSOLE: Let me ask you as a matter of routine -- I have to deal with general matters -- in your examination of service supplies, including AC and

DC power and water and air and so forth, do you as an applicant investigate the case of potential excesses of such services as well as the so-called failures of same? And in examining the failures do you examine them in the context of gradual or intermittent failure as well as simply a gross failure?

One case in point I can think of is a gradual error failure on a scram system.

MR. MC COY: Yes, we do, on the air system. We actually ran a slow loss of air test and discovered some problems which we have alerted the industry to.

MR. EBERSOLE: Do you also examine the usual control type of failure that would result in the excess system pressures such as excess air pressure and excess voltages and so forth, and are you prepared to meet excesses of supply?

MR. MC COY: I have to answer you in specifics, and yes, we did explore excess voltages, and we did explore slow loss of air.

MR. EBERSOLE: For many years I've never really satisfactorily believed that the main feedwater flow check valve, the vertical valves, would survive a pipe break, hypothetical pipe break upstream of that, and I was pleased to find the presence of hydraulic damping devices which gave me confidence that somebody would look at the problem; and in fact, they finally got a valve in place that was going to come down with appropriate velocity and intercept and reverse flow of main feedwater in the event a pipe break should occur.

Question number one: do you know the consequences if you have unimpeded flow of feedwater back into your station on the primary vessel in case you don't intercept the flow so you know what the terminal consequence is?

And second, with what degree of consequence and on what basis did you think your valves with such hydraulic devices are going to survive the enormous structural loads they will experience?

What is the basis of your confidence?

MR. RICHARDSON: Let me try and answer the second part of your question first, Mr. Ebersole. We have an analysis under way to evaluate the effects to first of all determine the pressures and loads you expect for those situations. We are looking at the piping system that leads out to those check valves, and in my experience a significant loading during that situation that you represented. The valves themselves, there has been some testing, I think some Swedish testing, under those particular cases with some valves. I don't remember the particular size offhand. But they have demonstrated that the valves maintain their integrity.

In our analysis we will be giving the vendor the particular loadings on those valves and asking them to certify their performance. Our particular design does not have hydraulic cables.

MR. EBERSOLE: Does the staff have any observation on this matter? You know we have a generic problem, but in this particular case it is rather important.

MR. NOVAK: I have nothing specific to add to that.

MR. EBERSOLE: Earlier on in one of our meetings one of your electrical people gave a presentation which reflected, at least in my view, a completely unrealistic reliability of the AC power system. The reason that turned out to be that way were two real reasons. He included that tertiary system that you have got that is independent of the core spray. He included that as part of a package called AC power. And secondly, he rather completely eliminated consideration of any common mode failure. In short, that analysis was altogether unprofessional.

MR. OKRENT: Jesse, are you sure it was Grand Gulf or some other applicant?

MR. EBERSOLE: I hope it was Grand Gulf.

MR. OKRENT: No, I don't think so.

MR. SCHWENCER: It was my recollection we discussed this on Perry, Mr. Ebersole.

MR. STAMPLEY: I hope our presentation was more professional than that.

MR. EBERSOLE: Excuse me for making that mistake.

Now, regarding a discussion we had yesterday, I'm going to make a summary statement here and you can shoot me out of the saddle if it's wrong.

The applicant claims that following any "accident which requires mitigation there will always be subsequentially such mitigating capability as to allow a single random active failure in the mitigating system without disabling the mitigating function." That is, I believe, a true statement.

Would you agree?

MR. RICHARDSON: That's right.

MR. EBERSOLE: He has not, however, verified the need to use coincidence or confirmation to prevent damaging safety system response to spurious signals. He will examine this problem and provide a written response at a later date.

Is that a fair statement of what you agreed to do?

MR. CESARE: Would you repeat that last part?

MR. EBERSOLE: He has not, however, verified the need or the lack of need, for that matter, to use coincidence or confirmation to prevent damaging safety system response to spurious signals such as he might derive from an interrupted sensor line.

MR. CESARE: I think that is a fair summary. Our initial survey is that we have, but we will give you a written response.

MR. EBERSOLE: All right. Thank you. That is all I had.

MR. SHEWMON: Would you like to summarize then?

MR. OKRENT: Well, the only point I will note is that it seems from this last set of questions that Mr. Ebersole may be interested in having some words which would ask that a look be taken at whether the various kinds of debris have been looked at hard enough with the possibility of causing difficulties. That is my interpretation, and he should correct me if I am wrong. If my interpretation is correct, he should prepare some words that he thinks are suitable, and he should do that within the next couple of hours, since we are supposed to get to this matter at the end of the day.

I don't have anything further to add, Mr.

Chairman.

MR. SHEWMON: Well, it is traditional. Does anybody have a reason why we can't write a letter on the proposed advance 100 percent power, and if not, I will take it that we can. And we might adjourn the meeting and then reconvene it in closed session in about five or ten minutes, and we will take the last items of today on the agenda and move it up.

(Whereupon, at 11:45 p.m., the meeting was recessed for lunch, to be reconvened the same day.)

AFTERNOON SESSION

MR. SHEWMON: We now turn to severe accident research plan, which has come before two subcommittees, Kerr and Okrent. Kerr will begin.

MR. KERR: Thank you, Mr. Chairman.

We have just received a handout that is relevant to this discussion, which is a discussion of the document NUREG-0900, with a number of revisions, entitled Nuclear Plant Severe Accident Research Plan. The document is in a form labeled Draft Revision 2, made available to the ACRS in early April. It was sent by covering memorandum from Dircks to the Commission, which was described as a paper containing plans for producing research information needed to confirm regulatory decisions in the severe accident area, including methodology for preparing the cost of new requirements with a risk reduction, and generalized reduction in the uncertainty of the PRA.

I am sorry. This quoted a d rective from the Commission which said that such a paper should be provided to the Commission by February of 1982. This 0900 is a response to that directive from the Commission.

The document provided some of the bases for ACRS comments contained in its letter of July 14, 1982, on the NRC research plan for fiscal 1984 and 1985. The comments from ACRS included statements indicating that there was a need for a better identification of the questions likely to arise in the process of formulating an approach dealing with severe accidents, and the Committee also suggested that it would be wise to design a research program to correspond to that approach once it is formulated.

The Class 9 subcommittee met with members of the research staff on May 28th to discuss NUREG-0900. At this meeting, the subcommittee expressed a need for better correlation of the proposed research with some approach or possible approaches to dealing with severe accidents. The staff agreed that additional explanation of its proposed work and its relationship to an approach would be helpful, and at a subcommittee meeting on August 6th, provided the Class 9 subcommittee with revisions to the original draft as an attachment to memorandum SECY-82-203.

You have a copy of that as part of this handout. This does not constitute a complete version of 0900, but rather the revisions which, when appropriately correlated with the original document, will constitute a revised document.

Now, let me read briefly from the Dircks memorandum, which says that, "Since the Commission's proposed safety policy statement and guidelines" -- this is NUREG-0880 -- "the severe accident research plan 0900 was formulated to develop generic bases to determine how safe they are, and engineering guidance on where and how their level of safety ought to be improved. The analysis to address these should be performed using improved probabilistic risk assessment methodology as benchmarked for exact data and analysis."

In parallel with this activity, the EDO transmitted a memorandum to the Commission labeled SECY-82-1, dated the 3rd of January of 1982, and made available to the ACRS also in January of 1982, and comments on that document were sent to the Commission by the ACRS in a letter dated February 8th, 1982. This memorandum proposed to deal with severe accidents in the process of licensing for standard plants, and let me read from that document, if I can keep this set of documents that I have straight.

"This policy paper," we are told, "summarized various changes," that is, in 82-1, and this is 81-1A, "various changes in rules, policies, and regulatory practices that collectively supported a change in the approach for severe accident rulemaking. The change was one of, according to this, the new approach would replace the unfocused, long-term generic rulemaking with severe accident rulemakings on specific standard plant designs and regulatory decisions on classes of existing reactors which may or may not include rulemaking."

So, the emphasis is on standard plants with regulatory decisions on existing reactors, maybe or maybe not dealt with by rulemaking. As a result of the comments and questions raised by the ACRS and the Commissioners on the original 82-1, the memorandum was revised, and a version labeled 82-1A, which I just read, but this language that I read was also in the 82-1, has been prepared for the Commission's consideration. This was also distributed to the subcommittee, I guess, slightly before the August 6th meeting.

Let me indicate a few key points in this version which may help provide a context for consideration of NUREG-0900, and I read from Pages 15 and 17 and 18 of 82-1A, on which we find, "We do not expect," says this document, "our present views on severe accident considerations to change substantially as a result of ongoing NRC sponsored or industry research, with respect to the fundamentals of the present designs and their adherence to our safety policy. However, we expect the research results to identify further worthwhile refinements in the design.

Research will also help to develop more accurate probabilistic risk assessment methods for use in regulatory decision-making, and to provide greater assurance of adequate protection of public health and safety."

Then, on Page 17, "The Commission is considering the question of whether additional regulations should be issued at this time to require more capability, to mitigate the consequences of severe accidents in operating reactors and plants under construction. Although there are large programs presently ongoing that will provide information related to this question, they have not yet produced significant new insights into consequent mitigation features sufficient to support further regulatory changes, nor have they shown a clear need to add such features."

I think that is probably what I had in mind there. At this point, we also recall certain efforts toward development of safety goals, and since 82-1 proposes to use the safety goals in dealing with severe accidents, their present progress and direction are important.

At a meeting on August 6th, the PRA subcommittee, which met on the morning of that same day, discussed the draft of an implementation plan, and you will remember that the Commission after receiving 0880 asked the staff to prepare an implementation plan, and the one that we had was dated June 25th, 1982, and among the issues discussed are some that would seem to bear on the research program described in NUREG-0900.

Now, again, if I can distinguish among these, in dealing with the implementation, we find in this document the Commission's safety goal explicitly includes the risk from external events except in sabotage, when it does not require that such risks necessarily be quantified at this time.

PRA is useful in a relative evaluation of various structures against deterministic criteria in making realistic evaluations of the strength of existing structures, and in providing greater confidence in the decisions regarding estimates of the relative seismic hazards. However, the uncertainties associated with applying the PRA to the external hazards at this time are much greater than associated with the estimates of the hazards from internal sources.

Therefore, there is difficulty in using the estimates for seismic events, floods, and fires, and comparing them to estimates from adequate risks of internal hazards. However, the NRC believes that sufficiently low levels of risk attributable to external events can be achieved by applying NRC's current deterministic criteria. For this reason, the numerical guidance on the likelihood of core melt presently to be apportioned between external and all other accident initiators under the assumption that their contribution of external events to core melt frequency.

For this reason, numerical guidance is to be apportioned between external and other accidents under the assumption that core melt frequency is generally low, provided NRC's deterministic criteria are met, so that the assumption in here is that it will not be quantified, but it will be assumed to be generally low compared to internal causes.

The Commission's goal does not establish the numerical guideline on the availability or performance of containment structure. Whether or not such a numerical guideline is eventually established, the containment performance under core melt conditions would have to be better understood in plant specific evaluations if individual and societal risks are to be effectively and consistently used in the plant specific safety assessments.

Now, having said that, we then go to Page 23, I think it is, in the SECY-82-1A, in which we find it is clear that core melt accident evaluations and containment failure evaluations should continue to be performed in the context of probabilistic risk assessments for a representative sample of operating plants and plants under construction for all future designs. These studies should improve our understanding of the containment loading and failure characteristics for various classes of facilities, and the analyses should be as realistic as possible, and should include where appropriate an adequate schedule for loading and so on.

Yes, the implication is that although they aren't going to be used in the determination of meeting safety goals because of their inaccuracy, they should still continue to be made for a certain number of individual plants, because we will learn a good bit from that, and improve the technique.

In addition, we find that in addition to energy absorption capabilities mentioned above, several features may decrease the chances of containment failure for some accidents and some containment designs are listed in Item 2, II.B.8 of the TMI action plan, namely, filtered venting containment, hydrogen control features, and core retention devices. The NRC has been studying these and other mitigation features, and is now in a position to give the following preliminary guidance about them for the design of the plants during the construction permit applications.

I recognize all of you have read these things, and I am just trying to refresh your memory. Then, on the filtered vented containments we find the following guidance. Future applications for both BWR's and PWR's, filtered vented containment systems or variation of such systems should be provided if these yield a cost-effective reduction in risk. Some recent information indicates these systems may not be cost effective for large dry containments, while other studies indicate these may be of value for some suppression containments such as the Mark III design of GE. GE has also considered a wet well vent for standardized Mark III design. However, these preliminary conclusions need to be addressed, and final conclusions reached for new design before they are applied to future plants.

Core retention devices. Over the past several years, studies of large reactor containment buildings indicate that the classical core retention devices are probably not cost effective in reducing the atmospheric release of radiation. Post-accident flooding of the reactor cavity may be all that is necessary to establish this. However, the designs or the unique or desirable pathway characteristics should be carefully weighed in future CP applications before deciding that this concept can safely be dismissed.

Also, the materials of construction can reduce or eliminate aerosols and combustibles and non-condensibles. These contribute to the overpressurization threat to the containment building integrity and should be considered in an integrated evaluation of the adequacy of containment performance. That is the guidance on core retention devices.

Well, I have tried to indicate some of the things in these various documents that bear on the problem in the staff's present approach without commenting very much on 0900. You will hear, I think, a staff presentation on the present version of 0900 and I sort of tried to put things in context. I will be glad to try to answer questions from other members, and other members of the subcommittee may also want to comment.

MR. OKRENT: I have some comments, unless the others would like to speak first.

MR. SHEWMON: If I might perhaps do a disservice, but summarize things, I would like to summarize the situation as I heard it, and would like to comment on it. The old plants seem to be okay, or at least we don't see that we need to rush to change at this point. We will look hard at the next generation of proposed plants, though the only indication of the changes needed in new permit plants might have to do with checking the effectiveness of filters, vented containments, or other ways to cool rubble when it is beneath the reactor, and out of this the staff gets the clear message that they need a large research program on development and modeling so that they can better model Class 9 accidents after, so when they do get ready to write these new regulations. Is that correct?

MR. KERR: There is a two-phased approach to research that says in about two years we will make a decision, but the 82-1A seems to apply, that if they have to make a decision now, it would be that perhaps minor changes would be needed in existing plants, but no major ones, and then there is a follow-on period of research for perhaps another three years, if one looks at the 0900, and that will be confirmatory.

MR. SHEWMON: Confirmatory that we probably don't have to do anything?

MR. KERR: Well, confirmatory presumably that the decision that was made was a correct decision.

MR. SHEWMON: And the other thing that was -- the classical, whatever word that was used for under the core structure, I take it classical refers to magnesium oxide crucible like we had in floating nuclear.

MR. KERR: I guess so, Paul, as far as I know.

MR. SHEWMON: That is the only one I know of.

MR. KERR: I think so.

MR. SHEWMON: That is all I had.

MR. OKRENT: Well, let me make a few or several comments. First, I might note that 82-1A was issued at least to the ACRS after the July ACRS meeting, and I for one found it interesting that the forwarding memo from Dircks to the Commissioners proposed that they put it out for comment before the August ACRS meeting, since it seemed to me to be a fairly important proposed policy position, even if it was only being published for comment. I found the proposal by Dircks incompatible with the recommendations of the various groups about the role of the ACRS to be strengthened.

Fortunately, it is my understanding that the Commissioners reacted for whatever reason that they wouldn't deal with this before September some time, but nevertheless I think the fact that the recommendation was made is almost astonishing to me. Nevertheless, it exists. We also have in hand a second proposed policy statement on safety goals, and of course we have the proposed implementation plan. I would like to urge that the Committee give priority to this trilogy at the

September meeting, the October meeting, at special meetings, whatever it takes.

In my opinion, it is more important than almost anything else that we might have to do. In fact, I wouldn't rule out deferring a case unless they were waiting on the baited breath, as it were, and needed a good reason for delaying it. So, I urge that you give serious attention some time during this meeting to that question.

MR. SHEWMON: Is 82-1A something I have sitting in front of me, if I only could find it?

(Pause.)

MR. OKRENT: Now, I find a kind of schizophrenia in the staff, and maybe it is me, but on the one hand, Mr. Ernst, if you will, preaches against the bottom line syndrome, saying, or using PRA, let's be careful about spacing everything on the risk numbers that come out at the end. It is going to be more useful in other ways. But in 80-1A, if you listen to the kind of tentative conclusions that Dr. Kerr just read, there were conclusions that these plants are okay. They are safe enough, and they meet the safety goals, which to me is the ultimate use of the bottom line.

Well, I just mention that for the moment. You can reflect on it. Let me just for a moment pose questions concerning some of the tentative conclusions that the staff seems to be drawing in 82-1A, and see if you think you understand that, or if maybe they can explain them, as to why they feel so relatively assured with regard to let's say the existing plants. I am a little bit at a loss to understand. For one reason, I haven't seen the documentation that is behind the conclusions. Perhaps they have them. They make a very good case for them, but I haven't seen them. I am aware of a variety of things, and I will just tick some of them off.

There was this recent study on precursors, which I am not endorsing. In fact, I have some questions on some of the methodologies, but nevertheless it wasn't in the direction that core melts were very infrequent. We have said many times that the existing PRA's are incomplete, they don't include many potentially important contributors, even though they could have dealt with external events. And as one example of how one's conclusions can be changed by what I will call not completely implausible changes in assumptions, if you take the study done at Zion in which they evaluated the risk reduction factor for a filter vented system put on their containment, whatever the original risk is, and if you, instead of assuming that they did like they did, that its likelihood of damage in an earthquake was fairly large, in other words, that it was not designed with a large seismic capability, but with a capability no better than, if I recall correctly, the fueling water storage tank or one of the components that had a lesser look.

If you assume it is good as the piping in the primary system, you can change the efficacy of the same system by a considerable amount, perhaps a factor of five, so it might go from three to fifteen, and these are round numbers, and at the moment it is not clear to me in fact why that system cannot be designed to be as good as some of the better seismic systems there.

Another kind of example where changes in assumptions can be important, if you take again in the Zion PRA, their dominant risk events, which is a seismically caused core melt, and a delayed containment failure, where some other heat removal system might help, if you double what they call their systematic uncertainty factor, beta, which is provided by expert opinion, it is not as if we have a lot of data on fragilities. My students recently estimated the core melt probability goes up by roughly a factor of ten, and my understanding from conversations with PLG is, this is reasonable. That is about the change one would get.

But just changing, increasing the uncertainty, and so you get a bigger overlap of two probability distributions, and similarly, if instead of using the seismic passive curve, let's see, that compares big earthquakes, that was used there. You use the one that was used in WASH 1400, you get another factor of ten. So, these are not small numbers in what was the dominant risk scenario, and of course they would also change both the efficacy and the cost effectiveness of various measures that either reduce the chance of this event occurring and leading to core melt or mitigating core melt.

With this kind of thing in mind, I can't tell what the basis is for the staff's seeming conclusion in 82-1A, and I think the Committee needs to learn more. Now, as was mentioned in fact at the subcommittee meeting in the morning by Mr. Ross on last Friday -- I am sorry, not Mr. Ross. Well, you are right. It wasn't Mr. Ross. But in any event, there are measures being taken for whatever reason in other countries, let's say, dealing with PWR's, both in the area of preventing core melt and in mitigating core melt, which in many cases provide increased efficacy over what we have in our existing PWR's, and it seems to me there is a minimum. Before the staff arrives at some conclusion, they ought to understand what these are all, what the reasons are, and why if these are not relevant with the U.S. to do and so forth, and that that should be up front, that the conclusion may still be the same, but what I am saying is, at least we ought to know of things like this.

Just a couple of other points, and then I will stop. In SECY-82-1A, one of the sections that Dr. Kerr read from indicated a trend, let's say, toward thinking that post-accident flooding may be very effective for a PWR. In fact, I think this is an interesting idea. I know in the ACRS there has been interest in having this pursued, and I think it is interesting to see what has arisen in the Zion PRA, but I am not sure anyone has seriously assessed the pros and cons of this compared to other possibilities, and on some overall basis judged that in fact there aren't some possible negative features to this, features that could lead to what you might call a PWR 2 type release, maybe with a low probability, but nevertheless a big uncertainty in this low probability, enough that it becomes less of a clearcut conclusion.

What I am referring to is, can we rule out some large pressure pulse combined with something or by itself or whatever? Do we know enough about it? I hope it comes out very well. But again, one has to be a little cautious about jumping too far in the direction that he would like to jump. That is all I am saying.

And just one last specific kind of thing. We have heard from the staff in connection with their discussions on siting policies which in the end will mesh in with this, that they don't see differences among seismic, east versus west, with regard to their calculations of delayed effects. They all tend to look rather similar. The limited studies that I have seen done, aside from the staff's, leads me to be less convinced that this is indeed the case, and that the factors may be considerably more than two among the sites currently in use.

And so there are all these kinds of things where I for one am apprehensive about the way the staff has seemed to suddenly have arrived in positions and positions which may not stand up technically, and may not stand up politically when there is a change in administration or something.

That is the end of my comments. Dr. Kerr, do you have any comments on what I have said?

MR. KERR: I have no comments.

MR. SHEWMON: You had things you wanted from the staff?

MR. KERR: The staff asked for, and I

certainly approved of their making a presentation to the Committee on 0900. I think it is usual for them to present to the full Committee before we write a letter. The agenda is rather short, and is a presentation by Mr. Ross and his designees of what he wants to say about the documents. Appropriate?

MR. SHEWMON: Dr. Ross.

(Slide.)

MR. ROSS: This describes the severe accident research program.

(General laughter.)

(Slide.)

MR. ROSS: I understand that the deadline is 4:15, but the question is, is there any pressure to contract that time?

MR. SHEWMON: No, but there is pressure to meet it.

MR. ROSS: We will meet it.

MR. SHEWMON: Actually, we are giving you 20 minutes to start. You say you are going to need all that time?

MR. ROSS: At least.

All right. The definition of the severe accident research plan is that this report number, which consists of the main portion of SECY 82-203, with some amended pages that are coming down here the same way that their 203, and that is, it will be by memorandum that Mr. Dircks we hope will sign, and I gave you an advance copy. The amended pages are undergoing final review in the office of NRR. I can't preclude that there will be some changes, although we have assurance that NRR is in substantial conclusions and substantial agreement with this in the form you now have it.

(Slide.)

MR. ROSS: Going to the next slide, there are some related items which we just got through discussing. I want to point out that with respect to either 1A or the safety goal, that like Mark Anthony, we are not here to either bury or praise, either one. That will be left to others.

(Slide.)

MR. ROSS: We received -- that is, the staff received near the end of January of this year what is generally referred to as a staff requirement memoranda. This is instructions from the Commission. They instructed us as they commented on the original version of SECY 82-1. There were some particular things that were listed for the Office of Research. They told us to make sure the IDCOR effort continued, that we ensured that the necessary research critical to the approach, meaning severe accidents, continued, and they also gave us a deadline with respect to producing the technical information on operating reactors, both from IDCOR and NRC, by mid to late 1983.

I didn't put Point Number 3 on, but it is in the staff requirements memoranda.

(Slide.)

MR. ROSS: The general purpose of the plan is to develop generic answers or bases to determine how safe plants are and where and how they ought to be improved, and clearly, this is referring at this juncture to operating reactors, and I believe that if you refer to a document previously cited, although, like I said, I am not here to either praise or condemn it, SECY 82-1A, Page 15, there is a clear impression if the Commission in fact endorses this statement, that the Office of Research is supposed to gather information by the end of '83 that could and perhaps would be used on operating reactors.

(Slide.)

MR. ROSS: Now, the three ingredients that I would just mention, how safe should plants be will have to come from the safety goals. The other two ingredients will have to come from the research program. How safe are they, and how do we make them safer, if indeed they are not as safe here as the safety goal, in whatever form it comes out, says they ought to be. Well, you will use risk assessment methods to see how safe they are, and different techniques on how to make them safer, and an important portion of these techniques would be value impact theory or cost analyses as well as risk reduction analyses.

(Slide.)

MR. ROSS: Some objectives of the severe accident plan, and I won't read all of these, I would like to emphasize. On methods, we are talking about methods for accident evaluation. Detailed methods would include using computer codes such as RELAP for the thermal hydraulics, detailed core information from codes such as SCDAP, S-C-D-A-P, primary system details using TRAC melt, and details of the containment using the contained core code, among others.

Fast running methods, the so-called risk codes would be the MARCH family, as replaced by its successors eventually coming down to the MELCOR, the final version. The next to the last bullet obviously, since we won't have a complete PRA for every plant, we are talking about surrogate plants, and the last bullet, risk reduction potential, we are clearly doing both prevention and mitigation.

(Slide.)

MR. ROSS: Now, the next slide we spent some time on in the subcommittee last Friday. It is a decision analysis process. Some of the acronyms from the present state of knowledge, we are looking at probability of core melt and risk. Given a safety goal as input and not determined by this program, but given as input, you would compare what the plant has with what it ought to be. This is how safe should it be. And if you come out in this decision point or decision branch either acceptable or unacceptable, you look at different modifications.

This is either core melt portion of the safety goal or the consequence portion, and then you can come down to acceptable, going down the ALARA track. You need value impact theory both here and here. Along the line we would be looking kind of at the research as to its risk reduction potential also.

(Slide.)

MR. ROSS: Again, one of the topics on this slide we had considerable discussion on Friday. We talked about what you would need. Some of the research would help support a safety goal. Certainly it would be related. You can look at a severe accident plan and say, by subelement entitled Research or Safety Goal, it appears that a lot of the research would be clearly relevant to the safety goal as we now understand it. I think the key word here is "reasonably accurate". The discussion we had Friday is how accurate is accurate enough, and we had to confess up that we don't have a precise answer. I think we will know one when we see one, and a lot of the comparisons on what reasonably means will come when we start comparing calculations with detailed codes versus calculations with the so-called risk codes for the same initial environment conditions.

(Slide.)

MR. ROSS: Okay. Two portions of the PRA were emphasized. One is the likelihood, and we previously mentioned the precursor study, but as far as getting better or reasonably accurate complete PRA calculations, we will be improving the model and the data base and comparing it with such things as the precursor study or operating experience that would come in here.

With one exception, we don't expect to generate a new PRA, and we do hope to do a BWR Mark II to add to the PRA's that presently exist. Other than that, we would be improving what we have with better data and better techniques.

(Slide.)

MR. ROSS: On the consequence side, we will be doing consequence calculations with better risk codes, with present MARCH CORRAL family leading into MATADOR, and eventually to MELCOR. We will also be for some sequences, for some plants, we will be doing detailed calculations that are comparative experiments, and then these codes and these codes get compared and this is where we hope to say we have a reasonable and accurately complete method.

Now, we expect a lot of transference between this and this as the models develop. For example, the zircaloy steam reaction models developed here. It may be possible to lift these out and put them directly in the risk code without unduly lengthening the run time. To the extent that can be done, it will.

MR. LEWIS: Did you define accurate while I was out of the room?

MR. ROSS: No. I said I could not define it, not in terms that are easily quantifiable. It is like finishing the Sistine Chapel. When you are through, you know it. And I don't know any better way to put it than that. I don't think any code developer can really answer that question.

MR. MARK: Denny, if you can't get a fast running zirc water reaction code that is any good, are you going to go ahead anyway?

MR. ROSS: I don't know what the answer is, fully and completely. We know that some of the codes run in minutes and some run in hours, and if you take -- if you start taking pieces of the detailed codes and putting them in the risk codes one by one, pretty soon you have got the detailed code that runs along time, and it is not useful. I think you will have to look at the models case by case and see which ones you can move in and which ones you can't.

MR. MARK: I think there are some that run in minutes.

MR. ROSS: Pardon me. If the detailed codes would be as good as the experiments say they are, absent any experiments for the detailed codes, they don't have any credibility, either. I am not sure what the point was.

MR. OKRENT: The experiments may destroy their credibility.

MR. ROSS: They could. It wouldn't be the first time.

MR. MARK: You made an implication that unless the code ran fast, you wouldn't use it.

MR. ROSS: Sir?

MR. MARK: You seem to have implied that

unless the code ran quite fast, you wouldn't use it.

MR. ROSS: In some arenas, that is true. The risk code becomes ineffective as the run times increase. How can you do hundreds of runs with a code that takes several hours to work? So you would tilt toward conservatism. Let me ask Mark Cunningham about how long would it take to do a typical MARCH CORRAL calculation at present, if you know, and if you don't, ask Gary.

MR. CUNNINGHAM: I know the MARCH calculations are a matter of a few minutes, very, very fast running code. The MATADOR calculations, I am not sure, but they are still much faster than many of the detailed codes.

MR. ROSS: Now, George, how long does SCDAP take to run?

MR. MARINO: It is designed to take about 15 minutes, but that is just part of the whole analysis.

MR. ROSS: I think this area, we are still feeling our way around. We are not sure how much of the detailed code. We do know that in the thermal hydraulics stuff which MARCH is going to provide, the TRAC family could be very long, whereas for a TMLV prime you may be talking about several hours.

MR. MARK: I am thinking of a paper of a few years ago, a careful paper, I believe, a long study by

Marino at RES where he brought into attention the fact that if you didn't take account of the transfer of energy by radiation, you didn't take account of the real amount of steam that was there, then of course you got nonsense out and you got a very fast.

MR. SHEWMON: Onward.

(Slide.)

MR. ROSS: Slide 11 is just the outline for the plan.

(Slide.)

MR. ROSS: Most of the rest of this discussion will focus on Chapter 5 of the plan, where most of the technical material is, and we have in Chapter 5 13 research elements, and for each element we describe the element, the issues to be resolved, the interfaces, the background, and the plan of work as a function of time.

(Slide.)

MR. ROSS: The 13 program elements are as stated here on Slide 13.

(Slide.)

MR. ROSS: Now, the next discussion, I want to talk about the risk family, which is 5.1, 10, 11, 12, and 13, and do those before we get into the rest of the program elements. An overall view on the risk sections shown here on Slide 14 shows the five ingredients, sequence probabilities, the risk codes. This is the MARCH MATADOR family. The risks stated, the plant, the value impact, and the final conversion, not of just the risk elements, but also the detailed elements, into what is called regulatory analysis, which is a catch-all thing which would produce such things as guides, standards, standard review plan elements, or perhaps another Commission, SECY-84-1B or something like that. I think in the interest of time, we should skip to Slide 18.

(Slide.)

MR. ROSS: As you page through there, if you have any questions you want to bring up, we can go backwards, and I will try to condense. Slide 18 would emphasize two levels of research, a short-term level which we hope to finish very soon within the year, which we are updating the MARCH CORRAL family, converting it into MATADOR, and taking into account what we can in a reasonable period of time, like the first of '83, which would be -- and Version 1 would be used then.

For the 1983 assessment of value impacts in parallel development we are working on modular system of risk codes called MELCOR, which would incorporate all three of MARCH, MATADOR, and CRAC. It has a modular structure, so we can do what I just mentioned. You can take out, for example, consistent with run time, the metal/water reaction and put it in another one, and you can do something that is more narrowly a best estimate assessment.

MR. MARK: What is meant on that by the wonderful phrased, "improved deficiencies?" Do you mean they are more deficient than others?

(General laughter.)

MR. ROSS: That is a good question.

MR. BENDER: Denny, before you take that off --

MR. ROSS: I would like to take it off in a hurry, if I could.

(General laughter.)

MR. BENDER: If you take the "d" off of "deficiencies," it will be okay, but I would really like to know in what way MATADOR is going to be a better kind of computation. What are they really doing to make it better?

MR. ROSS: I don't know if you want me to take a crack at that or not. Is there one of the slides you want me to put up?

MR. CUNNINGHAM: There is not really any slide that goes into this. Is your reference to the difference between what will be in MATADOR as opposed to CORRAL?

MR. BENDER: Well, why is MATADOR going to be better than CORRAL and MARCH have been? In our previous discussion, when we have said that they tend to distort the problems instead of describing it.

MR. CUNNINGHAM: Well, MATADOR is simply going to be a replacement for CORRAL. It is not going to treat the phenomena that MARCH treats.

MR. BENDER: Well, I really was not trying to -- just tell me why the analytical -- this new code development will represent a better picture of what the actual phenomena are, and what makes it so worthwhile to push it.

MR. CUNNINGHAM: The MATADOR, within the MATADOR code -- let me back up. In the CORRAL the treatment of the removal of radioactive material in the containment building was basically a semi-empirical fit to CSE data. There was a concern in a couple of areas about that, that one, with that kind of formulation you could not account for large amounts of inert material being introduced also which the experiments over the last couple of years had suggested would be coming off along with fission products.

Also, it doesn't account -- it did not account for specific removal mechanisms which people thought to be potentially important, so the MATADOR version has been developed to include these additional removal mechanisms to account for the inert materials.

MR. BENDER: Well, maybe I haven't asked the question properly. I know there have been some refinements, but are the refinements enough to give you great confidence that the new codes are a significant -- more than significant, I guess, but a better and useable representation of the behavior of the containment system as it relates to these radio nuclide movements?

MR. ROSS: Mark, let me interject here, and Mr. Bender, if you would advance to Slide 45, you will see some more information where the contained code which is supposed to be an integral calculation of thermal hydraulic and fission products, and I think that is the detailed containment code, the aerosol code that we would benchmark MATADOR and eventually MELCOR, too.

MR. BENDER: You can broaden it as much as you want to. It doesn't change the question.

MR. ROSS: No, but I am saying that what we refer to as truth, the best estimate assessment of containment response in terms of fission products would more likely be in the contained code than they would be in MATADOR, MATADOR again having the virtue and the vice at the same time. Now, many of the settling mechanisms as I recall that are in MATADOR are not in CONTAIN and vice versa, but CONTAIN still has more virtue.

Bob Curtis, did you want to say anything more about how we are going to get to the heart of things in the CONTAIN code? Because I think that is really a thing you ought to focus on.

MR. CURTIS: In the CONTAIN code, we are programming with a modular system in which we can treat individual phenomenon in the best way that we know how, and we are trying to couple this code development with phenomenological experience, so that the people writing the code know as much at least as the experimenters know about the phenomena they are describing, and as we know more, and it becomes a significant improvement by virtue of the experimental work that has been done, we will tear that module out and put in one that better describes what we are talking about.

MR. BENDER: Well, I think that is not quite the answer to the question that I was asking. Will the new module that you plug in when you take the other one out, will it be enough better so that we will be able to say now we can really describe the way in which these containments behave, and we will then know what the radio nuclides are that are getting to the edge of containment that will then be distributed somewhere by an analysis using some form of the CRAC code?

MR. BERNERO: You may recall in the context of the Indian Point analysis mostly done by Speece and Meyer in NRR there was a curve that some of us starting calling the mosquito curve. It was a plot of aerosol in suspension, as a function of time, calculated by a number of codes, and it looked like a side view of a mosquito on your arm about to sting, and all of the curves were essentially congruent in the short term. They all went up, and you could use a short, a fast-running code, or a detailed code, and get essentially the same result. However, the mosquitoes with hind legs, all of the -- a large family of -- I will call them phenomenological codes, longer running codes, were essentially showing decreasing aerosol as a function of time. CORRAL, the risk aerosol code, or fission product transport code, was all by itself, and the farther out in time you went, the more divergent it was from apparent truth.

(Slide.)

MR. ROSS: Here it is.

MR. BERNERO: The risk code which is the upper line.

MR. ROSS: Except there is no meaningful data yet.

MR. BERNERO: But the apparent result, if you use a risk code for a parametric analysis of the risk reduction effectiveness of things that work on late containment failure, you will get a false impression of value using that risk code. Now, that risk code CORRAL is basically only gravity settling of aerosols. You include other mechanisms for aerosol settling, and presumably you will come down closer to apparent truth.

MR. MARK: How long does it take to run the risk code versus the best estimate?

MR. BERNERO: I don't know.

MR. ROSS: Bob, do you know how? The CONTAIN code has only recently become operational.

MR. CURTIS: These codes are all aerosol codes. And as such, are all relatively fast-running. It is because they treat a very limited number of phenomena, and they use methods which just don't take that long.

MR. ROSS: In the thermodynamic equations, the bulk transport compartment.

MR. CURTIS: These, with the exception of the NAUA code, for example, I don't believe any of those give you any substantial treatment of the steam and atmosphere, for example, which is just that much more calculations that are needed.

MR. ROSS: But since CONTAIN does have the thermodynamics, the run time might turn out longer.

MR. CURTIS: The model that is in CONTAIN is essentially the same in its physics as one or more of these codes. It was preprogrammed a little bit to be more compatible with the driving system, but the underlying physics all comes out of experimental programs and special purpose models that were developed along with those experiment programs.

MR. BENDER: If you develop the CONTAIN code, could you throw the rest of them away?

MR. CURTIS: It says that if you find it convenient to use your own algebra on the computer for a specific problem, you will probably continue to do it that way, but that if you want to do an integrated problem of the total containment response, you will use an integrated code to drive through the full range of various phenomena.

MR. MARK: Do any of these contain the results of the effort that went into the HARM codes? Because that was a very serious effort to discuss aerosol.

MR. CURTIS: The HARM code was one of those you saw on the mosquito chart, and the methods of the HARM code are in fact, if you will suppress a couple of features in CONTAIN with respect to aerosols, you effectively have the HARM code. We have -- the HARM code has an assumption that the aerosol distribution begins and remains an analytic log normal function. We have an option of using a discreet distribution of the aerosol size in CONTAIN, in addition to the log normal assumptions.

MR. ROSS: If you would change over to Slide 23, we talk a little bit about Element 12, which in turn was discussed at length at a three-day meeting in Albuquerque between the NRC and the IDCOR group last month.

(Slide.)

MR. ROSS: The general purpose is the cost benefit studies, where you look at the reduction of the core melt probability and or risk by add-on features that would either prevent or mitigate, either prevent core melt or mitigate the consequences. And on a risk benefit basis, try to come up with some measures as to whether this modification or combination of modifications was worthwhile.

(Slide.)

MR. ROSS: On the next slide, you see some candidate improvements. This is not an exhaustive list. This is just some that have been flagged. You can see that the list includes what you could call prevention of core melts, and some would call mitigation.

(Slide.)

MR. ROSS: These comments or add-on features would be in the analysis at Sandia, would be tried singly and in combinations, and the risk reduction and core melt reduction calculated for the combination. These are some extremely preliminary results. This is for a BWR Mark III, and it is showing, for example, the changing in frequency and the consequences, and the population dose. These are just the features one at a time. The study, when complete, will exhaust the list by two or three at a time, presumably on out to all at a time if it makes sense, and also work at the cost of the feature or features.

If you display cost versus risk reduction or core melt frequency reduction you might come to some conclusions as to whether one or more features happen to be worthwhile.

MR. BENDER: Is this a PRA analysis, then?

MR. ROSS: I would call it a differential PRA analysis because the absolute value is not so important as the change, but it is definitely taking a PRA analysis and assuming that the plant is slightly difference, as the candidates indicate, and then doing a differential assessment of either the consequences or the core melt frequency.

MR. LEWIS: Why does a high volume unfiltered containment vent reduce the core melt frequency by a factor of 13?

MR. ROSS: Say that again.

MR. LEWIS: Why does a high volume unfiltered containment vent reduce the core melt frequency by a factor of 13?

MR. CUNNINGHAM: On the boilers in general and in the case of the PRA that was used here, the Grand Gulf RSSMAP, one of the important sequences was a long-term loss of containment heat removal. In that case, what happened was the containment heated up to the point of failure, the containment failure. Then all of this time the ECCS was working. At the time of the containment failure, the ECCS was assumed to fail with some probability such that the containment failure led to the core melt.

The high volume vent prevented the gross overpressure failure such that you would not get the ECCS failure.

MR. LEWIS: Okay. I understand the sequence, but you are telling me by a factor of 13 that was more important than the total of all other sequences?

MR. CUNNINGHAM: I am sorry. I am not quite sure what you just said.

MR. ROSS: The table is incomplete. Maybe that is what is confusing.

MR. LEWIS: You are telling me that by a factor of more than ten that was a more important sequence in the analysis than the sum of all other sequences that would not be mitigated by the high volume core vent or containment vent?

MR. CUNNINGHAM: Or that combination, yes -- that those kind of sequences, that the overpressure leading to the containment to the core vent. I believe in this case I believe that is correct. Yes, in this particular RSSMAP that was a very important or dominant sequence by a factor of 13.

MR. WARD: And I think that is consistent with what Levy was showing us here a good many months ago.

MR. OKRENT: It was based upon a similar, if not the identical, study, I think.

MR. WARD: A good thing.

(Laughter.)

MR. BENDER: There is no subjectivity in this analysis at all.

MR. OKRENT: Limerick in its PRA also concludes that. I do not know that it is a high volume. I do not know what the term "high" means, but in any event the containment vent ability and the ability to continue to get water into the vessel leads to significant reduction in core melt frequency.

MR. LEWIS: But the ability to get water into the vessel --

MR. OKRENT: Is presumably more reliable than the ability to take it out of the containment.

MR. LEWIS: I am surprised.

MR. OKRENT: This is not what you would call a full-scope PRA. It is a limited PRA.

(Slide.)

MR. ROSS: Let's look at the next element that is not in the risk family -- slide 26. I will just summarize briefly on one side.

The SASA program, which is a few million dollars per year pure analysis program, on multiple failure events.

MR. KERR: Mr. Ross, I feel some obligation to remind you and the Committee of yours and my discussion that what the Committee was likely to be interested in -- and I did emphasize that the Committee had already made several comments that they would like to see what questions in the regulatory process were being answered, and I expect you are going to do that later on and I just wanted to remind you before we got too far along into the process that I did mention that as something that I thought the Committee would like to look at.

MR. ROSS: What we did, Professor Kerr, was specifically that if you peek ahead a little bit to 29 you will see that for the ingredient for which there appeared to be the greatest controversy we have a specific listing of those questions.

I do not have a specific listing, for example, for SASA. The questions it is supposed to have in the writeup in Chapter 5, we list the issues as they progress. I just do not have the slide, except for element 5.4, which is 40 percent of the money and 100 percent of the controversy, which is one reason why I am going to be briefer on everything but element 5.4.

MR. OKRENT: I can read you what it says in the long-range research plan that we were handed out. It says SASA, the element, the issues being resolved by this element are: one, severe accident policy; two, licensing and safety concerns generated by abnormal transient operator guidelines.

MR. ROSS: That is bullet two on the slide.

MR. OKRENT: Five is NRC unresolved safety issues. It is a very ambitious set of things that are being resolved by SASA according to the long-range research plan.

MR. ROSS: Well, all of these have not been resolved, of course. It addresses some of them. For example, blackout and A-30 and A-44 and, to some extent, if A-47 involves multiple failures it will eventually address that -- such as total loss of feedwater.

As I said, it is pure analysis. There are four laboratories working -- Sandia, Los Alamos, Oak Ridge and Idaho. Final reports have emerged. Mr. Curtis is holding up one and I feel confident the Committee has them.

Other than that brief description of the SASA program, I am not going to go into any more detail unless there are some questions. Slides 27 and 28 did have supplemental information. However, the best evidence is to actually get the report, which, as I say, do exist.

(Slide.)

MR. ROSS: Let us look at slide 29 now. Pardon me -- slide 30.

I think we can characterize the main questions of the behavior of damaged fuel in three parts -- the fission product source term, the hydrogen source term, and the fuel behavior induced loads. Before the interaction of the molten core retention devices, if there is water on the containment floor, the reaction of the X vessel debris with the water.

But those are the main three questions. How accurate is the source term? How good is the hydrogen source term and the timing of the relief and what is the magnitude and timing of the fuel behavior induced loads on the containment?

A supplementary question which you could really relate to both two and three is under what conditions is damaged core to be coolable. I would characterize that as of lesser importance than the first three.

Now in order to answer these questions from the experimental side, there are three facilities being used, as shown on this and in fact we are also considering other facilities for fission product source term. But as far as the behavior of damaged fuel is concerned, the PBF, the ACRR and the NRU in Canada are the three facilities that will generate information under the subelement.

(Slide.)

MR. ROSS: Some pictorial views which will not show up well in your handouts but which we can provide better hard copies. The severe fuel damage test train. This describes a capsule which is in the process this month of being inserted into PBF and is going to be run for the first time next month.

It has 32 fuel rods which inside a mini-ball tube will be subjected to nuclear heating in a boil-dry atmosphere and eventually in a series of five tests they will get up to temperatures on the order of 3500 or 3600 degrees Fahrenheit. There will be five experiments in this PBF Phase I from September '82 being the first one to mid or early fiscal '84 for the last one.

(Slide.)

MR. ROSS: A cross-section -- really two cross-sections of that capsule. The left side is the cross-section of what is called current design. That really should read "Phase I" with the fuel pins and the insulator. If Phase II is run, it would involve temperatures up to the oxide-melting 5,000 Fahrenheit and it would look like the right of the center line. You will see it will be, again, a 32-rod bundle with a tad more insulation.

MR. MARK: I was not completely clear, perhaps because I cannot read the slide here nor can I read it there, just what is to be measured and determined when they bring this up to 3500 degrees Fahrenheit.

MR. ROSS: You say for the 3500?

MR. MARK: Well, any one of them -- hydrogen evolution or fission product release or what?

MR. ROSS: I will point out that for a long period of time this test was focusing on the behavior of the fuel and was not going, although everything that comes out is trapped, condensed, counted and so on to get a total recovery, it was originally not going to develop fission product transport and plate-out mechanisms in a manner analogous to what might happen in reactor coolant system piping. That was not going to be done.

We are currently looking at a proposal to modify the exit piping to in fact produce additional information on that. But, George, why don't you list specifically what, for example, would be in Phase I.

MR. MARINO: The hydrogen release rate as a function of time up to the melting temperature of the cladding. We certainly had that pinpointed. We will be measuring fission products in all of the tests.

In the first three tests we will be doing it more just of the conductors and the post-test evaluation that contain fission products, and in the later test we will use irradiated fuel and we will get a lot more sophisticated, but we are still working that out now on the fission product detection and mass balance of all of the fission products.

We will also do PIE analysis of debris formed and use that debris to characterize any of the feasibility studies so that they will have the right sizes to use for the core feasibility.

MR. MARK: And you will have things like the amount of water passing down as the heat goes up?

MR. MARINO: Oh, yes. The steaming rate, the hydrogen-steam ratios, yes.

MR. MARK: This will not cover such things as what will happen as it goes into steam or water?

MR. MARINO: The environment will be steam.

MR. MARK: In the collector?

MR. MARINO: In the collector. It will go that way and we will be able to detect the hydrogen.

MR. MARK: And you will be able to detect the fact that it has dissolved?

MR. MARINO: Yes.

MR. ROSS: There are a number of sample locations with remote observations that in given times -- that is, before and after the condenser too, isn't it -- the sample bombs?

MR. MARINO: That is the first. The later test, the sample bombs, are the very first thing, you see.

MR. ROSS: And there are some later on downstream, aren't there?

MR. BENDER: Is the intent to find out what comes out when it comes out or how it comes out?

MR. ROSS: Exactly.

MR. BENDER: All three things?

MR. ROSS: As I said, if you are successful in getting the exit coolant piping, you might also get what plates out between the reactor in the first place, where the exit mixture is condensed.

MR. OKRENT: To my knowledge, we do not understand how the iodine will get out of the fuel prior to melting, just in a fundamental way.

MR. MARINO: That is a good point, but most of the experts today believe that the fundamental way it does get out is through the paths generated by the fission gases prior to the release of the iodine and it goes out through the edges.

MR. OKRENT: But do the experts know what the paths are that are developed by the fission gases if we accept that as one of the possible mechanisms?

MR. SHEWMON: In the usual vein of seeing, that only seeing is believing, that metallurgists tend to work on, there is a fair amount of stuff that shows that channels are developed along these grained edges and that is a path for the gas to get out faster than any other mechanism.

MR. OKRENT: That is not the question. It is the detailed question of how these things along individual grains link up and at what rate, and does the iodine move along these paths. One agrees that you see tunnels on grained boundaries.

MR. SHEWMON: There are cracks, and if these get the gas out, they let the iodine out. What is the question?

MR. MARINO: Exactly. That is what they assume at Argonne and I do not think anybody questions that.

MR. ROSS: We can get some pictures if it is necessary. The separate effects test at Oak Ridge that are also heating fuel up to as high as 5,000 Fahrenheit on irradiated fuel, which should feed into a consistent path into the model development.

MR. KERR: What specific question about severe accident regulation does this answer, or is it just to increase our general knowledge?

MR. ROSS: There is a considerable feeling that the Agency overestimates the fission product source term during severe accidents and, if so, perhaps we are overregulating. Let's find out.

MR. OKRENT: But which severe accidents -- those that do not core melt?

MR. ROSS: I am sorry -- what?

MR. OKRENT: Which severe accidents -- those that do not core melt?

MR. SHEWMON: Those that overheat the fuel and releases iodine is what we are talking about.

MR. ROSS: The test program, if we are successful, will cover all core melts up to 5,000 degrees Fahrenheit. Now it will cover various scenarios. If you mean one where the fission products are released at low pressure versus one where they are released before the primary system fails and the current system is 1,000 pounds, I do not know if it makes any difference, but let me inquire.

George, are any of the PBF sequences fuel failed at 1,000 pounds pressure?

MR. MARINO: All of them. The pressure beyond 1,000 psi.

MR. ROSS: Will any of them be as low as 100 pounds?

MR. MARINO: No, but the ACRR test will be at lower pressures.

MR. ROSS: If your question is is the release sequence dependent -- is that the question?

MR. OKRENT: No. Where does the risk arise? The risk does not arise from sequences where the fuel stays solid and stays in the vessel.

MR. ROSS: Agreed.

MR. OKRENT: Okay. That's all.

MR. ROSS: We will stipulate to that.

MR. OKRENT: Your experiments are in the area where risk does not arise.

MR. ROSS: But Phase II does not have solid fuel. It goes to 5,000 Fahrenheit. The biggest problem now it to find the funding and support to run Phase II.

(Slide.)

MR. ROSS: This is a picture of the test train that again you will not get from your slides. It was installed about two weeks ago in a transfer task and is probably at the PBF by now, if it is not already inserted into the reactor.

Physically, George, where it is? Do you know?

MR. MARINO: It is in the reactor now.

MR. ROSS: Go over to slide 38.

(Slide.)

If you recall, one of the questions had to do with the nature and timing of hydrogen. Let us talk about the hydrogen generation and control subelement, Element 5 or 5.5.

Part of the research program concerns both experiments and analyses for hydrogen generation and control. The problems listed for some accidents and at least one real one, you can release hydrogen and it can burn, possibly detonate, resulting in containment failure or damage to equipment.

The solution is to try to define the maps where these bad things can or cannot happen. You do it with certain analysis methods and certain experiments which would validate these models. Most of this work is being done at Sandia, both the analysis and the experiments. However, we are on the verge of concluding a contractual agreement with Nevada Operations Office of EPRI to run and also a number of foreign countries are involved to run some hydrogen experiments in a 52-foot diameter sphere located at Test Cell C.

MR. KERR: Let me see if I understand. Suppose one takes 50, 34 or 36 -- I forget what the number is -- which requires that brand new plants are designed for 100 percent reaction. Now is this research designed to tell you whether you need 100 percent -- not 100 percent but 80 percent or 120 percent. Or is this research that we do not know yet really how to design?

For 100 percent we have to do this research in order to know how to design for 100 percent, in which category is it, or is it none of the above?

MR. ROSS: I think none of the above is the right answer, Professor Kerr. I suffered through the McGuire hydrogen hearing at the Appeal Board and a lot of the calculations that came up and you could assume severe core reaction -- metal-water reaction -- and in terms of pounds per second a lot of hydrogen produced, and you could actually have a mixture coming out of the assumed break -- and this is in an ice condenser. And depending upon the steam, the mixture might be steam-rich and it would not burn. In theory it could even be hydrogen-rich and would not burn.

And I think the object here is not to prejudge what is needed but to find out what is right.

MR. KERR: Let me restate my question.

MR. ROSS: This is to give the most likely sequence. Now what needs to be done in terms of regulation in order to be conservative, I assert you cannot tell. 100 percent may not be conservative.

MR. KERR: Let me restate my question because I did not do it very well.

Apparently, if I understand the rule correctly, we now have a regulation that says a licensee coming in for a CP today has got to design for 100 percent metal-water reaction.

MR. ROSS: For CPs.

MR. KERR: Yes. And he has got to have that capability. Now if indeed such capability exists, why do I need to know how much hydrogen comes off and what the time sequence is? What is it? Does this research give me some information which permits me to make a design under the assumption that today in spite of the regulation people do not know how to do it?

Or is this regulation -- I mean, does this research -- is it some that will tell me that maybe 100 percent is too much and it really ought to be 80 or 60?

MR. ROSS: If the net result is the latter -- that is, if we find out that it makes total nonsense to reuqire 100 percent and it should be 31 percent -- that is unacceptable answer and it might even be a good enough justification for the research.

MR. KERR: I am trying to find out what your judgment is. Somebody planned this program in the context of the requirements that one designs for 100 percent metal-water reaction, not for all plants, and so maybe it is for those plants that have to design for only 75 percent.

What I am trying to get at is what is it about existing regulations and new regulations that said we need this information?

MR. ROSS: In the first place, I am not trying to bypass the issue. This element 5.5 was not intended to explore what the fraction of the core was that would reach -- that would have a metal-water reaction for a given sequence that would be more than the preceding element.

This one is going to take variable amounts of hydrogen and permit it to burn. And if it will detonate, we will let them do it.

MR. KERR: But if you can already design plants to handle the problem, presumably you can or you would not have a regulation for operating plants that says you have got to handle 75. I mean, you must have some confidence that people can do that. Otherwise, you would not make them do it.

Why do you need research to tell you how to do it?

MR. ROSS: I come back to the hearing. The regulatory staff had no computer code and still has no computer code to audit the calculation of hydrogen being transported around the containment. It had none.

Now if the Agency is going to be an auditing agency, in audit calculations it follows that it has to have analytical techniques to calculate it.

MR. KERR: But it does not have to do research to collect data. Somebody has to collect the data.

MR. ROSS: I just have to disagree with you. If you want a computer code to analyze hydrogen transport distribution and combustion in the containment, you are going to have to do the research.

MR. KERR: I did not say you did not have to do research. I thought you were talking about experimental research which had to do with hydrogen generation. You are not. You are talking about analytical work.

MR. ROSS: This is combustion and burning work, but there is in fact work in the previous element -- intended experimental and analytical work -- intended to quantify the most likely hydrogen production rate for a given scenario. That is true that is an element of the plan. It is just not this one and it can and will affect the predicted outcome of the sequence.

Perhaps I should not have passed up the sequence.

MR. KERR: I do not have any reservations about the research. I am trying to find out what information now exists which permits people to design the plant to handle 100 percent metal-water reaction.

MR. ROSS: Let me see if I can explain it this way.

(Slide.)

This is 37. If you were going to use today's technology you would be predicting the green line that says for a given sequence fission gases -- in your handout there are similar charts for hydrogen produced and peak clad temperatures. The green line would show that if the zircaloy did not move and migrate downward but stayed where it was in the reactor and there was no diffusion boundary between the cladding and the vapor channel, so you just kept on reacting, whereas it may be that the yellow line or even the purple line is the truth.

Now you cannot tell in advance, it is my assertion, which is conservative. You can make an assumption and get the green line. The cladding stays there and just reacts.

Now there are experiments -- in fact, I have got some pictures -- to show that is not true. That is not what happens to the cladding in an adverse environment.

(Slide.)

MR. ROSS: Now a similar curve -- and perhaps I should have started with this -- is with the temperature. You can calculate the cladding using today's techniques -- that is, 100 percent reactor. The zirc just says there until it is all reacted for a given transient or this is temperature and Kelvin and time and seconds. You come up to right here.

This portion of the curve is where the cladding is being heated by the decay heat. At this point, the contribution of the metal-water reaction is stronger and you start heating up on a different line.

Now here you can make an assumption. One is that the cladding stays there. That is what the present techniques are. And you go on up into this region here. You can do what experiments say or what the computer code SCDAP would say and let the clad migrate downward and you get the yellow line.

Now given this, the hydrogen production rate would be different for the green and the yellow. The fission product source term is going to be different and if you are going to do a consistent analysis you are going to have to do it all the way through where the depletion of the burning of the hydrogen and the depletion of the source term by settling of the plate-out or whatever is all done with consistency.

Research is needed to do this kind of stuff.

MR. OKRENT: Denny, can I give you an example of what in my opinion, if it were to be answered by an experimental program, would provide a response to Dr. Kerr's question that at least to me is more meaningful?

I earlier in this discussion mentioned that there is enthusiasm for PWRs toward keeping flooded cavities and I said, however, I had not seen the thing thoroughly examined to see that there was not a negative aspect to this that led to an early release that had a sufficiently high probability that it was a concern.

Now if one analyzed this and asked himself what is it that could lead to the negative aspect and he concluded that it was vital to know not only the mode in which the vessel failed but the hydrogen that was in the vessel at the time this occurred, and so forth and so on, and that in order to do this you had to go back and look in detail at the progression and so forth, at least I could understand that it is what I would call a risk issue.

My problem is I do not see where the risk arises if you do not get the fuel melting and I am with Dr. Kerr. I do not see where that part of it relates to this question of designing for 100 percent hydrogen. What I read from your own consultants is for some scenarios maybe 100 percent hydrogen is not enough if you go through core melt.

If you have an arrested accident, it is strictly a matter of judgment. You choose a number.

MR. BERNERO: Let me interject here. For years we regulated with respect to hydrogen on the basis of the generation of a very small amount in an essentially large air atmosphere, two percent, half a percent, one percent, or four percent, or something like that, where the phenomenology of hydrogen and its combustion could be understood with relative ease.

At Three Mile Island the accident was arrested before full-scale core melt, by all evidence, and a lot of hydrogen was generated. The Commission, having been told repeatedly that hydrogen is there in large quantity even though you haven't melted the core, chose to regulate hydrogen at the level of 50, 75 or 100 percent metal-water reaction, even though it is questionable whether it is physically possible to have 100 percent metal-water reaction without already having melted the core.

The Commission regulations, which now as then spoke of the fraction or extent of the metal-water reaction -- 75 percent, 50 percent, or 100 percent -- those regulations don't speak to the crucial problem of the hydrogen-steam source term, how did it come out and how did the water come out with it.

In the rulemaking process itself, we have published a NUREG document which offers one of the alternatives for choosing the hydrogen-steam source term. This research is necessary so that the regulatory process is confirmed as adequate and the mechanisms by which licensees analyze and demonstrate they meet the regulation is that those mechanisms, those designs, are indeed adequate.

Right now you can go into different accident sequences and get sufficiently different hydrogen-steam source terms to get almost any containment pressure you want and any result you want, and to the same extent 100 percent or 75 percent metal-water reaction is manageable by igniters in one accident sequence and not in another.

MR. KERR: If I understand what you're saying, what I think you're saying is that at the present time you are not confident that anybody knows how to design for a 100 percent metal-water reaction, and that this research is necessary in order to be able to do it. And you are the first person who has told me that. If that is the case, that to me is significant. I have not heard anybody else say that.

MR. OKRENT: I don't think he said that, but maybe I'm wrong.

MR. KERR: I'm not sure he said it. That's why I tried to repeat it.

MR. BERNERO: That when the ECCS regulations first went into effect there was sufficient confidence to license plants, but not sufficient confidence to suspend research in ECCS phenomenon. It was, perish the word, confirmatory research, and to a very great extent the hydrogen research is that. There is a very limited data base for the designs that are being accepted in the licensing process.

MR. KERR: There is a question. I don't care how limited the data base is, but there are enough data so that one can design a system to take care of it after all, I assume.

MR. BASSETT: Professor Kerr, I would like to address that. We don't have sufficient confidence that the igniters as placed will do the job, because we don't have enough knowledge of how the hydrogen propagates around the containment.

MR. KERR: I asked Mr. Ross if the situation was such that you didn't -- weren't confident that anybody knew how to design for 100 percent metal-water reaction.

MR. BASSETT: And I think Dr. Ross responded that we had to have confirmation of those designs and we needed data to get it.

MR. KERR: Wait a minute. You have to have independent capability, the implication being that there exists a car-bility in the industry, but you need independent capability.

I think Mr. Bernero telling me that he isn't confident that anybody knows how to design for 100 percent metal-water reaction -- did I misunderstand you?

MR. BERNERO: Not quite.

MR. ROSS: Since I was the licensing witness at Maguire, let me respond as to what licensing did at Maguire at the time. Once scenario, one core melt scenario called S2D -- I guess extra-small break is S2, and failure to recirculate -- that produced a hydrogen source term. There was a calculation of the thermal hydraulics consequences, including hydrogen, with MARCH.

MARCH was relied on to calculate for this one sequence how much hydrogen came out per unit time, how much steam came out per unit time, into the lower compartment of an ice condenser. It was said then that that was good enough as an interim basis, but that more calculation should be done and the NRC should be able to calculate it independently and more sequences should be investigated and the deficiencies in MARCH ought to be fixed, because there were many deficiencies of MARCH, which were being used for the purpose for which it was not intended.

Now, all of those licensing deficiencies will be remedied when the hydrogen finally comes to a conclusion, I assert. And the Commission, and I suspect the Committee, knew this full well when they did what they did on Sequoyah, because the same position was taken on those two plants. It was called interim.

MR. MARK: If you could design for 100 or 150 percent hydrogen, if it was coming at the rate of one or two percent a minute, and not merely generated at that rate but released into the atmosphere -- and that is what the Grand Gulf people prepared to do, at 163 pounds per minute.

But of course, the hydrogen doesn't necessarily come just that way. It came at a much slower rate in TMI if it was coming across the three or four hours that the hydrogen was being formed. But nobody is too confident that it appeared in the containment at the same rate that it was formed in the core.

If you create this and then let it come out in a very short time, then the igniters won't do it, and that's one option. You have a real drop of all of the zirconium into the water in a pool under conditions where it can oxidize rapidly; you can form the 100 percent in a short time.

So I think there is lots of room for study here. Whether this is getting at the essential points fully persuasively, I am not clear on.

MR. ROSS: Well, the program is supposed to, the nature and timing of hydrogen release, as well as fission products.

We have only 45 minutes and we have several other important elements, so I would like to go on if at all possible.

Element 6 is fuel structure interaction. This is the ex-vessel work, mostly at Sandia.

(Slide.)

For large UO-2 melts that would be poured on concrete. The efficacy of various core retention devices to be explored.

(Slide.)

A very crude explanation of some of the work, to be showing what happens when some of the molten material would -- you would have an alumina bed that was dry and you add some water, and this would develop some penetration for 15 inches of the aluminum gravel. It looked like it penetrated about the same, but when you put a fine bed underneath the core bed it tended to stop the suppression and the melt spread out a bit.

This is just illustrative of some of the tests going on at Sandia.

(Slide.)

(Slide.)

I mentioned briefly the computer code CONTAIN, which is an integrated code for both thermal hydraulic phenomena and fission product phenomena. There is a related code called CORCON, the core-concrete interaction. Both of these codes are being developed at Sandia.

(Slide.)

If you move over to Slide 49, containment integrity. This experimental work is being done at Sandia, and it involves a better understanding of when and how the containment fails in the ultimate sense, that is going beyond the so-called design basis until the containment begins to fail.

Now, we have had in the licensing arena some capacity estimate, and I think the Committee was considerably pained when it sort of sat in judgment on this capacity estimate in the Sequoyah, there being such a wide range, factors of four and five in the interior pressure.

So experiments will be done: static pressure, unsymmetric pressure, and perhaps, if you look forward for five years, to the seismic effects.

(Slide.)

Now, Mr. Costello is still here, if there are any details in the area of the isometric models. But point number two is when and how the penetrations fail and what constitutes failure, how much deformation. And for both steel and concrete vessels it is hoped we can get a better handle on this.

Now, there was a workshop in June which -- where was that, Jim, at the Marriott?

MR. COSTELLO: The Quality Inn in Crystal City.

MR. ROSS: That dealt with this, and presumably we will use the output of that workshop to help refine our research claims.

Unless there are any questions, I would like to then look at the last research element that we haven't covered, and it is second in terms of controversy and second in terms of expense.

(Slide.)

Fission product release and transport. Now, if you wanted to convert these one through five statements -- this is Slide 52 -- into a question, then these would be the questions that we would want to answer through research element 5.9, the various elements dealing with siting policy. This would be the best estimate source term.

MR. SHEWMON: If I wait until the next slide or someplace on there, would I get into questions of how much of the fission product condenses out fairly close to where it went through its last orifice, instead of is floating around in the atmosphere?

MR. ROSS: Yes and no. One of the engineers in Bernero's division has called what you are talking about "spontaneous rain." So far the world hasn't given us a name for what he's talking about. But let me try to illustrate on this figure.

In this model containment, if it fails here and if at the time of failure there is some pressure in the containment, then whatever goes out would tend to expand, cool and rain. Is that what you're talking about?

MR. SHEWMON: No. There is a word I can't think of now, but it means that you take very fine colloidal suspensions and when they collide they coagulate.

MR. BASSETT: Is that conglomeration?

MR. SHEWMON: That's a good word, but that's it. EPRI came back in six months or a year ago and said you people had grossly underestimated this because it goes as the third or fourth power of concentration, and thus these things all come through an orifice and a great deal of this will have condensed out someplace, and therefore your source term of assuming all of this is uniformly suspended and then goes out when it leaks, is a gross overestimation.

MR. CURTIS: This is precisely the problem for which the proposed experiments are under way, and that is to look at the attenuation factors within the actual primary vessel before you even get to containment.

MR. SHEWMON: Who is this?

MR. CURTIS: Marviken. In a vugraph a little bit later we'll show that.

MR. SHEWMON: That condenses it out long before you get to the rain?

MR. CURTIS: Yes.

MR. ROSS: But, given that it got out, there would be some additional attenuation.

(Slide.)

This is slide 54. There would be some additional attenuation and we are yet deciding whether we want to put that into our research program or not. And I think, Bob, it would be in CRAC if we had it at all.

MR. BERNERO: It could be. There is some work going on right now just to see how worthwhile it is to pursue it. We have a choice of where to put it.

MR. ROSS: But if you're looking at Slide 54 and you see the E_c, question mark, that means there is probably some fission product with retention there. And we're not sure how much and we're not sure whether we're going to study it or not.

MR. SHEWMON: What are you doing on aerosols and how they're generated and how they agglomerate and thus how much of it gets into the containment environment?

MR. ROSS: Let me explain using this slide. But the answer as far as the primary system is concerned, how much retention and plateout is there between the core and the exit to the containment -- we are developing a computer code called TRAP-MELT at Battelle-Columbus, and we intend to do some experiments to validate that code at either Marviken or PBF and some other places, or all of them put together.

MR. SHEWMON: When these guys beat up on you on source terms, it seems to that is more of a potential for order of magnitude change in source term than anything else I've heard of.

MR. ROSS: Let's look at some values, and let me make sure I've got -- yes, this is 54A on your handout. This is very speculative factors on what you can get from research, and in the primary system there is a factor of ten. This is this fission product attenuation.

There is additional speculation perhaps that the core could be a factor of ten. Obviously, this is species-dependent. This model right in here would be for those reactors having a suppression pool, the BWR's, and you would have some additional attenuation if you were above the cool pressurizer. If you were dry you wouldn't have it.

The Marviken experiment, which is Slide 55 on your handout, is intended to look at this and this, the non-radioactive simulant, either out on the piping or the bubbles through the liquids.

MR. BENDER: Denny, when you're doing those experiments in Marviken, what is the source of the nuclides?

MR. ROSS: Let me give two names which don't mean much: corium and fissium, two simulants. Now, there is some controversy over what constitutes an appropriate simulant. There was a week-long meeting in Marviken a month ago, or rather, in Stockholm, where we had eight or nine countries represented, and I'm not sure that got settled.

Who is going to discuss this? John, did you inherit this? Do you know what the simulant is?

MR. BASSETT: Yes. There is a discussion now of using a corium-type simulant, but to include a lot of the cadmium-silver control rod type material in the mix.

MR. ROSS: Isn't Oak Ridge helping considerably?

MR. BASSETT: Yes. There's a vugraph you're going to come to which shows John Parker's experiment, where they're heating up simulant bundles and getting an estimate of the amount of aerosol that is being generated. These are both with and without steel.

MR. SHEWMON: If you go away from silver-cadmium control rods, you will probably change it a large amount, because that generates a lot of aerosol. It bumps into things and condenses out. So even separate from what they are talking about is other kinds of questions.

MR. BENDER: There are so many darn many variables that I am not sure that we can ever determine whether we know that the experiment has relevance to the spectrum of events. Is that what is controversial? You said this was the second most controversial.

MR. ROSS: Yes, I think so, Mr. Bender. Last Friday we had some discussion which wound up incomplete as follows: To what extent, on Marviken or anything else, have the range of scenarios been studied such that sufficient fission product retention, we felt we had good across the board coverage.

And I mentioned that there were some detailed studies done in association -- in fact in preparation for the meeting on Marviken, there were four such reports. And when you read these as a whole I think they will convince you that the range of scenarios have been considered -- temperature, pressure and so on -- because the temperature of the exit piping is important. The pressure at the time of the release is important.

Sam, have you had a chance to get those reports down to the ACRS, or is that in work?

MR. BASSETT: That is in work. It should be done this week.

MR. BENDER: Does that mean it's being studied parametrically?

MR. ROSS: Yes. There are eight or nine different tests, which include a range of pressures or a range of scenarios, like a large-break LOCA and a TMLD' and so on.

MR. BENDER: Thank you.

MR. ROSS: Now, on the same chart, if you overlay the red you will see a family of analytical methods, and the title is "Codes Being Developed." And if you have a little -- if I can find one with a box around it, that means it's a non-NRC code. And this is 54-C in your handouts --

(Slide.)

-- where we have codes for the core. And notice we have TRAP-MELT for the primary system, and M is mechanistic and the P is probabilistic. And in the containment we have either CORRAL or MATADOR, depending upon the timing. The non-NRC code is NAUA, and we're working on the CONTAIN and various other things, the codes.

Let's see. We still have our unknown here. Oh, pardon me. The non-NRC EPRI/GE pool model, which I'm not familiar with. Okay, that is the different models related to the source term.

(Slide.)

And you can have some experiments, then, to help validate the models at different places. The same general legend applies. The box means it's a non-NRC experiment, but we expect to have access to it. And you see different experiments scattered around for the containment, for the primary system.

You will notice Marviken here. Marviken is in the very throes of development. There is no contract signed, there is no assurance it will be run. It is being developed and various countries are being asked by Sweden to contribute. We certainly intend to participate, but it will probably be through EPRI as the United States' main agent, and then we will work with EPRI.

(Slide.)

Let me put one more version up of this. There are the loadings typed. You say CONTAIN, for example. We have codes down here for pressure, temperature and mass. So you get the primary containment thermal hydraulic conditions, the hydrogen loadings code, the RALOC and HECTOR, and down here the COCMEL.

(Slide.)

I will put on one brief slide. Slide 55 is just a quick picture of what Marviken looks like, and this is a facility that has been in service about 50 miles south of Stockholm for a long time. I guess it was a reactor, but it was used a lot for containment loading tests, so it has got -- it is called a reactor, but it is really not a reactor any more; a pressure vessel, a pressurizer.

The idea will be to release at prototyrical temperatures and pressures, meaning the piping will be very hot, fission product simulants and track them to a mass balance around the system. If this experiment goes, it will start early next year and it will run through calendar '84-'85, at a total cost, the last I heard, of about $10 million.

Do you know anything different, John? Is that close enough?

We do have a status report on Marviken.

(Slide.)

The purpose is to develop things like TRAP-MELT, and the status is we're still designing the technical isotopes and we're awaiting agreements on the funding. We should know by 1 December whether the project is going to go or not.

I think I will stop at that point, Dr. Shewmon, and suggest the last bit of time would be better spent on the questions and answers.

MR. SHEWMON: I suspect we will find a few.

MR. OKRENT: The question of radioactive source from accidents is certainly one that warrants thought. I guess my own feeling at the moment is that it needs at this stage an order of magnitude kind of thinking, because if you become satisfied that for certain scenarios there is an order or two orders of magnitude reduction, we may not care whether it's one or two, for example, compared to others, and it may not be important at least at that stage to be very precise.

And also, it seems to me that if, for what are now called the dominant contributors, you find, at least as they are now characterized, you do get one or more orders of magnitude, they may not be dominant contributors any more.

Now, what I don't know about, whether it exists, whether it has been done, or so forth, is whether there is a systematic look to see in an order of magnitude way, first, which scenarios with our current, let's say, intuitive thinking are likely to lead to lower releases and which scenarios, were they to occur, would lead to larger ones.

And then, having identified the kinds of things that could lead to larger ones, what would be the causes of the scenarios and how much do we know about them? For example, certain internal missiles is one possibility, internal to the containment, if there is a concurrent accident and a considerable loss of containment integrity, maybe not for all scenarios but maybe for some of that class, for example.

In the absence of my having seen a study of this kind, I find it hard myself to judge how important it is to look in great detail at things. I can't tell, it might be that the Marviken thing is very important, or it may be that for the kinds of scenarios where it is relevant you already know quite a bit about the order of magnitude.

Do you understand the question?

MR. ROSS: Very well. But I will have to ask Bob to respond.

MR. BERNERO: I wonder if the Committee recalls when NUREG-0772 was published -- and I can't remember the date -- one of the Commissioners, Commissioner Ahearne, asked a question along these same lines. And there was a memorandum prepared by Staff that is a very important supplement to 0772 in which there is a display of the relative risk reduction that is envisioned by looking at individual accident sequences and their characteristics, and separating the ones where there is a more likely reduction of the order of magnitude of source term from those where that reduction is not so likely.

And this can shift the balance, as you indicated. I don't recall any specific results from that memorandum. It was circulated widely. I can get you copies of it if you would like.

MR. SHEWMON: The noble bases constitute a very small part of the source term, is that right?

MR. BERNERO: Of the risk, yes. Noble gas release is not -- does not constitute, even total noble gas release, any dominant offsite risk in our present model.

MR. SHEWMON: Well, my guess is that I will bet your memo supports, or should if it doesn't, is that a very large part of the source term would depend on the models or the results of exactly what is being done with aerosols here, because you guys assumed in your bounding way to take it all in gas and most of it is going to condense out someplace.

MR. BERNERO: But the problem is, in the WASH-1400 model which has carried over to this day the decontamination factor of the reactor coolant system is one. I mean, everything gets out. And the issue is what decontamination factor should be put in for each sequence on a large LOCA.

Intuitively, a big burst, very low pressure core melt, you might have everything come out at one extreme. On a station blackout, where you melt the core at 2500 pounds and everything is going out a tortuous path through a relief valve in the pressurizer, you might have a DF of 100. And that is the issue.

MR. CURTIS: Another scenario with respect to the containment atmosphere is that the time of containment failure, if you could hold the containment failure together for many or several hours, the decontamination through the various agglomeration and fallout mechanisms is rather substantial.

MR. SHEWMON: Yes, but also, any time you go through a vent, as he said, A, your double-ended pipe break is nothing but a bounding geometry. Don't ever take it too seriously, as you just did.

And the other one is that when you rupture your containment that too was going to be an orifice of some sort, which will condense it in. And so I would come back to my assertion that you probably are very substantially underestimating or over-estimating, whatever it is.

You're getting too large a source term from what you would get if you understood things, and I would personally encourage you to understand that part of it any way you can, and argue with the members of the Committee if that is what they need.

MR. BERNERO: That is exactly -- I think if you summarize the results of NUREG-0772, it says just that: We think we are overestimating the release from the system of what people usually call the source term.

We are overestimating it. We are not too sure how much and where. The overestimate is prevailing and that is what we're trying to work on.

MR. ROSS: I have two housekeeping matters I forgot. One is, Professor Okrent was mostly right, it wasn't the morning, it was the afternoon, and I did say that reactors in other countries had design features, let's say, more conservative than the U.S. reactors.

And the other thing is, why are we here today? And we are here, the Staff is here, because we have a tentative but fairly firm appointment with the Commission on September the 9th to discuss this subject. What we believe would come out of that meeting would be a Commission instruction to either continue executing the plan as described or to modify it consistent with the new Staff requirements memoranda.

Obviously, we want an ACRS comments of concurrence on the plan. That is why we are here. And the timing is important.

MR. OKRENT: If I can come back to the question, I remember the memo and it is related to the question I raised, but it didn't address it in sufficient depth or in a sufficiently broad way to, I think, give the kind of guidance that one might get for what are the most important things to look at analytically or experimentally with regard to behavior of the fission products.

And so at the moment, as I say, I still can't tell where there are important things to be ascertained by experiments and where you can already judge, if the things go this way you get a considerable factor. And I don't really care if it's a factor ten lower, as it were, when you are looking at the risk in some way and trying to judge its acceptability and so forth.

MR. ROSS: I'm kind of interested in the answer, also. It seems to me like one way to answer it would be to take, let's say just take a typical plant, let's call it Surrey, and rerun Surrey with what we think the answer will be two or three years from now, what we think TRAP-MELT will be and what we think CONTAIN would be or whatever, and look at the change in risk.

And then we say, gee, we found out a lot of things. Let's suppose that TRAP-MELT, we put in a factor of 100 and it didn't make much difference in the overall risk; that, if that were true, then we would say, why are we spending all that money on TRAP-MELT, if we did that then for a number of reactors and for all the important scenarios.

I think what you're asking for clearly can be done and probably should be done. I don't know how long it would take to do it and whether in fact we could do it on a timely basis, let's say if you wanted to write a report in a month.

All I can do is agree with you that it sounds interesting.

MR. OKRENT: You would envisage something similar, but not quite what I had in mind. But in any event, let me just leave it.

MR. KERR: Denny, in the implementation of the plan it is very strongly stated, I believe, that one does not yet have enough information to predict with any certain confidence the behavior of containments.

MR. ROSS: Yes, that is true.

MR. KERR: If one went through 0900, could one identify the research, if indeed it is designed to do this, that would at the end of the program permit one to describe the behavior of the containment with confidence, or is that part of 0900?

MR. ROSS: It is part of 0900.

Let me get Jim Costello one minute on this subject, since he's been waiting patiently all afternoon.

MR. COSTELLO: James Costello from the NRC Staff.

I think the answer to the question is, yes, sir, Section 5.8.

MR. ROSS: Say some more. You've got 60 seconds.

MR. COSTELLO: Okay. I think it is a fair statement that today no one can predict with confidence the location and actual failure pressure of the containments. Past work, starting from Sequoyah and carrying on through the IDCOR efforts, is aimed at trying to get a handle on what loads above design pressure can be handled with some degree of confidence.

Does that answer the question?

MR. KERR: Well, I don't know what the question is, because what I find in the implementation plan is that we don't know enough about containments to describe their performance. Now, containment performance can mean the way they fail with overpressure, it could mean a variety of other things, yes.

I mean, I don't know quite what the language means in the implementation plan. What I read is, we don't know enough to describe them well enough to predict risk.

MR. COSTELLO: Those are not my words, but I interpret them in the context to mean --

MR. KERR: They may not -- they are Commission policy, though.

MR. BERNERO: Could I interject here that if one has a safety goal, presumably that means people are calculating that part of the risk analysis to see whether or not the number comes out like the safety goal, above the safety goal, or below the safety goal. Those people who have given indices of containment performance give it usually as a conditional probability, such as, given that there is a large-scale fuel melt what is the probability of a large-scale release from containment, and of course defining "large" in some way.

MR. KERR: I feel like a little boy who really, when he asked where he came from, wanted to know whether he came from Boston or New York.

(Laughter.)

What I'm really trying to find out is whether this research program has been designed to answer what the position paper says is an unknown.

MR. BERNERO: I was there when the words were put in the position paper, and what it is really referring to is that if you sit down today to calculate containment performance such as I just defined it, conditional probability of large-scale release, you have to be inserting or using values for the generation of hydrogen, the whole spectrum of containment loadings -- hydrogen, steam, ignition, things like that, and the behavior of the fuel when it goes into the pool, how it cools, the coolable debris beds or not. It includes the fission product transport, all of the things we have just been discussing today. Virtually every one of those research elements comes in.

MR. KERR: Is your answer yes, no, or none of the above?

MR. ROSS: We came down here on Sequoyah two or three years ago and said the failure pressure may be 67 pounds or 82 pounds, and the ACRS had to say, well, it's at least 45 pounds, which one's right?

The research element 5.8 is intended to answer when the containment fails.

MR. KERR: I hate to be against information, but really, all I want to know is whether in your view this position that I find in 82-1A, which is that you don't know enough now to calculate containments --

MR. ROSS: You don't know how to accurately calculate when containment fails.

MR. KERR: That is not what 82-1A says. That may be what it means, but that's not what it says. It says that one does not know enough --

MR. ROSS: What page are you on?

MR. KERR: It is not 1A, I'm sorry. It's the implementation plan that says that.

MR. ROSS: I'm not involved in that.

MR. SHEWMON: You do work for the Commission, I hope?

MR. ROSS: Frequently, very hard, many hours a week.

MR. KERR: I have enough documents here. On page 11, the Commission's goal is not establishing a numerical guideline on the availability or performance of the containment structure, whether or not such an empirical guideline is eventually established. Here it talks about "performance of the containment structure." I don't know what that means.

MR. BERNERO: That is a safety goal, such as a conditional failure probability.

MR. KERR: And I just wondered if you could say, we don't care about that, so we are not going to build that into the research. What I'm asking is, is that built into 0900?

MR. BERNERO: Yes.

MR. KERR: To get your position?

MR. BERNERO: Yes.

MR. KERR: Now, the answer I got from Mr.

Costello was, I think, that it is in 5.8.

MR. BERNERO: That is one element of it. That is the physical fracture failure pressure mode of the containment.

MR. KERR: But upon the completion of the research program in 0900, you would have, at least at this point, the expectation or the hope that you would no longer be in this position, but you would indeed be able to describe the performance of containment?

MR. BERNERO: With sufficient reproduceibility or reliability to consider having a formal performance standard.

MR. ROSS: I guess I will have to speak up. I'm a little concerned where the dialogue is going, because the people who formulated the research program, in particular 5.8, had something in mind. The people who wrote what you just got through reading there and which ultimately could be Commission policy may have had something else in mind. And I don't want to leave you with the impression that these two views have been examined for completeness and are interchangeable. I don't know that to be true, and without looking I can't assert that it is so.

MR. KERR: Well, you see, when we say that we would like to be able to correlate research programs with questions that have been raised by the Commission, we don't really quite know which part of the Commission we're talking about. But I had in my assumption that proposed implementation plan, were it to become policy, would raise a question, or could, and the question is how do containments perform.

Now, I would think if that is important to severe accidents -- and I think it is -- that one at least, in setting up a research program, would give it some attention. One might decide that's too tough for this five years, we'll put that in the next five-year plan. I don't know.

That's the reason I was asking. Is it in this five-year plan?

MR. ROSS: We can accept the question, but in order to fully answer it I would have to find out who wrote that and make sure that whoever wrote that knew what the research plan was and said, yes, that is what I had in mind. And I can't do that standing here.

MR. BERNERO: I can.

MR. KERR: I would have thought that it might go in a slightly different direction. The people who were setting up the research plan would say, what are the questions that the regulatory people need to have answers to.

MR. ROSS: Well, you see, we did this. This was done with NRR. We didn't wake up one morning and say, put this in the containment research plan. We were thoroughly involved with the licensing office for a year on this. We are getting material they very much need in individual licensing cases.

But if Bob has specific knowledge of who wrote that, then he can answer it.

MR. BERNERO: That implementation plan was drafted by NRR, and I cooperated and assisted in that from the very outset. And the reason for that paragraph and the reason for that recommendation is focused on the fruitfulness of using a numerical guideline for containment performance today. That is the sole basis of it.

There is no hint there of denying the utility containment performance mitigation.

MR. KERR: Now, at some point if this correlation does exist somebody must have in mind how one is going to describe containment performance.

MR. BERNERO: Yes.

MR. KERR: Where could we get that?

MR. BERNERO: Well, in NUREG-0739, the so-called ACRS safety goal, define three hazard states, and the third hazard state was containment performance accurately, succinctly designed. But it ignored dominant accident sequences like the ones we discussed earlier.

MR. KERR: I don't think the Staff takes the ACRS comments verbatim, any more than we take yours. What I'm trying to find out is whether there is a document somewhere that says, here is what we want to be able to describe about containment performance. I would find it educational and it would be nice, if it exists, to see it.

MR. SHEWMON: Professor Kerr, this is all very interesting. We are coming to the end of the allotted time, and thus I would hope the period.

MR. KERR: I have been watching the clock very carefully. I had assumed that you were depending upon me to fill the vacuum.

(Laughter.)

MR. KERR: If that is no longer the case, I will stop.

(Laughter.)

MR. SHEWMON: You have been ably assisted, but while you are filling this vacuum I would be interested in having you also comment on whether you are likely to draft a letter or have drafted a letter.

MR. KERR: I have drafted a letter.

MR. SHEWMON: Okay, good. Then you go fill it any way you want to.

(Laughter.)

MR. SHEWMON: Chet?

MR. SIESS: Can I try to answer your question?

MR. KERR: If you want to become part of the Staff and speak for them, fine.

MR. SIESS: No, but I've been following the containment program. It started out pretty much to be a containment integrity program. Performance was ended when the containment burst due to pressure.

I think now it is pretty much redirected toward answering the question of when, where, how, and how much the containment will leak. It involves more than the structural integrity. There is work going on or planned on penetrations and isolation valves, et cetera. And I think the questions have been asked properly, but I don't think they're going to have answers in five years.

MR. OKRENT: Can I comment on this? One of the questions that Mike sent in to the Subcommittee meeting, which we discussed indirectly, was, if there were to be a containment performance design objective, how would we get from where we are now to there. I think that is another way of restating what Professor Kerr has been asking, and I must confess I have not figured out how the Staff expects to get from where we are to there.

There are things that relate to this question in the research program that I haven't seen laid out, and these are things we need to know in order to get from here to there. And this is how the research program answers these things.

Now, maybe I missed it. Tell me that page.

MR. COSTELLO: Let me try to respond to the question along these lines: that the basic bit of information about a given containment is at what pressure will it begin to leak more than is acceptable, and that is more or less what we are after.

MR. OKRENT: That is clearly part of the question, but it is only a part. And having all of that information doesn't get you from here to there.

MR. BERNERO: I'm sure, Dr. Okrent, you recognize that in a full risk analysis that is done today one calculates the values that would fit in the equation for any index of containment performance that one might reasonably choose, becase one calculates the probability of full-scale core melt, you calculate the probability of large-scale release and the risks attendant to it.

No on denies that the risk is calculated today. We do it, the industry does it, whoever does the risk analysis does it. The question is, is it useful in the trial safety goal? Is it useful to put it in the quantitative guidelines and specifically go to calculate that ratio? And if we do, you will find yourself -- and this is why I think the Staff holds back from doing that.

If you look at the containment analysis, the containment event for the Zion risk analysis, we have discussed in a variety of our Subcommittee meetings here the questions, the uncertainties about that, where the mechanical failure pressure of the Zion containment, the coolability of the core melt debris when it lands on the floor, the likelihood of failure of the containment cooling -- so many factors, and the uncertainty or the debate if you are calculating that ratio varies over two or three orders of magnitude.

And to what purpose should we engage in that mathematical exercise? The Staff's choice of useful guidelines for quantitative safety goals, if you will, during the trial period is core melt and the ultimate public health risk, which is really the whole purpose of the thing.

MR. OKRENT: I'm not sure whether you've told me that you have no research program that could get you from here to there, meaning the information on which you could base the containment performance design objective, or there is one in there, or just what it was you were telling me.

MR. BERNERO: The research program is geared and structured to give us the information to make that a more reproduceible or reliable calculation. I think it is many elements of the research program that address all of those things: how do cores melt, what is the energy release, what is the fission product transport, what is the reliability of the cooling in the reactor building coolers, and how does the machine fail, how does the containment crack open and leak?

All of those things are being addresed in the research program, and at the end of four or five years I think we will be, in my view, in an excellent position to make a reasonable calculation of that index of performance. We even make the calculation today. I just don't think it is fruitful to go around comparing the Westinghouse-Pickard, Lowe & Garrick index with the NRC-calculated index, because they are all over the ballpark.

MR. OKRENT: Well, we are at the witching hour, so I will leave this.

MR. SHEWMON: Did you allow time for a break in your vacuum or not, Mike?

MR. BENDER: Yes.

MR. EBERSOLE: Could I ask Bob a question before we break?

If we had any evolution of this business and adopted what I call at least, and have for many years, a more rational pattern of containment design, which would say at some point, I am going to give up on the reality of my ability to estimate the pressure containment data, I will follow ancient and standard code practice and I will protect it by an automatic and reliable defense system which will discharge into a coarse middleman's filter, which will bubble through an ordinary pool right straight through to the atmosphere, would I then have a basic ability to work through what this thing might be able to do?

MR. BERNERO: Well, I'm not sure that that would speak -- you would have to define the index of containment performance for that. But what you descrive is what you might call a rupture disc filtered containment vent system.

MR. ROSS: That sounds like what the Barsebac facility in Sweden would have. Given that you wanted to do that, you would still need 5.8. You need to know where to set the valve.

MR. EBERSOLE: When one starts these things it means a lot where you start, if this were a good starting point.

MR. SHEWMON: Jesse, why don't you discuss this further with them while the rest of us are taking a break, and then we can come back in five minutes to the next topic.

(Recess.)

MR. SHEWMON: Take it away, Dade.

MR. MOELLER: The subject for discussion is control room habitability and I am pleased to provide some background information on this subject for you.

In the reviews that the ACRS did of Susquehanna and Fermi 2 nuclear power plants, several members noted wide differences in the operating capabilities of the HVAC systems for their control rooms, and indeed we asked one of the fellows to look into this and provide additional information for us.

We further noted in the reviews of LERs that there was a continuing increase in the number of failures of various components in control rooms of the operating plants and there were problems even among those under construction. We noted, in addition to these items, a paper by Murphy and Camp of the NRC Staff on control room habitability and they had raised a number of questions.

So, stimulated by these observations, the Subcommittee on Reactor Radiological Effects held a meeting on May 14, 1982 to learn more about this situation and to really familiarize ourselves with any problems that might exist in this area. And we found at least, to my way of thinking, that the problems are rather widespread and that there are many occurrences of failures in control room air cleaning and air ventilation systems that are not being reported in the LER system.

And, therefore, we concluded that perhaps this represented an area that would be of potential interest to the full Committee. Today we have lined up a group of four people to appear very briefly, each, before the full Committee for a briefing on this subject.

And these people consist, first of all, of Dr. Ronald Bellamy from the NRC Staff, now currently assigned to the TMI nuclear power station. Ron has many years of detailed and knowledgeable experience as well as work in the field of air cleaning, and he is going to be discussing with us control room air cleaning requirements and air filtration systems criteria for air filtration systems for control rooms.

Secondly, we will have Dr. Louis Kovach appearing before us, President of the Nuclear Consulting Services, again a person with years of experience in this field, including numerous reviews and evaluations of air cleaning and air ventilating systems of control rooms in operating plants. I thought you would interested to have him share some of his observations with us.

Lastly, we have two additional speakers who will present information. One is Leo Klaes from the Tennessee Valley Authority. He will discuss that group's experience in control room habitability design as seen by one of our major nuclear power utilities.

And lastly we have William Miller with us from Sargeant and Lundy, who will discuss the architect-engineering approach to control room habitability design.

Jesse Ebersole was at our Subcommittee meeting. Jesse, did you have any comments?

MR. EBERSOLE: No. I might comment on a peripheral matter. We are rapidly moving into an era when just biological habitability has now gotten to be a tight thermal environmental control because we are using more and more susceptible apparatus that has a very narrow acceptance band of temperature performance. And I think Arkansas Nuclear 2 is an example of what not to do in this aspect and maybe we can touch on that later.

MR. MOELLER: Yes. I think that is an important aspect that we are thinking not only of people but of equipment that might be vulnerable to poor performance in terms of ventilation and so forth.

Well, then, we will move on and call on our first speaker, Ron Bellamy, whom I have already introduced and who will be talking on the design, testing and criteria for control rooms.

Each of you has a handout of Dr. Bellamy's comments.

MR. BELLAMY: Thank you. The brief presentation that I would make in the next ten minutes will be a short version of what I presented to the Subcommittee on May 14. That is why the date on the first page of the handout that you have says May 14. I thought it would be more confusing to change that than to leave it the way it is.

(Slide.)

The regulation the NRC Staff uses in its beginning review of control room habitability systems is GDC 19 of 10 CFR 50, Appendix A. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 Rem whole-body or its equivalent to any part of the body for the duration of the accident.

There was some discussion at the Subcommittee meeting about the lack of specificity in this general design criteria. There are also some fairly specific guidelines. The word "shall" is used and the control room shall be provided and it shall maintain a safe working condition for the operators specifically under accident conditions of cooling, including loss of coolant accidents.

(Slide.)

MR. BELLAMY: Specifically, to discuss the air filtration systems provided to protect the workers inside the control room, the regulatory guide has been issued by the NRC Staff. That has gone through two revisions. Revision 1 and 2 of that regulatory guide did come through the Advisory Committee on Reactor Safeguards prior to its issuance.

It has -- this regulatory guide has also received extensive public comments in both Revision 1 and 2 and the Staff will be initiating a Revision 3 to this regulatory guide come January of 1983.

Regulatory Guide 1.52 has a very cumbersome title. It is entitled "Design, Testing and Maintenance Criteria for Post-Accident Engineering and Safety Feature Atmosphere Cleanup System and Air Filtration and Absorption Units of Lightwater-Cooled Nuclear Power Plants."

The introduction to the regulatory guide, general design criteria 19 is stated as being applicable to the interpretation of the regulatory guide and the introduction clearly defines the control room air filtration as being an engineered safety feature system. Once it is considered as an ESF system, the bases for the regulations as implemented in Regulatory Guide 1.52 and the recommendation for how to design that filter system to protect the operator then follows.

There are a certain number of environmental factors that need to be considered during a design basis accident. Differential pressure, both through the filter system and from without, from outside the filter housing to inside the filter housing, the dose rate and the integrated radiation dose on the components of the filter system, the relative humidity of the air, the temprature -- both maximum and minimum temperatures -- for the incoming air.

There is guidance given on the system design criteria. The typical components that one would expect in an engineered safety feature atmosphere for the control room would include a demister to remove the particulate water droplets, a heater to reduce the relative humidity of the influent air to a level where the assumed methyl iodide species would be absorbed more efficiently on the carbon, a pre-filter to remove the bulk of the particulate material, a high efficiency particulate absolute filter, a carbon adsorber to remove the radioiodine species, a second redundant bank of HEPA filters, a fan to move the air, and a housing to incorporate all of the components.

And I do have a sketch here.

MR. SHEWMON: On that temperature, is the maximum and minimum set for the comfort of humans or the efficiency of the operations of machines?

MR. BELLAMY: The specific temperature is for the efficiency of the machines.

MR. OKRENT: Could I interrupt for a minute to understand just a philosophic point in this question of the systems? What is the likelihood of an event that you think you need to, if it exceeds this likelihood, that somehow this control room air filtration system should deal with it -- an order of magnitude?

MR. BELLAMY: I am not sure I can follow the question. What is the likelihood of an event where you would need the control room habitability systems? It was required at Three Mile Island Unit 2.

MR. OKRENT: Well, for example, if you thought that chlorine were going to be released once in 100 years, you would install something to intercept it, to detect it and intercept it, right?

MR. BELLAMY: Yes, they are installed.

MR. OKRENT: Would it be one in 1,000 years? Would you design something to intercept that chlorine was coming in? One in 10,000 years? At what point do you say I do not have to worry about it in my air filtration system?

MR. MOELLER: At the moment, if chlorine is stored, you know, anywhere nearby and if basic meteorological calculations show that enough could escape or if it all escaped, if the concentration by the time it reached the intakes to the control room was sufficient to affect the people, they require detection units.

MR. OKRENT: Is that independent of the likelihood?

MR. MOELLER: Yes, it is independent of the likelihood.

MR. OKRENT: Well, somehow probabilistic considerations must enter into the decision somehow.

MR. SHEWMON: What he is saying is he cannot conceive of there not -- other people may conceive of that.

MR. MOELLER: I have not -- to my knowledge I have not seen probabilistic calculations used in the design of the control room habitability system. Is that right?

MR. BELLAMY: I believe so.

MR. MURPHY: I am Ken Murphy from the Division of Risk Assessment. The regulatory guides in terms of the toxic gas do use a 10^{-6} number inherently in them in terms of operating capacitation. We are considering using a number probably higher than that in the future work, on the order of 10^{-5} in the frequency of operator incapacitation for toxic gas releases.

MR. OKRENT: You are really planning to go up to 10^{-5}? Is that a best estimate or a conservative calculation or what would you call it?

MR. MURPHY: That is like a strawman number.

MR. OKRENT: I do not know what a strawman number is.

MR. MURPHY: Well, one that has not been approved.

MR. OKRENT: Again, sometimes the Staff has said well, if you get 10^{-6} but conservatively if you get 10^{-7} best estimate you do not have to consider something like airplane crashes, for example.

MR. MURPHY: Well, we are talking about operator incapacitation. We have no idea whether a transient is occurring in the plant, whether a plant is in imminent danger. The working number is what we are looking at.

MR. OKRENT: What is the basis?

MR. TROOD: Harry Trood. Standard Review Plan 2.2.3 does talk about the probabilistic acceptance criteria that you have just quoted and we have recently and particularly in NUREG-0737 referred to Standard Review Plan 2.2.3 and so the licensees are under that Standard Review Plan permitted, if you will, to show probabilistically that they do not have to provide protection if they are below the acceptance criteria.

MR. OKRENT: Which are?

MR. TROOD: 10^{-6}, I guess, conservatively, and 10^{-7} realistically.

MR. OKRENT: If those are the criteria that are operative today, do you exceed those or miss those if you assume you have a two-plant site and one of them has a serious accident with regard to the other plant? Has anybody analyzed this?

MR. TROOD: Yes. Basically in the Accident Evaluation Branch we feel with respect to radiation, which is specifically what you are addressing, that if we have detectors in the intakes that we have acceptably covered the cross plant problem.

MR. OKRENT: So in other words you think you will meet this even in that event? That is what you are telling us?

MR. TROOD: Yes.

MR. SHEWMON: Please let's let the speaker continue. I would like to not chew his fifteen minutes up all with our questions.

(Slide.)

MR. BELLAMY: This is a schematic of a typical air cleaning system with moisture separators, the particulate droplets, a heater, a bank of HEPAs, carbon absorber and more HEPAs. It is straightforward.

(Slide.)

Now for the design of these systems to protect the operators from the radioiodine, particularly after the release in an accident, you start with the assumption that you need the filter systems. There is no assessment done to any 10 to the minus anything. This is do it or die. We need the filter systems. Start with the assumption that you need the filter system.

It needs to be redundant. It needs to be seismic Category 1. It should be a certain flow rate so it can be tested and maintained properly. It needs appropriate instrumentation so you can monitor the performance of that filter system and with appropriate control room readout during an accident.

The ANSI standard has been published, ANSI-N509, which gives the basic design guidance and qualification testing criteria for each of the components in the filter system.

Now one of the major changes in the regulatory guide from the previous revision was that the high efficiency particulate absolute filters had previously been sent to a Department of Energy-sponsored filter test station for retest -- the qualification retest prior to installation, prior to shipping and then installation in the filter system at a commercial nuclear power station.

Four years ago the Staff reviewed the data available from the Department of Energy facilities and concluded that it was no longer necessary to send those filters cross-country to the station and then to the site. This conclusion is being rethought now by the Staff and the applicable data is being rereviewed to determine if we should stick with this position or change it on Revision 3.

Activated carbon is the only adsorbent used in this country in the controlled systems filter for radioiodine removal. We are worried about the maintenance and the accessibility of the filter system.

There needs to be some space between the componen s for that accessibility. There should be permanent test probes in the system so you can perform in-place testing.

The in-place testing is done per the plant's technical specifications. There is a section in the tech specs that refers to the testing of the control room habitability systems. I have a specific plant example and a standard technical specification example that I will quickly put on the board.

Visual tests need to be done for any obvious deficiencies. Air flow distribution tests to make sure that each section of the filter is seeing its appropriate flow. The HEPA filter banks need to be in-place-tested with DOP to verify it is 99.9 percent leaktight and then the Staff will assume there is a 99 percent particulate removal credit during an accident.

There is a Freon leak test done on the carbon bank.

MR. WARD: Could you explain that, Ron? What do you mean by the 99 percent particulate removal credit? You say that is assumed. Why isn't that tested?

MR. BELLAMY: The in-place leak test, the HEPA filters up here are each individually tested to a particulate filter removal efficiency of 99.97 percent in-place leak test. It grows to 99.5 percent due to potential degradation of the system.

MR. SHEWMON: If they have not leaked they have not degraded. Is that the philosophy?

MR. BELLAMY: That is correct. And the Staff uses 99 percent as a conservative number.

A similar philosophy for the carbon. A leak test is done and then a sample of carbon needs to be removed periodically and sent to a laboratory offsite for radioiodine removal testing.

MR. WARD: Is there -- that ANSI standard I guess specifies the test method. Is there a specification? What is the specification for iodine removal?

MR. BELLAMY: It depends on the bed depth and the credit that you were assigned to that filter system. It is in the high 90s. It is 95 -- 90 or 95 percent generally and the frequency of this in-place test is generally on an annual basis. The technical specifications talk about 18 months and a carbon test of 720 hours. But for a general guideline you can say it is on an annual basis.

MR. SHEWMON: Ron, you have about used up your time. What do you want to go to next? Do you have a slide you could use last?

MR. BELLAMY: I do not really need to put any other slides up except to indicate that the in-place testing criteria on this last slide --

(Slide.)

-- is very clearly included and identified in plant technical specifications as both limiting conditions for operation and surveillance requirements and if these in-place testing criteria are not satisfied, then the plant does have seven days to either repair the applicable filter system or be in cold shutdown, and that is clearly specified.

MR. SHEWMON: So what -- this system is a standby system and it is just a ventilation and temperature control that is used during normal operation. Is that roughly correct?

MR. BELLAMY: Generally speaking, the ventilation and temperature control is a separate heating and ventilating --

MR. SHEWMON: But that operates continuously and the rest of you call in only if you need it?

MR. BELLAMY: Correct.

MR. SHEWMON: Any questions?

MR. MOELLER: I think a comment might be in order on his item up there, number 2. The HEPA filter is not sent to the DOE test facility. Last week, at the 17th Nuclear Air Cleaning Conference there were two papers given that pertained to this subject. The first one was a summary of the rejection rates. There are three of these DOE laboratories that do these tests. The latest data for 1980 showed that 27 percent of the filters received at the Richland, Washington, test station were not found to be acceptable.

A second paper was presented in which a company developed some specifications and published them for bids. They put on paper a bogus company. They asked for bids for HEPA filters. They purchased filters from seven different companies, had all of them sent to the DOE lab for testing, and not one of the filters passed the test.

MR. SHEWMON: Is there a quality check? I mean, could you say gee, they did not get 99; they got 98.8 instead? Or was it 50 percent they got or how far out?

MR. MOELLER: The rejections were for a variety of reasons, but as the speakers said at the meeting, none of the rejections were on some minor point. They were all significant failure.

MR. SHEWMON: Thank you.

MR. MOELLER: Thank you, Ron.

MR. KERR: Being from Detroit, I am inclined to ask were any of the filters from Japan?

(Laughter.)

MR. MOELLER: No. These were all U.S. manufacturers.

MR. KERR: Then I have a suggestion.

(Laughter.)

MR. MOELLER: The next speaker is Louis Kovach, again whom I have already introduced, who will be talking principally about his own experiences in reviews and evaluations of the air systems for nuclear power plants.

MR. KOVACH: Good afternoon. I will try to squeeze everything into the ten minutes.

I would like to start out with saying that we just recently had a chance to evaluate some Japanese filters and they failed.

(Laughter.)

And it is very refreshing to run into some Japanese product that does not work as well as a U.S. product. I cannot say the same thing for the overall ventilation systems.

In the discussions and the specifications that we have currently in existence we are designing plants. Currently we are installing some into the plants, but these happen to be old plants that you gentlemen collectively so far have not licensed. Unfortunately, almost all of the plants that are licensed today have far inferior systems to the ones that are required by the current specifications.

Additionally, many of these facilities are not being operated or maintained in a manner that would be required to protect control room personnel. This is generally for almost all filtration systems, not exclusively for control rooms.

I have -- the Subcommittee meeting went through a large number of filter system test reports of various operating utilities and all of them showed major defects. Some of these may be loopholes in the regulations. Currently the practice is to require testing at least every 18 months. The normal practice is to test the system and fix it and test it and fix it and test it until you finally meet the end requirement.

That end requirement is what ends up being reported by most utilities to the NRC and various estimates were given at to what the actual test was the first time it was performed. So naturally on this basis the end results are always showing 99.95 and the system is presumed to be everlasting because all end reports show every high efficiencies.

The point of the work that is included in getting the systems back up to this level is very rarely reported and the initial test reports are not required, so if there is any major requirement, in my opinion, in changing Regulatory Guide 1.52, which specifies the current requirements for testing, it would be to actually report te-- results as they are obtained and then specify the fix separately.

This type of reporting would give a much better history whether that 18 months is adequate and whether we should go to shorter or longer intervals, depending upon the systems.

Additionally, almost all of our currently-operating plants are the very early generation filter systems where we had had structural problems. We had maintainability problems. Filter systems are built on a basis that it is very difficult to maintain them, even under cold conditions.

If we assume maintaining them or exchanging filters in an actual loaded activity-containing position, some of them are nearly impossible. The same is true for many of the non-radiological problems. I had seen test evaluation reports based upon extrapolation of chlorine absorption capacity for carbon and other conditions that have nothing to do with actual chlorine exposure in the control room. And if we are taking realistic chlorine levels, the systems that we have installed now, these filter systems would be incapable of holding chlorine longer than for a few seconds.

There are absorption systems available to handle almost any of the chemicals that were discussed in the Subcommittee report. Very extensive research work was conducted both by the United States and some other countries during the Second World War in various chemical protection systems relating to gas masks. All of this information is available and on the basis of this you can design filter system protection that can permit shirtsleeve atmosphere in the control room. But I am not aware of a single control room to date that is so equipped.

The frequency question that came up of how often something like this may occur -- and I would like to preface it that the information I have is secondhand -- but at one of the reactors that is currently not operating yet during its construction period the construction crew had to evacuate three times because of chemical spills nearby. This gives you an idea as to the frequency at least of a particular site that is located in an area where chemicals are dropped.

MR. SHEWMON: These spills were not onsite but by one of the neighboring chemical complexes, is that correct?

MR. KOVACH: That is correct.

So naturally, based upon the location of the plant, in some cases this could be a critical problem. In other cases, it would not be. This is, I think, much more site-related than even the radiological consequences of an accident.

But even from a radiological standpoint, the currently operating systems are greatly undersized. They are much smaller than the plants that we are designing now and many of these systems are inadequate even for the undersized operation. Some of these systems would not be able to operate longer than a few hours. Some of these systems leak very badly. Many of them are located together with other filter systems for other areas of the reactor and cross-contamination possibilities exist.

And in many areas we have significant problems relating to the filter systems. For those of you who are interested, an actual listing of it in the Subcommittee minutes, we had long lists of randomly-taken test evaluation of control room filter systems and I do not want to go over these.

Additionally, I would like to make one other comment, that I happen to be chairman of a group looking at filter system behavior under accident conditions for OECD, and this problem that we have on reactor control room habitability is not just a U.S. interest. Other countries are significantly interested also in trying to generate information relating to the problems.

And at the same time I have to say that comparing leaks on a general basis, filter systems as they exist in most of the European countries, the protection capabilities of these filter systems is significantly higher than ours mainly because of the significant conservatism used in the design of these systems and the much stronger cooperation between chemical process engineering personnel in designing the systems, and not only HVAC-type personnel from a heating and ventilation standpoint. Particularly at the early stage, many of the filter adsorber trains installed in Europe were designed based on the chemical industrial experience and not on a pure HVAC-type concept.

Thank you very much.

MR. MARK: You mention that people do know how to build filters which would be much preferable to the ones now being used. What is the approximate ratio of the cost of one that is done well versus one that is normally acquired?

MR. KOVACH: I would say that the early filter systems that were built versus the current ones that are designed to the various ANSI and Reg Guide 1.52 criteria, the cost range is probably is the neighborhood of 5 to 1 to 7 to 1.

MR. MARK: A good one versus a standard one.

MR. KOVACH: Yes.

MR. SHEWMON: These are standard for old plants now.

MR. MARK: Well, I was thinking of the ones that you were telling us about that were designed during World War II and capable of chemical protection.

MR. KOVACH: If you are looking at chemical protection systems, the only cost I could say that for a typical control room you would be looking at about a quarter of a million dollars installed for chemical protection in addition to radiological protection.

MR. MARK: Whereas the radiological alone --

MR. KOVACH: On the current systems the radiological alone would be probably in the neighborhood of $100,000 for radiological alone in the control room systems.

MR. MARK: And another simple-minded question. These things are tested. How much commotion, how much exertion is involved in conducting a test? Do you have to stop the plant? I wouldn't suppose so.

MR. KOVACH: No.

MR. MARK: Is it a matter of an hour or a morning's work or what?

MR. KOVACH: The actual test itself is about ten minutes. Generally fixing the system well enough so that it passes the test can be several says.

[Laughter.]

MR. BENDER: You heard Dr. Bellamy's description of the system that the NRC now approves. How are the European systems different than those?

MR. KOVACH: The main difference is they are using up to 50 centimeters, plenums up to 50 centimeters, while we are using plenums up to 5 centimeters.

MR. BENDER: Are you talking about the carbon?

MR. KOVACH: Yes. They are talking about ten times longer residence times in the adsorber.

MR. BENDER: Are there any other significant differences that you can think of?

MR. KOVACH: No major differences.

MR. BENDER: Thank you.

MR. MOELLER: Lou, you mentioned, of course, the importance of a filter system, a good air cleaning system for protection of the operators and so forth at the Subcommittee, though, and I think the Committee should hear this at least from you. Did you not tell us that your experience has shown that there are operators and plants that because of their lack of confidence in the air cleaning system for the control room, that they actually might even fear to stay there in case of a challenge to that system?

MR. KOVACH: Yes, that is the case. There are some areas where we involved our personnel. I personally was involved in testing where they got very upset every time we run the test because the system has to operate and the habitability of the control room deteriorates to the point that there is actual discomfort.

MR. MOELLER: Even during a test?

MR. KOVACH: Even during the test.

MR. MOELLER: And how long was the test and how long is the system supposed to be able to operate?

MR. KOVACH: The test durations for that particular system are about one hour total time.

MR. MOELLER: And the system should be able to go for days?

MR. KOVACH: Yes.

MR. SHEWMON: The discomfort is humidity or temperature.

MR. KOVACH: Mainly temperature in that particular case.

MR. SHEWMON: Okay. Thank you.

MR. MOELLER: Thank you very much.

The next speaker, then, as I mentioned earlier, is Leo Klaes from TVA, and you, again, have a handout for his presentation.

MR. KLAES: My name is Leo Klaes and I am senior mechanical engineer in the Environmental Control Systems Section of the Nuclear Engineering Branch of Tennessee Valley Authority. I also have with me Mr. Steve Ness, who works in the Radiation Protection Analysis Group of the Nuclear Engineering Branch in case you should have some questions with regard to those aspects of our design.

I will only hit the high spots today, because of time constraints, on the talk that we presented at the Subcommittee meeting on May 14th. The features I will describe of the main control room habitability are based upon our Sequoyah Nuclear Power Plant, which is our most recently licensed plant. They are, however, generally applicable to all of our plants. But as Dr. Kovach mentioned earlier, the earlier plants, of course, do not meet all of the latest requirements.

[Slide]

This slide shows the general arrangement of the Sequoyah plant and the relationship of the control building to the other major buildings in the plant, here in a planned view and here in a second view.

(Slide)

This slide shows the general configuration of the habitability enclosure zone and those portions of the building that are covered by the habitability zone. This is a roof view showing the approximate location of the two intakes which are used, and they are approximately 250 feet apart.

(Slide)

This slide shows the main control room habitability design considerations which we address, the major ones. There are others, of course, such as maintainability, which I do not have listed here. To some extent that comes under system reliability. Radiation hazards, their sources, protection features, dose analysis results, toxic hazards, natural hazards, environmental control considerations, fire protection and system reliability.

(Slide)

This slide identifies the radiation sources, the gamma and beta sources due to radioactive air that enters into the control room from ventilation systems that are in operation, personnel access and other leakage paths, post-accident gamma sources surrounding the main control room due to releases from the containment into the environment and post-accident gamma sources from the primary containment atmosphere, post accident gamma sources in the auxiliary building due to in-leakage from the containment, and finally, ingress and egress between the main control room and the site boundary.

MR. BENDER: Dr. Kerr had earlier asked today in another review what you use as your source terms. What do you use as the source terms that you just mentioned? You know, you say beta and gamma sources and post-accident gamma sources. What specifically do you use?

MR. NESS: We use the Reg Guide 1.3 or 1.4 source terms in the containment due to a loss of coolant accident and use that inventory as the source for essentially the TID.

MR. BENDER: It is based upon the containment leakage as specified?

MR. NESS: Yes.

(Slide)

MR. KLAES: This slide identifies the

radiation protection features we incorporate into our design. First of all, we have a monolithic concrete structure which has heavy walls and floors and ceilings surrounding the main control room habitability area. Those doors which penetrate the monolithic structure are adequately protected against radiation. In some cases they are lead-shielded doors.

We employ as part of the design a low-leakage enclosure which is designed to minimize leakage paths so that we can have a minimum of supply air to maintain a pressurization feature, which I will discuss later, in order to minimize the ingress of noble gases into the main control room

We have radiation monitors which activate alarms and initiate emergency operating features. We have restricted flow emergency pressurization which ties in with the low leakage enclosure and air cleanup of emergency recirculated and pressurization air, and then, of course, portable breathing apparatus and protective clothing that are available if the other features do not maintain the level adequately low enough for personnel occupancy.

MR. MARK: Do those radiation monitors distinguish between gamma rays and beta rays?

MR. KLAES: Do you know the answer to that, Steve?

MR. NESS: I'm not sure about the monitors, what type monitor we have in there. We have area monitors in the control room itself, which would be a gamma, an area monitor, and there are monitors in the ventilation system but I'm not sure what type of monitor there is there.

MR. MARK: There are monitors that would pick up shine if it was there, but they would also pick up beta emitters if in there.

MR. NESS: Right. I'm not sure what type they are in the ventilation.

(Slide)

MR. KLAES: This is a very simplified diagram showing the environmental control systems that we have in our plant. The portion in blue indicates the normal supply air system, in this case 2000 cfm system with redundant active components, and various monitors, radiation, smoke, chlorine and high-temperature monitors, and this supplies air into the habitability zone and then it is taken by these air-handling units which recirculate that air and send it through coolers which temper the air and provide the required conditions for personnel comfort and/or equipment requirements for that critical equipment in the control room habitability area.

Now, if you have a high radiation signal or -- and I will discuss later some of these other conditions -- we will shut off the normal supply air, and the toilet room exhaust is secured and we go into what we call an emergency pressurization mode where we use this system in yellow which supplies 200 cfm of air in this case directly into the inlet of an air cleanup system, two of them here, so they are redundant, which is also mixed with 3800 cfm of return air from the normal air conditioning system. And then that is supplied into the air conditioning system and circulated throughout the plant.

We also have a radiation monitor located in the emergency pressurization system, and the radiation and smoke monitors located throughout the area.

MR. MARK: You spoke of the cleanup systems having these two, these redundant tasks.

MR. KLAES: Yes.

MR. MARK: Suppose one of them leaks like crazy and the other one is clogged up. What happens?

MR. KLAES: Then you wouldn't get the efficiency out of either one of them that you are required to.

MR. MARK: But you would still get the air.

MR. KLAES: Yes.

MR. MARK: Is that in some way monitored?

MR. KLAES: Well, you have a radiation monitor in the space, which, of course, would detect a buildup of radiation if those systems did not properly perform. And then, of course, if that should occur, then the backup is your emergency air breathing and protective clothing.

MR. KERR: How would Mr. Kovach rate your system? Would he think it a modern, up-to-date type system?

MR. KLAES: I think he should answer that. Possibly in terms of the design concepts, yes. I think in these earlier plants they do not meet all of the requirements of ANSI and 509 as far as accessibility and that sort of thing for maintenance.

MR. KOVACH: I would comment on that for you. I would rate that one very highly.

MR. MOELLER: That is why we invited him in.

MR. WARD: Could I ask you a question here? If there is an incident and these things are working and filtering out radioactive contaminants and one system begins to leak, you get an indication of that. You shut it off. Are the loads on the carbon or the HEPA filters in terms of radioactivity great enough to get any significant heat load? Have you looked at that? Are the loads very tight?

MR. KLAES: Yes, we have looked at them and they are very small in these areas. We do have some systems in other areas of the plant. The emergency gas treatment system that operates in the auxiliary building does have a potential for high heat buildup if you secure the unit. So we have a recirculation mode that continues putting a small amount of air through that system. But in this particular one there isn't too much reactivity.

MR. MOELLER: Leo, we have used up a lot of your time with questions. Try to wrap it up if you can in just a minute.

[Slide]

MR. KLAES: The next slide simply compares our radiation dose calculated against the acceptable doses, and it i' w ll within the requirements.

(Slide)

The toxic hazards other than radiation. We analyze the toxic hazards in accordance with Regulatory Guide 1.78, the chlorine in accordance with Regulatory Guide 1.95, and we also consider high temperature and smoke. On this particular plant we identified as potential hazards high temperature, smoke and chlorine, and if you will remember, they were shown, detectors were shown for those in the diagram that I showed you, and the design features for protection, of course, in addition to an alarm, we also activate the emergency modes of operation, except in the case of chlorine. We do not activate the emergency pressurization system, so that in that case we have essentially a nonpressurized system.

MR. MARK: In spite of Dr. Moeller's admonition, the picture you had up before of the filter trains, the habitability enclosure, the toilers are outside the habilitability area?

MR. KLAES: No, they are inside.

MR. MARK: This little arrow outside the picture?

MR. KLAES: That shows the toilet room exhaust.

MR. MARK: I'm sorry.

MR. MOELLER: That is a good feature. In other words, you can plan to stay there for quite some time.

[Laughter.]

MR. SHEWMON: Please proceed.

[Slide]

MR. KLAES: The natural hazards we consider are seismic qualification, of course tornado analysis, both pressure transients, wind and missiles, and, of course, flood. I won't go into any detail on those.

(Slide)

Environmental control, temperature control capability for both personal comfort and equipment. We maintain, as I mentioned, a slight positive pressure for noble gas control, an isolation capability for accidents, and then an air cleanup capability for accidents.

On fire protection, we use noncombustible equipment wherever possible, administrative control over the use of papers and log sheets within the main control room habitability to prevent a buildup of potential fire hazards. We have local smoke detectors throughout the main control room habitability zone, and those are redundant and serve from separate power sources so that if one should fail, another one is there to detect the fire.

The fire dampers and fire doors. Where fire dampers are used in systems and an inadvertent closure of a fire damper might cause overheating of a critical space or equipment, we provide double fusable links so that if one of them fails, we would still have one more to hold the damper open. And, of course, portable fire extinguishers which are readily available to put out any fire that we should be unfortunate enough to have.

And if all else fails, we have an auxiliary control room located in the auxiliary building and served by a completely separated ventilation and air conditioning system with no possibility of or little possibility of any interchange.

MR. SHEWMON: Why don't you let us read this as you make any closing comments you would like to.

MR. KLAES: Well, I guess really that is the last one I have, and I think I have covered all of my points.

MR. SHEWMON: Fine. Any questions?

[No response.]

MR. SHEWMON: Thank you very much.

MR. MOELLER: We will close, then, Mr. Chairman, with the presentation by William Miller from Sargent & Lundy.

MR. MILLER: I can spend the first couple of minutes just introducing myself. I am Bill Miller and I started working in the nuclear business with Sargent & Lundy about 12 years ago. I broke in designing control room habitability systems, and while I am not designing them any more, I have kept up with the state of the art. Up until a year ago, I was head of the HVAC Division at Sargent & Lundy. I have brought with me today Steve Ornburg, who is the current head of the HVAC Divisin.

I had prepared a 15-minute presentation and I had a text I was going to read from, but I really think that is inappropriate in light of the remainder of the vacuum that is left, so I will just hit a couple of high points and then we can wrap it up.

We are a nongoverrment agency, so we can afford to make some pretty slides. For any of those who are interested, I believe this is the operative slide. I believe this is the La Salle County control room. That is the most recent plant to receive an operating license, if I remember correctly.

(Slide)

Let's just look briefly at what our control room habitability HVAC system is. We have got an air conditioning portion of it which we have shown over the right separately, and then the air cleaning portion, which is on the left here, 3000 cfm versus 25,000 cfm. We do recommend that the two systems be kept separate for a number of different reasons. This duct work is large, and leakage in that duct work can cause problems with this unit, so we like to keep this system separate.

(Slide)

In the presentation I gave to the Subcommittee back in May we talked about the design bases, the guides, the methodology, the equipment construction, acceptance and pre-op testing, surveillance and periodic testing. I think what you really want to know today is whether the engineers know what they are doing in designing the system.

(Laughter.)

So let's go right to the design methodology.

(Slide)

We have to establish the habitability envelope. This is a very important part of the design of a system. The amount of area that is adjacent to the control room and above it and under it that you are going to include in the system design is critical because as that increases, so do the potential leak paths, so you don't really want to include anything more than you have to but yet there are areas that you have to cover, like toilets and kitchens and other areas that have to serve people who have to maintain their positions in the control room.

We then determine the preliminary equipment locations for the heating and cooling equipment. We would prefer that that equipment be located within the habitability envelope. It really cuts down on the number of problems that we have with duct work leakage.

We calculate the heating and cooling loads using conventional methods that are dictated by the American Society of Heating, Refrigeration and Air Conditioning Engineers and the wealth of experience that we have gained on the loads generated by the equipment over the years.

We established that primary air flow, which I said in an example was 25,000 cfm, we estimate the habitability boundary leakage by actually looking at the number of penetrations and the number of doors in the walls, calculating the leakages using commonly available references and then apportioning that leakage to the various disciplines, mechanical, electrical and structural, who will be responsible for designing the system. We used that leakage to determine what minimum makeup flow rate that will be required to maintain that minimum positive 1/8th inch pressure in the control room.

And then the real work starts in the Murphy-Kamp paper. There is a reference made that there is a great deal of different types of systems and models that are available to perform this protection function on the control room, and we at Sargent & Lundy happen to like the model where we recirculate a portion of the air from the room back through the cleanup system as well as clean up the outdoor air that we require.

(Slide)

This is what I was going to use this slide for. We like this. Isolation with filtered recirculation and pressurization. But there are different methods available, some of which give a better protection than others.

(Slide)

And then we calculate a bounding radiological iodine protection factor using some very simple equations that are in the references.

(Slide)

This is the type of calculation that we would use. Now, you have to keep in mind that this is a simplistic analysis that really doesn't include the actual leakages of potentially contaminated air that you could get into the ductwork that is between these various filters and the control room.

(Slide.)

We're just trying to get a bounding number in that. The iodine protection factor for a typical control room would be about 250.

(Slide.)

After we calculate that bounding IPF we will go in and we will calculate the hazardous chemical concentrations, and we will select the type of system which meets the limits that we have imposed. Then we will start to lay out the ductwork and the various equipment, and we will calculate the equipment and the ductwork leakage, and this is when the real work starts.

(Slide.)

Because we take the system and we analyze the air cleaning portion of it separately than the air conditioning portion.

(Slide.)

I will show you why. When we use ANSI-N509, which is the standard that Ron referred to, that kind of dictates how you design these air cleaning systems. You have to designate the different leakage classes that the different ductwork will be constructed to.

(Slide.)

And then you go through and you derive an equation which describes what the reduction in your protection factor will be for that leakage.

(Slide.)

And you come up with a protection factor or a reduction in your iodine protection factor, and for this particular system we made the ductwork pretty tight. We have got 11 CFM unfiltered in leakage, 16 CFM filtered out leakage, and we have a very modest reduction in the iodine protection factor.

(Slide.)

But now let's look at this air conditioning portion of the system. Now, this is the system that has got the larger ductwork, and it is leakier.

(Slide.)

We derived the equation that describes that portion of the system.

(Slide.)

And we calculate an iodine protection factor which is unacceptable. We need about 100, and it's down at 44, and this is using duct construction which is still good but just not enough tight enough.

(Slide.)

And then we go back through an iterative process and get that leakage down to the point where we can get the 100 iodine protection factor, and that's only 18 CFM of unfiltered in leakage, which probably means we will have to weld that ductwork up tight.

So the point we're trying to make, and we made it a couple of years ago at the air cleaning conference in a paper we wrote, is that when we go through and we analyze these systems, we cannot just look at the air cleaning portion of the system. We have to look at the whole system. We have to look at the air conditioning portion also. That is pretty obvious. But it is not really in the regulatory requirements except in the general design criteria.

(Slide.)

Then we have got the habitability and the leakage in the final design. We go back. We recalculate the concentrations and the control room IPFs and establish all the leakages. We have got to reconfirm the heating and cooling loads.

(Slide.)

This takes place over a number of years. And then I could get into a section on acceptance and pre-op testing. And I want to make a comment about that because it became very important in our proceedings on LaSalle. We spent about ten years designing the systems, and we generated literally tons of paper calculating what the design should be, getting the design out on drawings, specifying the equipment.

But really when it comes right down to it, the most important part of the control room habitability system proof is the pre-op test, and it is when we go out there and we test these systems and we show that they can do what they are supposed to do, and on the plants that Louis has talked about, the older plants that are in such questionable condition, I wouldn't be the least bit surprised if there are reports that show that these systems when they were designed and installed were tested and did meet the criteria, or else they wouldn't have been accepted.

And the problem, I guess, with the HVAC systems is that many of us, probably most of us, consider ourselves to be dabblers with them. We love to fool with thermostats. We love to reposition dampers, and we love to get just things so much more comfortable. I think there is a lot of that going on in the business, and I think that is what we have got to fight.

That's all I have to say.

MR. SHEWMON: Thank you.

MR. MOELLER: Questions for Mr. Miller?

MR. SHEWMON: How thick are the carbon beds at LaSalle?

MR. MILLER: I believe on the control bed we've got deep bed carbon filters. We have four-inch thick trays.

MR. BENDER: Is it common practice to weld the ductwork together?

MR. MILLER: It is common practice to have welded joints wherever you need minimum leakage, but for the most part in your air conditioning, the air conditioning systems, you try not to need it. It is much more economical and easier to maintain. If we could have phlange joints, bolted plange joints, and we can get pretty good leakage characteristics with them, too, if they are properly designed.

MR. BENDER: Do they stay tight?

MR. MILLER: Do they stay tight? Well, we think so. That is another, I think, weakness in let's say the state of the art. It is very difficult to go back and to leakage test individual joints after the whole system is installed. I don't know who I was talking to, Mr. Bender.

MR. BENDER: Smoke tests and things of that sort are commonly used for testing such systems. Are they any good?

MR. MILLER: Well, the smoke tests are kind of out of date. We would prefer that when the contractor is erecting a ductwork that he leak test the ductwork a few joints at a time when it is going in, and then he can put in a blank off plates, and he can do it fairly easily, and that is how a lot of plants are being done today.

The smoke test was sort of a test after you were all done and you wanted to find the gross leaks. There are better methods available which aren't as offensive as smoke tests.

MR. BENDER: That's after the system is assembled.

MR. SHEWMON: If you have a few joints, what does he look for? Do you use soap bubbles, a helium leak test or what?

MR. MILLER: Helium is a very good test for leakage, although I haven't seen it applied in ductwork systems. Soap bubble, a simple soap bubble test, turns out on positive pressure systems to be one that is used, and it yields good results. But on the negative pressure system it really doesn't work too well.

And those are the systems when you are upstream of, it is the negative pressure ducts on air cleaning systems that sometimes give you the biggest reductions in iodine protection factors.

MR. MARK: I believe you said that in the design you calculate the leakage. Then presumably when assembled you measure something or another.

MR. MILLER: That's right. I was talking about calculating the leakage on the boundary of the walls and the doors, and that is backed up by experimental data that was conducted some 15 years ago. I also talked about calculating the ductwork leakage, and those calculations are based on tests that we have run. And then we on our newer plants would define the leakage criteria that when the contractor is installing the system he would have to test that that meets. But after the system is started up you test the effectiveness of the habitability system by pressurizing the control room and proving that you can maintain that eighth of an inch positive pressure. That would be the final test for the control room.

MR. MARK: Is common experience also what you have calculated?

MR. MILLER: Our experience shows that what we calculate is what we can achieve, but we often have to do it in terms of building modules after we have done some caulking.

MR. BENDER: Excuse me. One last point. The joints that are used in these air systems I suppose are individually selected by various design organizations. Is there any standard that is available?

MR. MILLER: There are two sources for duct construction standards. The most commonly used by the sheet metal industry is the SMNCS standards, the Sheet Metal National Contractors Association. The types of construction which are acceptable for air cleaning systems have been delineated in ANSI-N509, and then ASME is currently, for the last five years have been writing a code for all of these systems, and the first code sections are due to be reviewed by the Nuclear Codes and Standards Committee of the ASME later this year. And so we will have the code that we have needed for so long hitting the streets.

MR. BENDER: Very good. Thank you.

MR. EBERSOLE: Is the use of any gaskets or plastic seals disallowed?

MR. MILLER: Plastics are definitely disallowed, but we have used some neoperene.

MR. SHEWMON: Gentlemen, I would like to call this to a close, and I would like to thank the speakers for coming in. It has been a very informative session.

(Whereupon, at 5:35 p.m., the meeting was adjourned.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ACRS/268th General Meeting

Date of Proceeding: August 12, 1982

Docket Number: ____________________

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Patricia A. Minson

Official Reporter (Typed)

Patricia A. Minson

Official Reporter (Signature)

NUCLEAR REGULATORY COMMISSION

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Ray Heer

Official Reporter (Typed)

Ray Heer

Official Reporter (Signature)

ACRS PRESENTATION

AUGUST 1982

GRAND GULF NUCLEAR STATION

MISSISSIPPI POWER & LIGHT COMPANY

MIDDLE SOUTH ENERGY, INC.

MIDDLE SOUTH UTILITIES SYSTEM

ACRS FULL COMMITTEE
AUGUST 12, 1982
GRAND GULF NUCLEAR STATION

A G E N D A

8:30 AM - OPENING REMARKS

8:45 AM - REPORT OF ACRS GRAND GULF SUBCOMMITTEE (AUGUST 11, 1982)

9:00 AM - REPORT OF ACRS FLUID DYNAMICS SUBCOMMITTEE (JULY 29-30, 1982)

9:10 AM - STATUS OF REVIEW

9:25 AM - PROJECT STATUS REPORT

9:30 AM - HYDROGEN CONTROL

10:10 AM - BREAK

10:20 AM - STAFFING - BWR EXPERIENCE

10:45 AM - JET IMPINGEMENT IMPACT ON CRD BUNDLE

11:15 AM - POST-ACCIDENT CONTAINMENT VENTING

11:30 AM - OTHER

12:30 PM - ADJOURN

STATUS OF REVIEW

AUGUST 11, 1982

ITEMS REQUIRING RESOLUTION PRIOR TO FULL POWER LICENSING

PMP DRAINAGE	UNDER REVIEW
CONTAINMENT ULTIMATE CAPACITY	RESOLVED
SQRT	OUTSTANDING
ENVIRONMENT QUALIFICATION	UNDER REVIEW
CONTAINMENT/DRYWELL PURGE	UNDER REVIEW
CONTAINMENT CONCERNS	OUTSTANDING
IEB 79/27	UNDER REVIEW
IBN 79/22	UNDER REVIEW
CONTROL SYSTEMS FAILURE	UNDER REVIEW
FAILURE RPV LEVEL SENSING	RESOLVED
INTERPLANT COMMUNICATIONS	UNDER REVIEW
DIESEL GENERATOR RELIABILITY	UNDER REVIEW
CONTROL ROOM REVIEW	UNDER REVIEW
POST ACCIDENT SAMPLING	UNDER REVIEW

Hydrogen Control

I. System Design/Qualification

II. Base Case Selection

III. Equipment Survivability

IV. Structural Capability

V. Local Detonations

VI. Testing

Hydrogen Ignitor System (HIS)

Description

- Ignitors Located in 90 Locations in the Drywell, Wetwell, Containment
  - 18 Ignitors Located in Drywell
  - 11 Ignitors Located in Wetwell
  - 61 Ignitors Located in Upper Containment
  - Distance for Adequate Separation/Coverage
    - One Train-Maximum Separation is 60 ft.
    - Two trains-Maximum Separation is 30 ft.
- Ignitor Assembly
  - GMAC Model 7G Ignitor
  - Welded Metallic Enclosure with a Spray Shield
  - Includes Access Provisions
  - Includes a Transformer for Voltage Stepdown
- Ignitor Power Supply
  - 120 Vac ± 10%, 60 Hz
  - Two ESF Divisions
  - Each Division is Separated into 2 Breakered Circuits
  - Remote Operation by Manual Switches in Control Room
- 1700°F Minimum Glow Plug Surface Temperature

HYDROGEN IGNITOR SYSTEM (CONT'D)

COMPONENT QUALIFICATION

- ALL ASSEMBLY COMPONENTS WILL BE QUALIFIED FOR:
  - SEISMIC AND HYDRODYNAMIC EVENTS:
    - ABSOLUTE SUM OF SSE + LOCA + SRVA
  - ENVIRONMENTAL CONDITIONS (IEEE 323-1974/NUREG-0588)
  - ENVIRONMENTAL CONDITIONS RESULTING FROM SUCCESSIVE HYDROGEN BURNS
  - SEISMIC QUALIFICATION PER IEEE 344-1975
- TESTING UNDERWAY - EXPECTED COMPLETION BY END AUGUST

OPERATION

- FOR EVENTS WITH POTENTIAL FOR EXCESSIVE HYDROGEN RELEASES [THAT IS, FOR CORE COOLING WITHOUT LEVEL RESTORATION, WHEN WATER LEVEL FALLS TO OR BELOW TOP OF ACTIVE FUEL (TAF)], MANUAL INITIATION OF:
  - HIS
  - CGCS (PURGE COMPRESSORS, VACUUM BREAKERS)
  - CONTAINMENT SPRAYS (TEMPERATURE MITIGATION)

BASE CASE SELECTION

- Realistic Initiating Event
- Realistic Scenario
- Basis for Sensitivity Studies
- Basis for Other Evaluations
  - Equipment Survivability
  - Systems Interaction
  - Procedure Application

Base Case Selection (Cont'd)

- Initiating Events Evaluated
- Recovery Events Evaluated
- Two Base Cases Result:
  - Stuck Open Relief Valve (SORV) - Suppression Pool Release
  - Small Break LOCA - Drywell Release

STUCK OPEN RELIEF VALVE

Initiating Events

- System Transient
  - Loss of Feedwater
  - MSIV Closure
- Inadvertent Valve Opening

Mitigating Events*

- Open Additional SRVs
- Initiate Containment Spray
- Energize the HIS
- Initiate CGCS

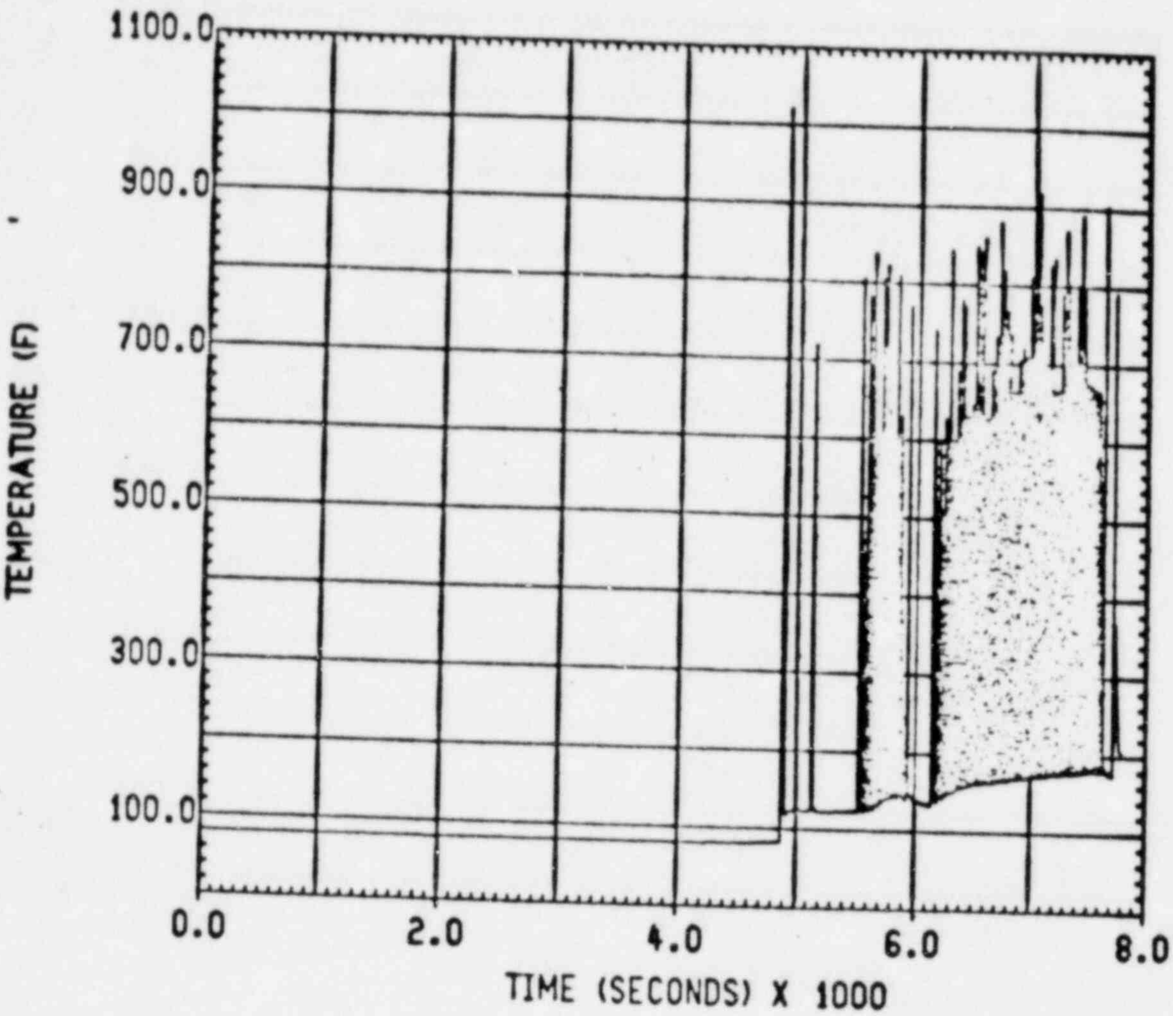
* Assumes no water available to the core

SORV BASE CASE DESCRIPTION

- MARCH Release Rates
- CGCS and Ignitors - Initiated at 20 Minutes
- Upper Pool Dump - Initiated at 30 Minutes
- 8 v/o Ignition and 85% Completion
- 6 ft/sec Flame Speed
- 1 Spray Train - Initiated After First Burn
- Wetwell Spray Carryover
- Forced Containment Burn

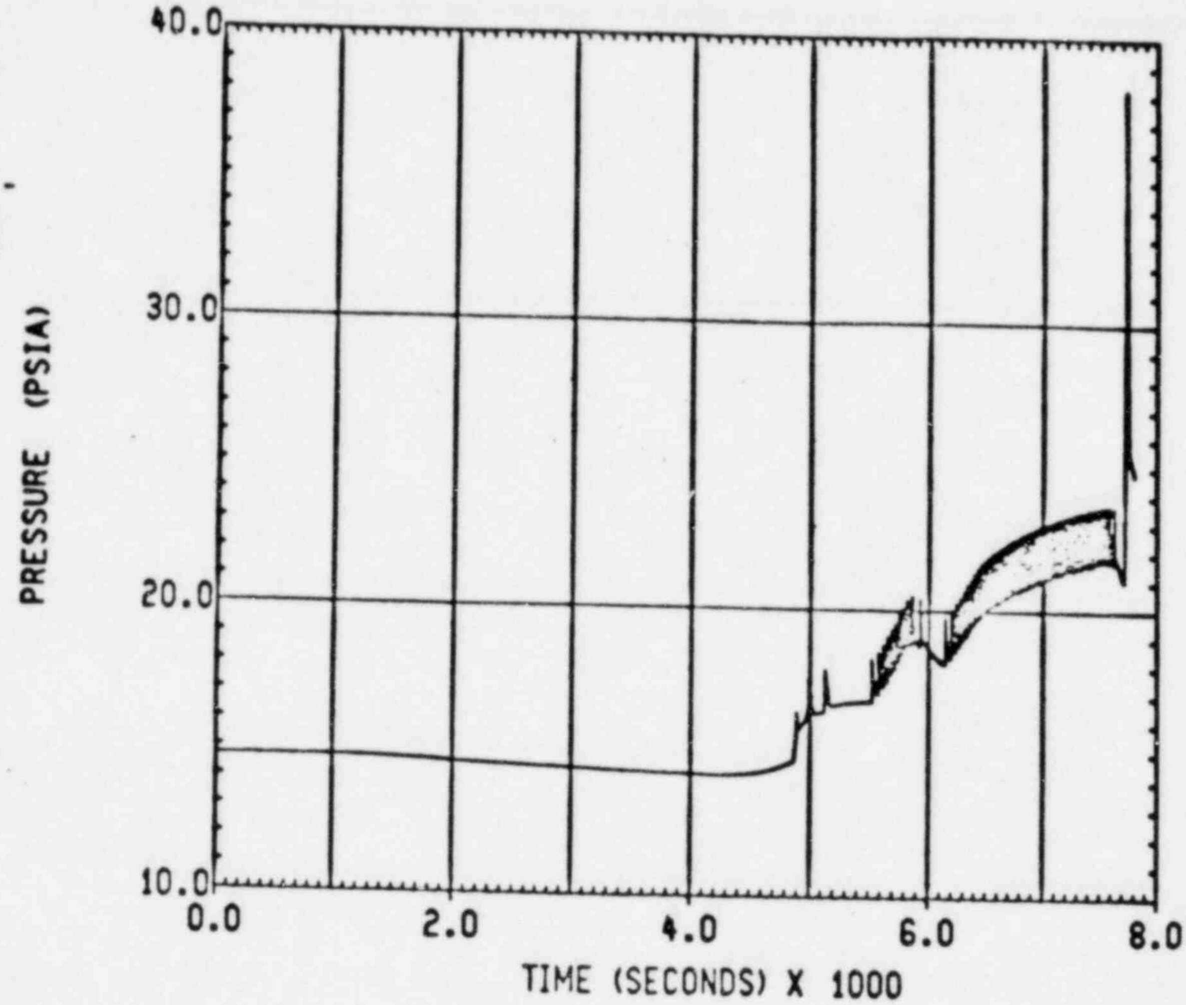
SUMMARY TABLE
SORV BASE CASE

	W/O FORCED BURN	W/FORCED BURN
NUMBER OF BURNS		
DRYWELL	0	
WETWELL	59	
CONTAINMENT	0	(1-FORCED)
PEAK TEMPERATURE (°F)		
DRYWELL	137	(193)
WETWELL	1020	(1020)
CONTAINMENT	197	(681)
PEAK PRESSURE (PSIG)		
DRYWELL	9.6	(18.6)
WETWELL	9.0	(23.5)
CONTAINMENT	8.8	(23.9)
PEAK PRESSURE DIFFERENTIAL (PSI)		
DRYWELL/CONTAINMENT		
FORWARD	4.2	(4.2)
REVERSE	0	(4.8)



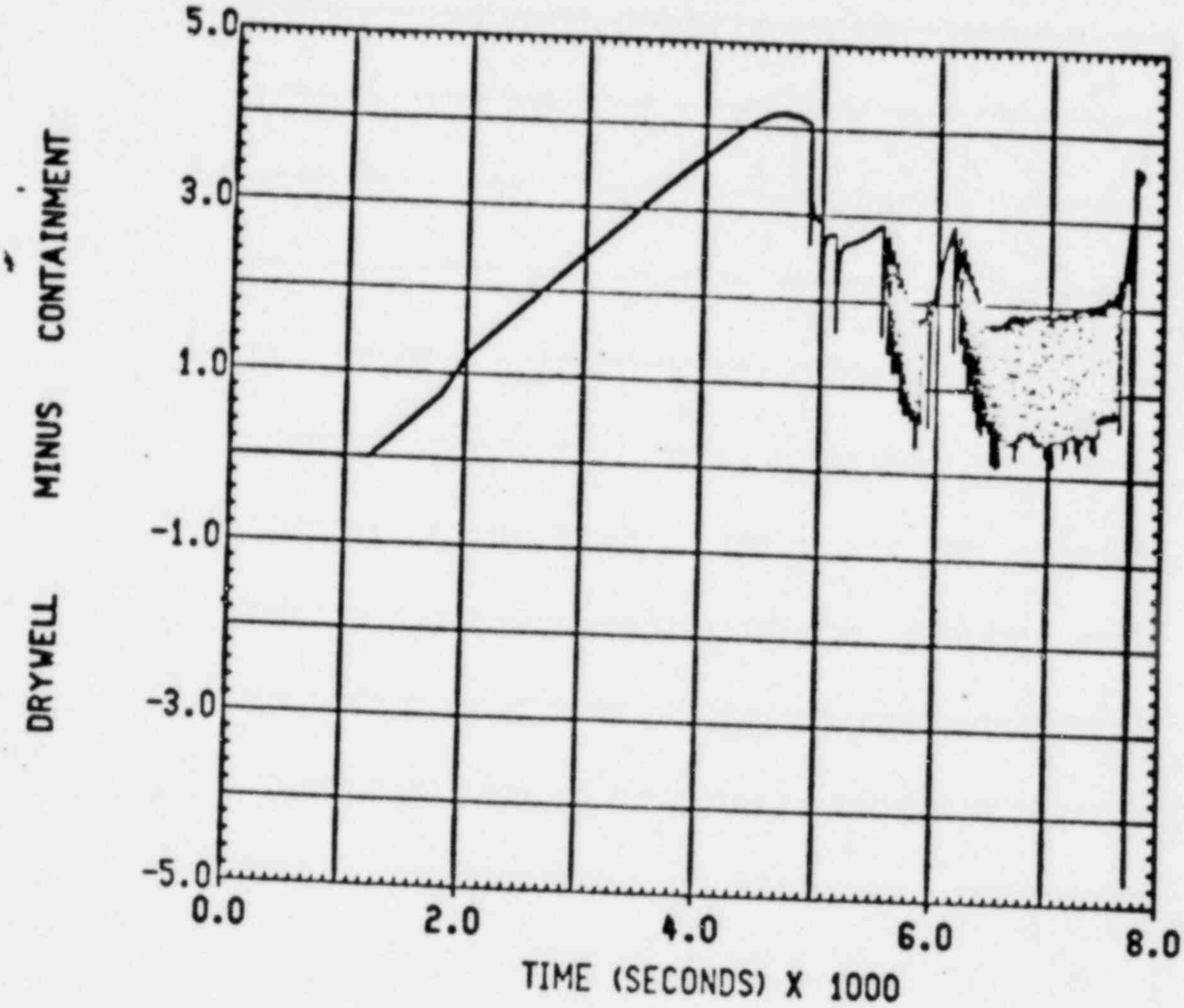
GGNS BASE CASE SORV

WETWELL TEMPERATURE



GGNS BASE CASE SORV

WETWELL PRESSURE



GGNS BASE CASE SORV

DIFFERENTIAL PRESSURE

SMALL BREAK LOCA

INITIATING EVENT

- RUPTURE OF SMALL/INTERMEDIATE SIZE PIPING

MITIGATING EVENTS

- OPEN SRVs
- INITIATE CONTAINMENT SPRAY
- ENERGIZE THE HIS
- INITIATE CGCS

Drywell Break Base Case Description

MARCH Release Rates

50/50 Release Rate Split at 20 Minutes

CGCS and Ignitors - Initiated at 20 Minutes

Upper Pool Dump - Initiated at 30 Minutes

Suppression Pool Drawdown - Initiated at 30 Minutes

8 v/o Ignition and 85% Completion

6 ft/sec Flame Speed

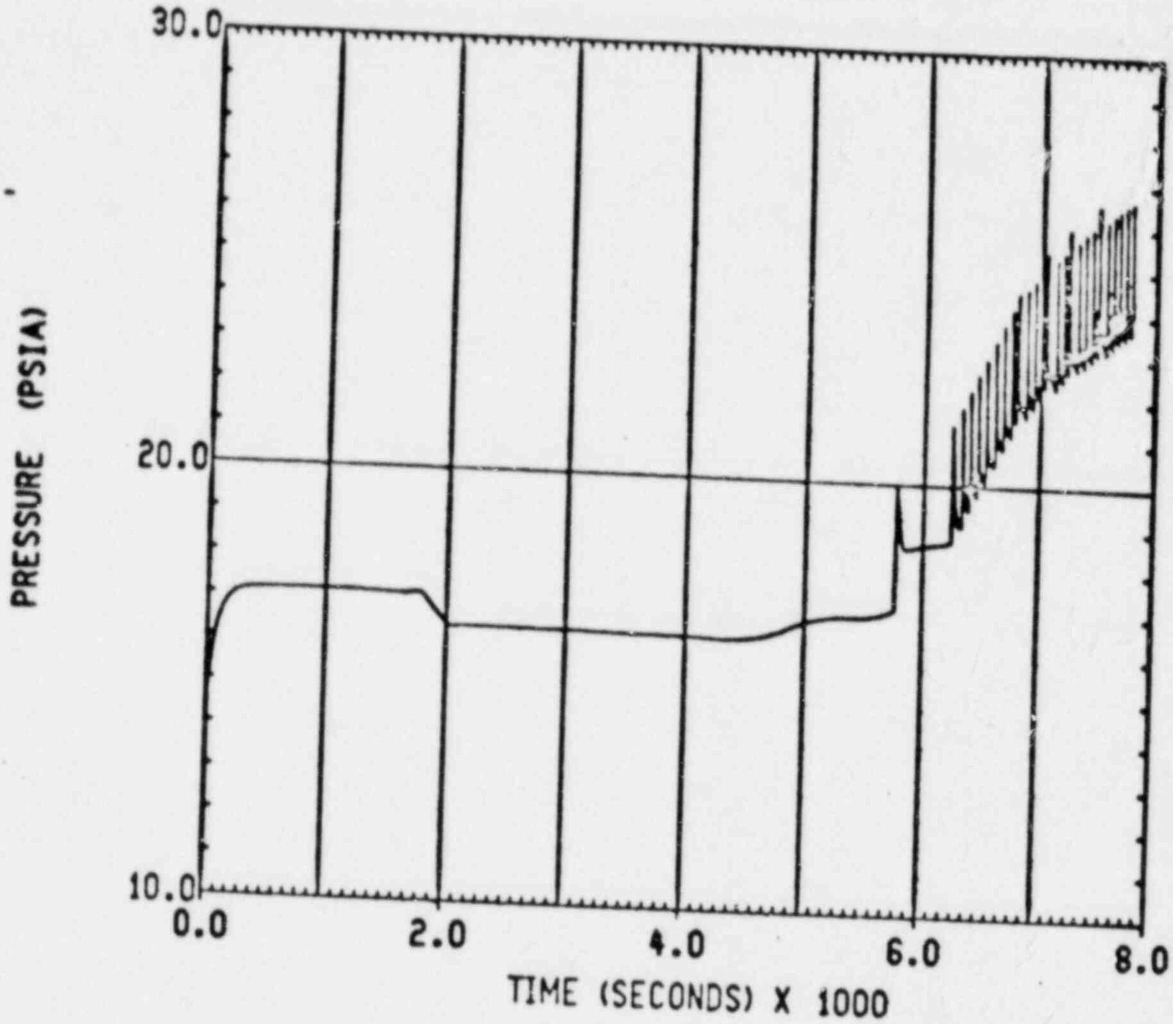
1 Spray Train - Initiated After First Burn

Wetwell Spray Carryover

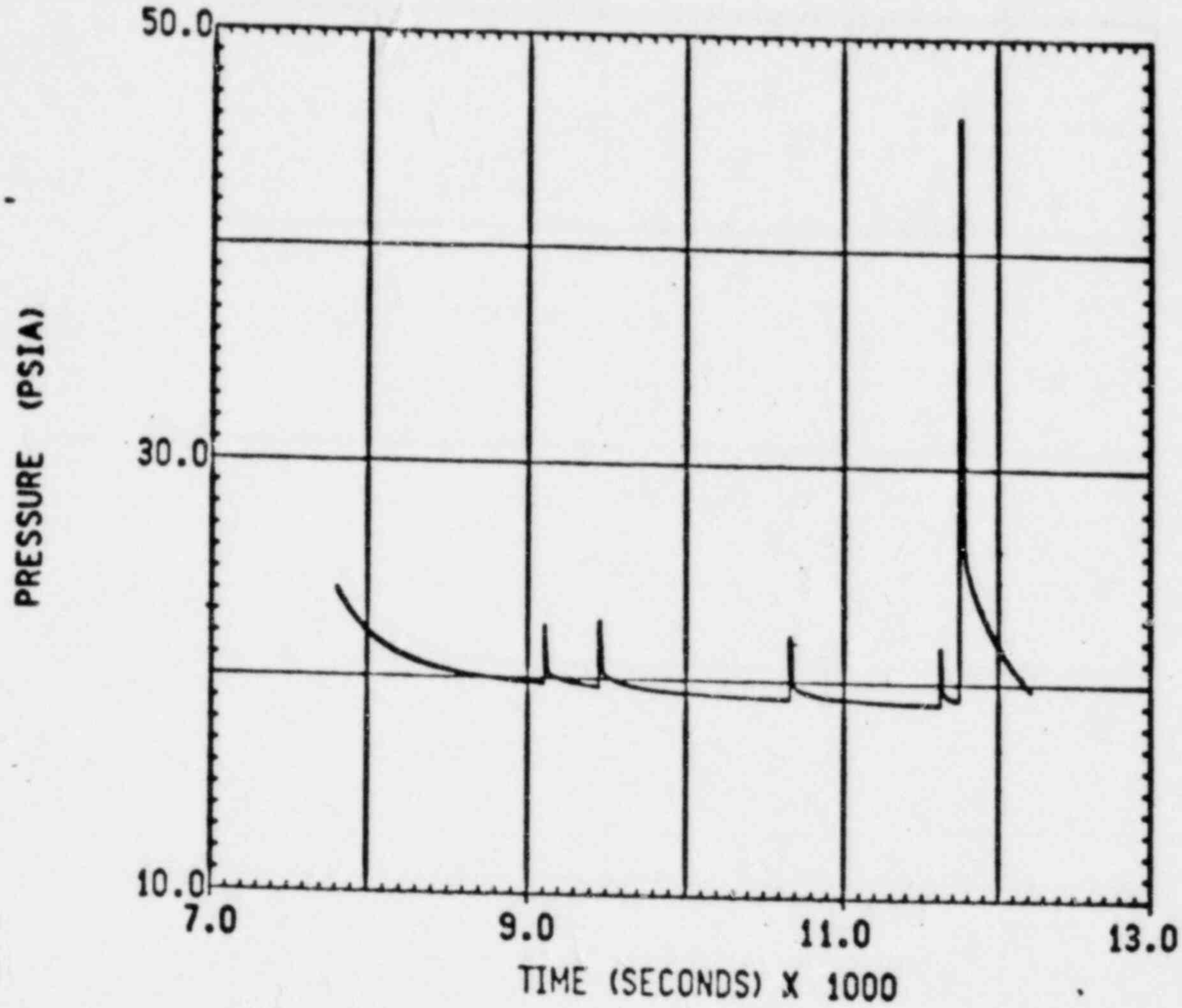
SUMMARY TABLE

Drywell Break Base Case

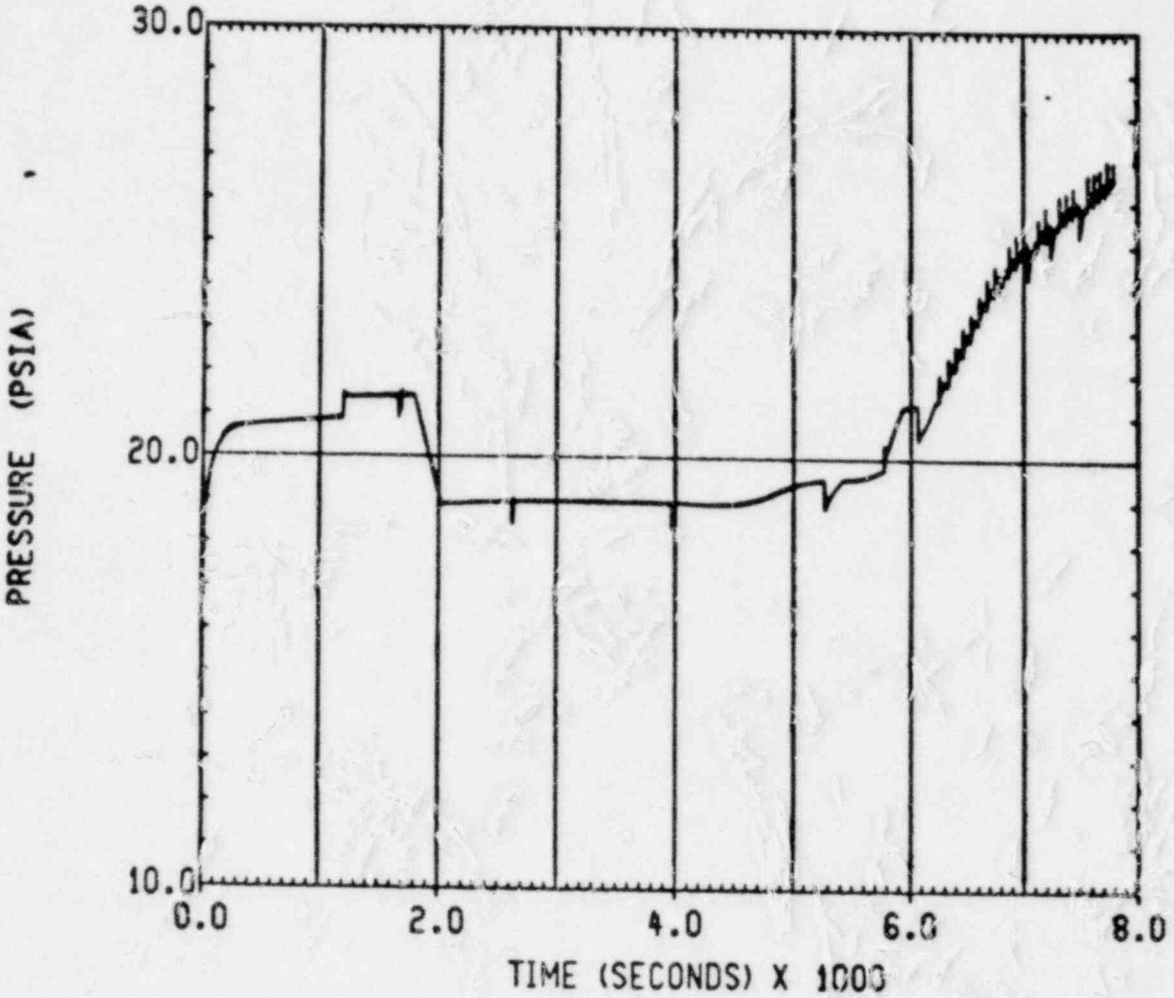
Number of Burns	
Drywell	1
Wetwell	32
Containment	1
Peak Temperature (°F)	
Drywell	707
Wetwell	2295
Containment	860
Peak Pressure (psig)	
Drywell	16.3
Wetwell	31.6
Containment	32.1
Peak Pressure Differential (psi)	
Drywell/Containment	
Forward	8.8
Reverse	17.6



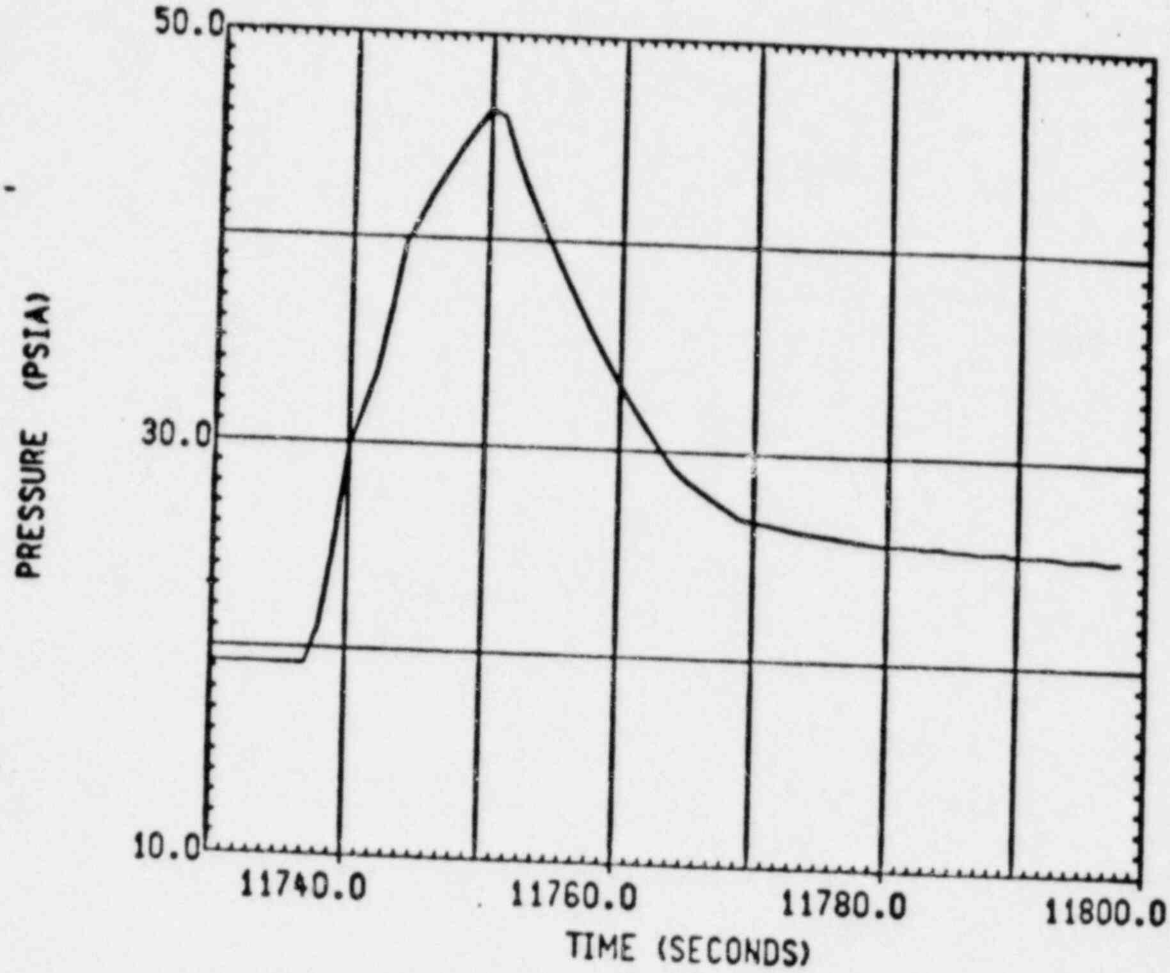
GGNS BASE CASE DRYWELL BREAK
SUPPRESSION POOL DRAW DOWN
WETWELL PRESSURE



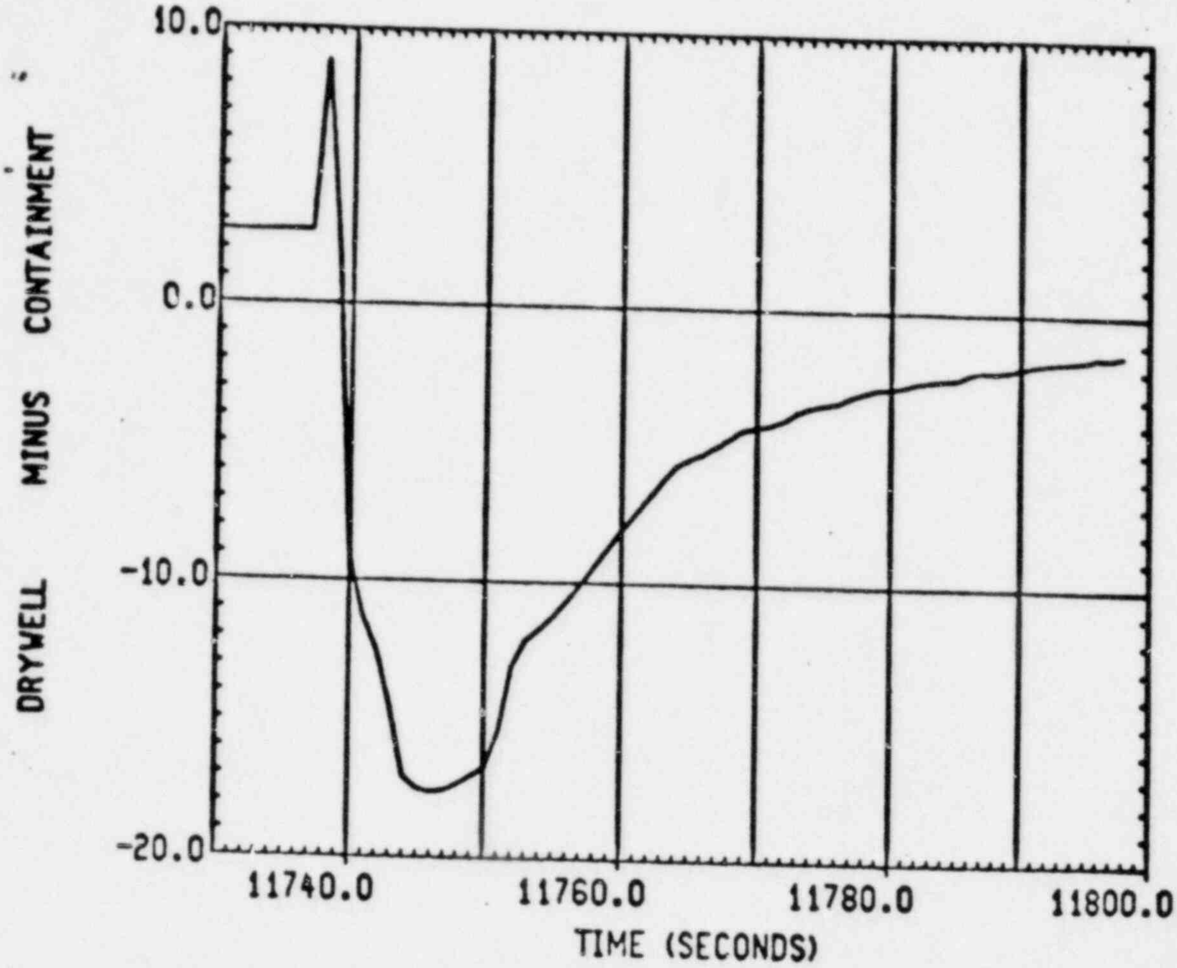
GGNS BASE CASE DRYWELL BREAK
SUPPRESSION POOL DRAW DOWN
WETWELL PRESSURE



GGNS BASE CASE DRYWELL BREAK
SUPPRESSION POOL DRAW DOWN
DRYWELL PRESSURE



GGNS BASE CASE DRYWELL BREAK
CASE DA4 - DRAWDOWN
WETWELL PRESSURE



GGNS BASE CASE DRYWELL BREAK
CASE DA4 - DRAWDOWN
DIFFERENTIAL PRESSURE

EQUIPMENT SURVIVABILITY

- DETERMINE THERMAL RESPONSE OF POTENTIALLY ESSENTIAL EQUIPMENT EXPOSED TO PREDICTED HYDROGEN BURN ENVIRONMENTS

- DETERMINE ABILITY OF THIS SAME EQUIPMENT TO WITHSTAND THE PRESSURES RESULTING FROM HYDROGEN BURNS

Equipment Survivability (cont'd)

Equipment Selection Criteria

- Maintain Containment Pressure Boundary

  - Containment Penetrations, Locks, Hatches
  - Containment Isolation Valves
  - Associated Instrumentation and Controls

- Recover and Maintain the Core, Mitigate Accident Consequences

  - HIS
  - SRVs
  - LPCS, LPCI, RHR Systems
  - Containment Sprays
  - Hydrogen Recombiners
  - Drywell Purge Compressors, Vacuum Breakers
  - Associated Instrumentation and Controls

- Monitor Course of the Event

  - Ctmt and Drywell Pressure Instrs
  - Ctmt and Drywell High-Range Radiation Monitors
  - Ctmt and Suppression Pool Temperature Instrs
  - Reactor Level and Pressure Instrs
  - Hydrogen Analyzers
  - Associated Instrumentation and Controls

Equipment Survivability (cont'd)

Temperature Effects

Evaluation Bases

- Equipment temperature response determined for:
  - Base case wetwell burn
  - Base case CTMT global burn
- More severe wetwell burn environment applied to all equipment regardless of actual location
- Modes of heat transfer - radiation, convection
- Models conservatively constructed to maximize thermal response of limiting component
- No credit taken for contact cooling from containment sprays or thermal shielding
- Equipment assumed to survive if:
  - Max. external surface T $<$ Equip. Qual. T, or
  - Max. internal T of limiting component $<$ Equip. Qual. T, or
  - Limiting component shown to maintain post accident function based on test data
- Conservatism of methodology verified by comparison against results of Fenwal ignitor burn tests (analytical results more severe)

Equipment Survivability (cont'd)

Temperature Effects

Results

- In all cases, equipment identified as potentially essential will survive the predicted hydrogen burn environment
- Considerable margin between calculated equipment temperature and temperature at which equipment operation would be threatened

EQUIPMENT SURVIVABILITY (CONT'D)

PRESSURE EFFECTS

EVALUATION BASES

- EQUIPMENT PRESSURE CAPABILITY EVALUATED FOR:
  - 24 PSIG PEAK PRESSURE
  - 5 PSID PEAK DRYWELL/CTMT DIFFERENTIAL PRESSURE (FROM SORV BASE CASE)
- EQUIPMENT ASSUMED TO SURVIVE IF:
  - QUALIFICATION PRESSURE > BURN PRESSURE
  - DURABLE, RIGID CONSTRUCTION (E.G., VALVE HOUSING) PRECLUDES PRESSURE EFFECTS
- MANY COMPONENTS HAVE PERFORMED FUNCTION WELL BEFORE ONSET OF HYDROGEN COMBUSTION

RESULTS

- NO ADVERSE EFFECTS ON POTENTIALLY ESSENTIAL EQUIPMENT EXPECTED
  - MAJORITY OF EQUIPMENT - QUAL. PRESSURE > BURN PRESSURE
  - PURGE COMPRESSORS, VACUUM BREAKERS EVALUATED UP TO 30 PSID

Containment Structural Capability

Containment Design Pressure: 15 psig

Drywell Internal Design Pressure: 30 psid

Containment ultimate capacity calculations:

- Using design specified strengths - 56 psig
- Using actual strengths
  - 70 psig upper bound
  - 67 psig mean
  - 62 psig lower bound

Drywell Ultimate Capacity Calculation:

- 67 psid (internal)
- Greater than 67 psid (external)

Ultimate Pressure Capacity for Containment Hatches and Airlocks

- Containment equipment hatch - 206.5 psig
- Lower containment airlock - 77.6 psig
- Upper containment airlock - 60 psig
- Drywell personnel airlock - 79.2 psig
- Drywell Equipment Hatch - 96 psig

Penetration Closure Plates are Calculated at 60 psig

Piping has been Evaluated at Retaining 75 psig (external)

Drywell Head Buckling Capacity - 89 psig (external)

Brookhaven National Lab Results for Containment Ultimate Capacity - 52 psig

LOCAL DETONATIONS

BACKGROUND

- As stated by Sandia and COMBEX, detonation of a substantial volume of hydrogen-air mixture in the Grand Gulf containment is unlikely since:
  - Accumulation of a detonable mixture is prevented by the Hydrogen Ignitor System (HIS)
  - Geometrical requirements for transition from deflagration to detonation do not exist
- However, at the request of the NRC, the effects of potential local detonations on structural integrity were evaluated for Grand Gulf structural components
- Structural Components Evaluated
  - Drywell wall
  - Containment shell
  - Lower containment personnel airlock
  - Drywell personnel airlock
  - Drywell equipment hatch

LOCAL DETONATIONS (CONT'D)

EVALUATION BASES

- GRAND GULF ASSUMPTIONS
  - GAS CLOUD LOCATED DIRECTLY ADJACENT TO STRUCTURAL COMPONENT
  - UNCONFINED DETONATION (TNT EQUIVALENT)
  - HYDROGEN CONCENTRATIONS VARIED FROM 25 TO 50 VOLUME PERCENT
  - VOLUME APPROXIMATELY 525 CUBIC FEET
- SANDIA
  - HYDROGEN CONCENTRATION-28 VOLUME PERCENT
  - PARTIALLY CONFINED DETONATION (CSQ CODE)
  - VOLUME APPROXIMATELY 28,000 CUBIC FEET

LOCAL DETONATIONS (CONT'D)

RESULTS

PEAK REFLECTED IMPULSE FOR
DETONATION OF A
25-28 PERCENT H_2 + AIR MIXTURE

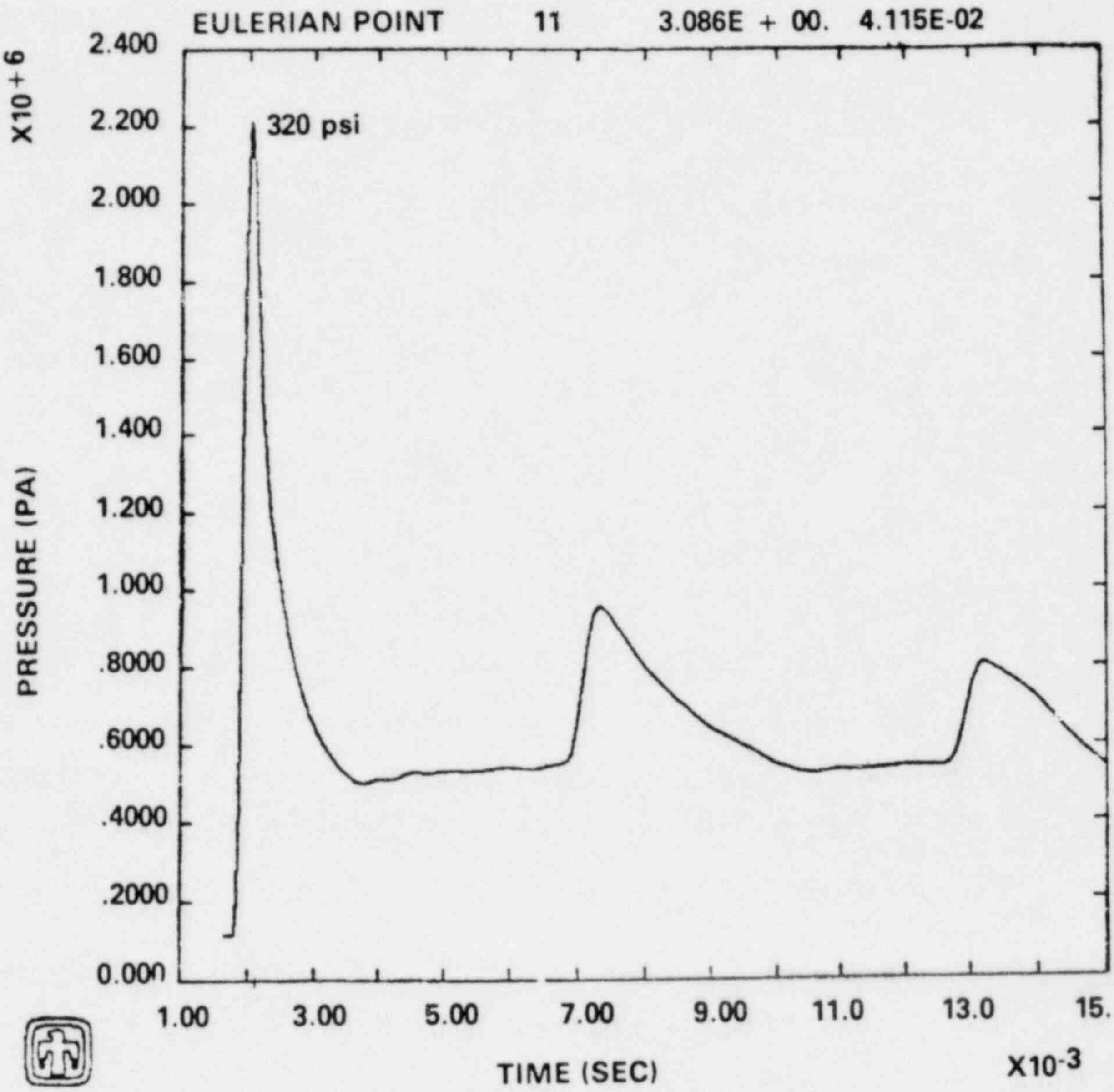
METHODOLOGY	IMPULSE (PSI-SEC)	VOLUME (CU. FT.)
MARK	.043	525
COMBEX	.115	525
GGNS	.176	525
SANDIA	.700	28,000

LOCAL DETONATIONS (CONT'D)

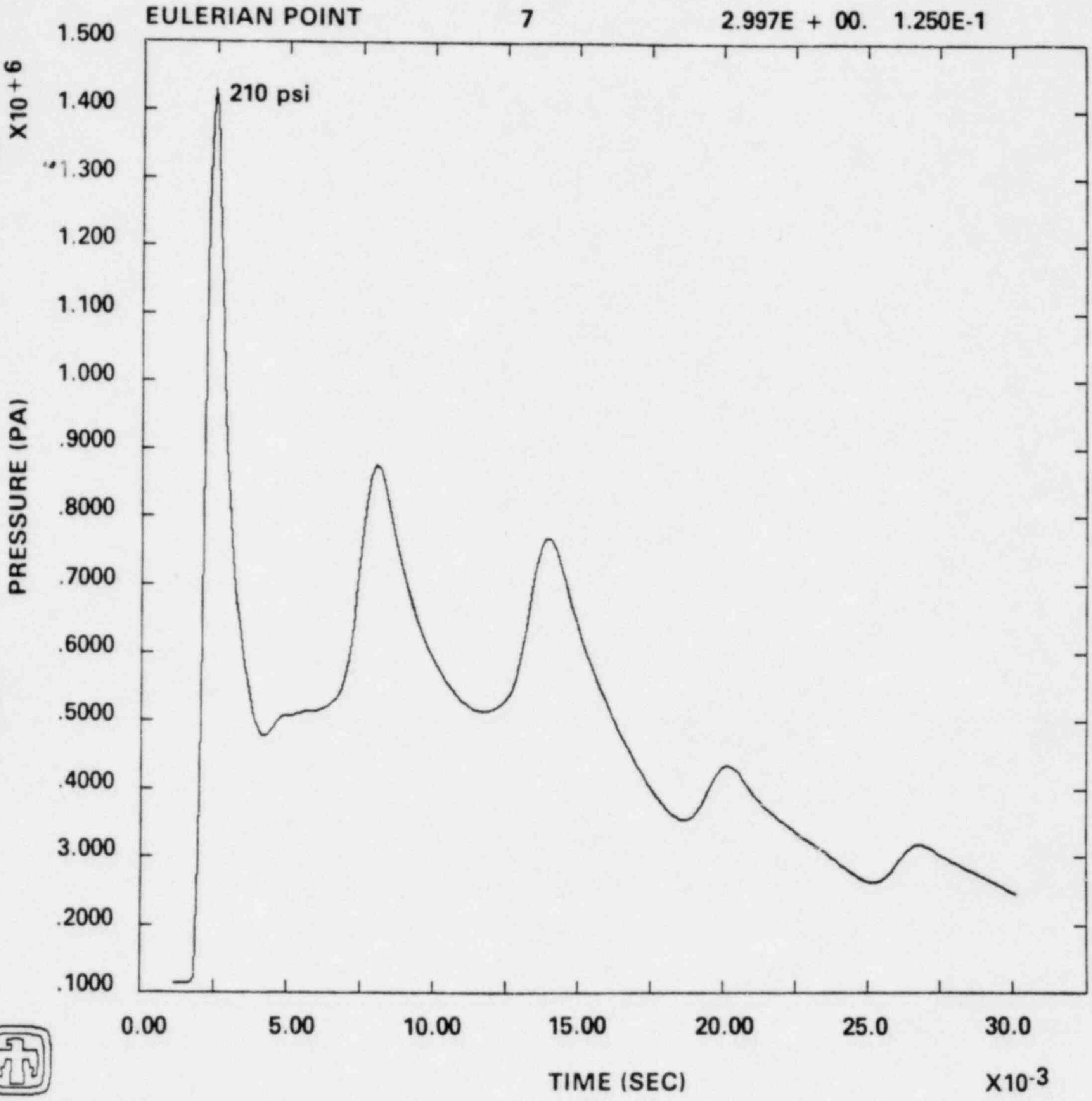
CONCLUSIONS

- CONTAINMENT STRUCTURAL COMPONENTS - ADEQUATE CAPACITY

- DRYWELL STRUCTURAL COMPONENTS - ADEQUATE CAPACITY



GRAND GULF 0.28 H2-DRY AIR



GRAND GULF 0.28 H2-DRY AIR (COARSE)

HYDROGEN TESTING PROGRAM

- Hydrogen-Rich Steam/Air
- Burn Testing above the Suppression Pool
- Hydrogen Mixing

HYDROGEN TESTING PROGRAM (CONT'D)

HYDROGEN-RICH TESTING

- AECL WHITESHELL FACILITY
- SMALL SCALE PHENOMENA TESTS
- INVESTIGATE FLAMMABILITY LIMITS IN POTENTIAL DRYWELL ENVIRONMENTS

HYDROGEN TESTING PROGRAM (CONT'D)

BURN TESTS ABOVE THE SUPPRESSION POOL

- New Facility
- Large Scale
- Investigate Burning above the Suppression Pool

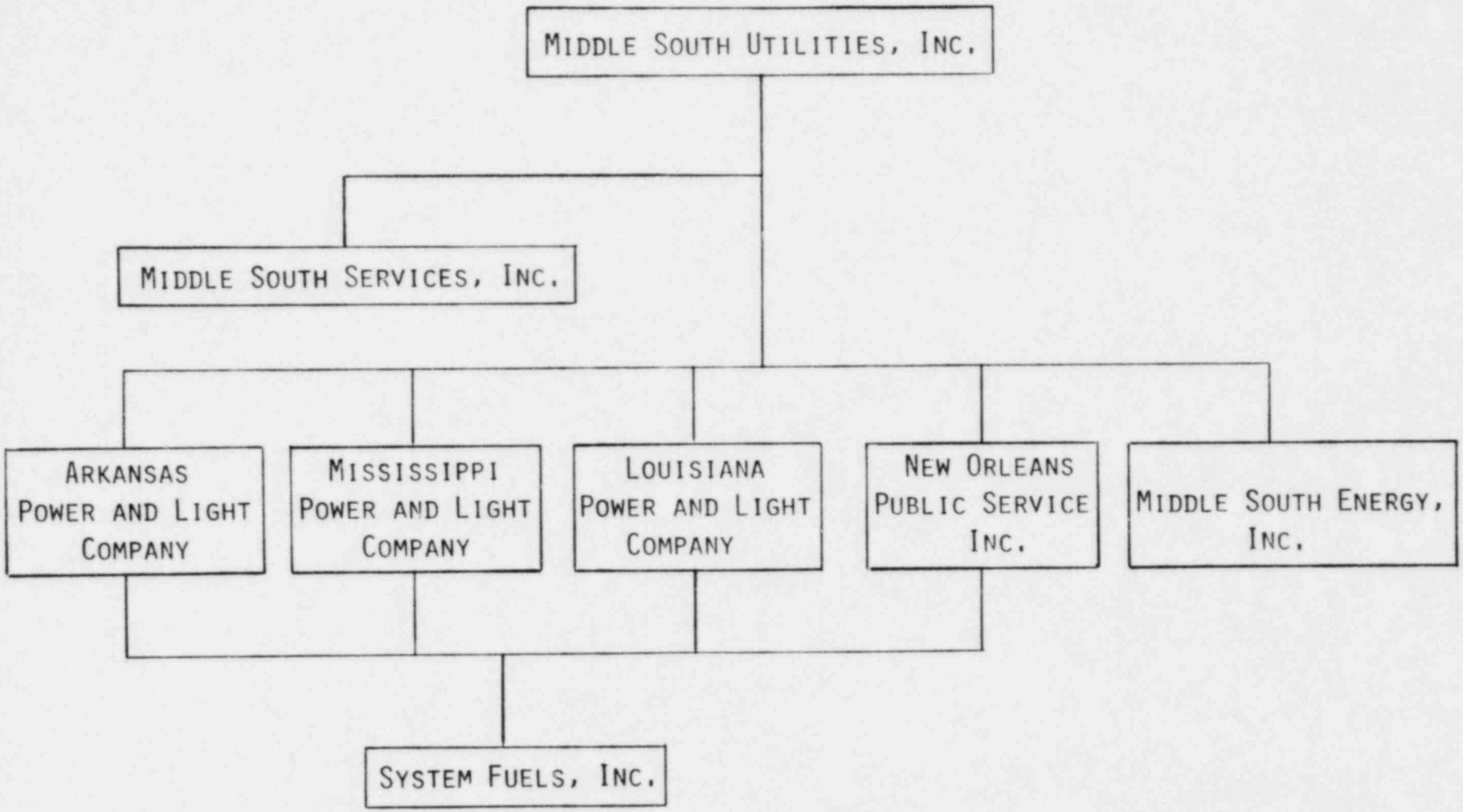
HYDROGEN TESTING PROGRAM (CONT'D)

HYDROGEN MIXING

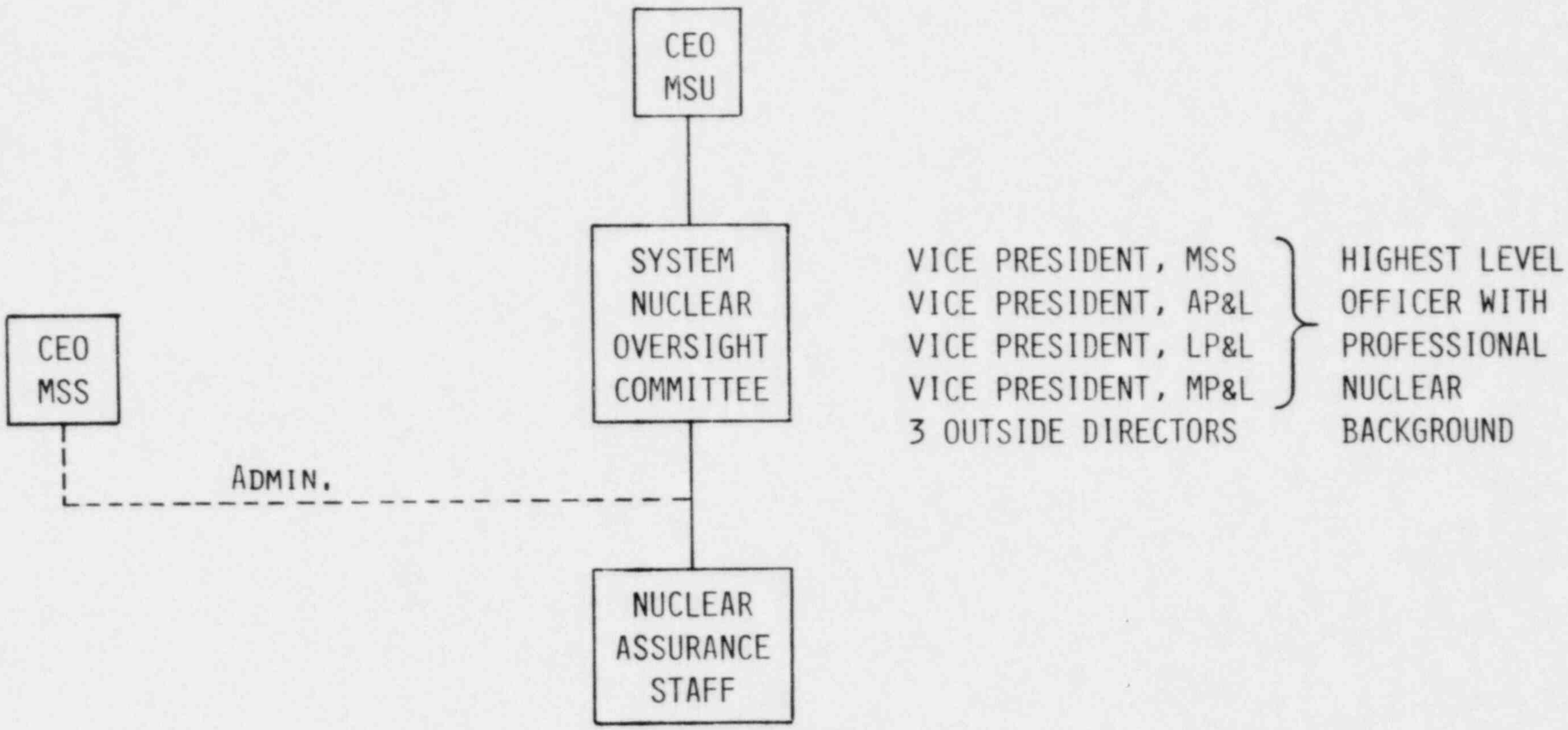
INVESTIGATE:

- UTILIZATION OF THE HEDL FACILITY
- EVALUATE MIXING IN THE SCALE TEST AND COMPARE WITH AVAILABLE DATA

MIDDLE SOUTH UTILITIES SYSTEM



NUCLEAR OVERSIGHT ORGANIZATION



NUCLEAR OVERSIGHT COMMITTEE FUNCTIONS

- OVERVIEW AND GUIDANCE ON SYSTEM-WIDE BASIS
- OVERSIGHT AND APPRAISAL OF NUCLEAR ACTIVITIES OF MSU SYSTEM
- DEVELOP STANDARDS OF PERFORMANCE
- IMPROVE SYSTEM-WIDE RESPONSE TO EMERGENCY CONDITIONS
- IMPROVED, COST EFFECTIVE SUPPORT SERVICES WITH CONTINUITY OF EXPERIENCE

NUCLEAR ASSURANCE ORGANIZATION FUNCTIONS

SUPPORTS NUCLEAR OVERSIGHT COMMITTEE

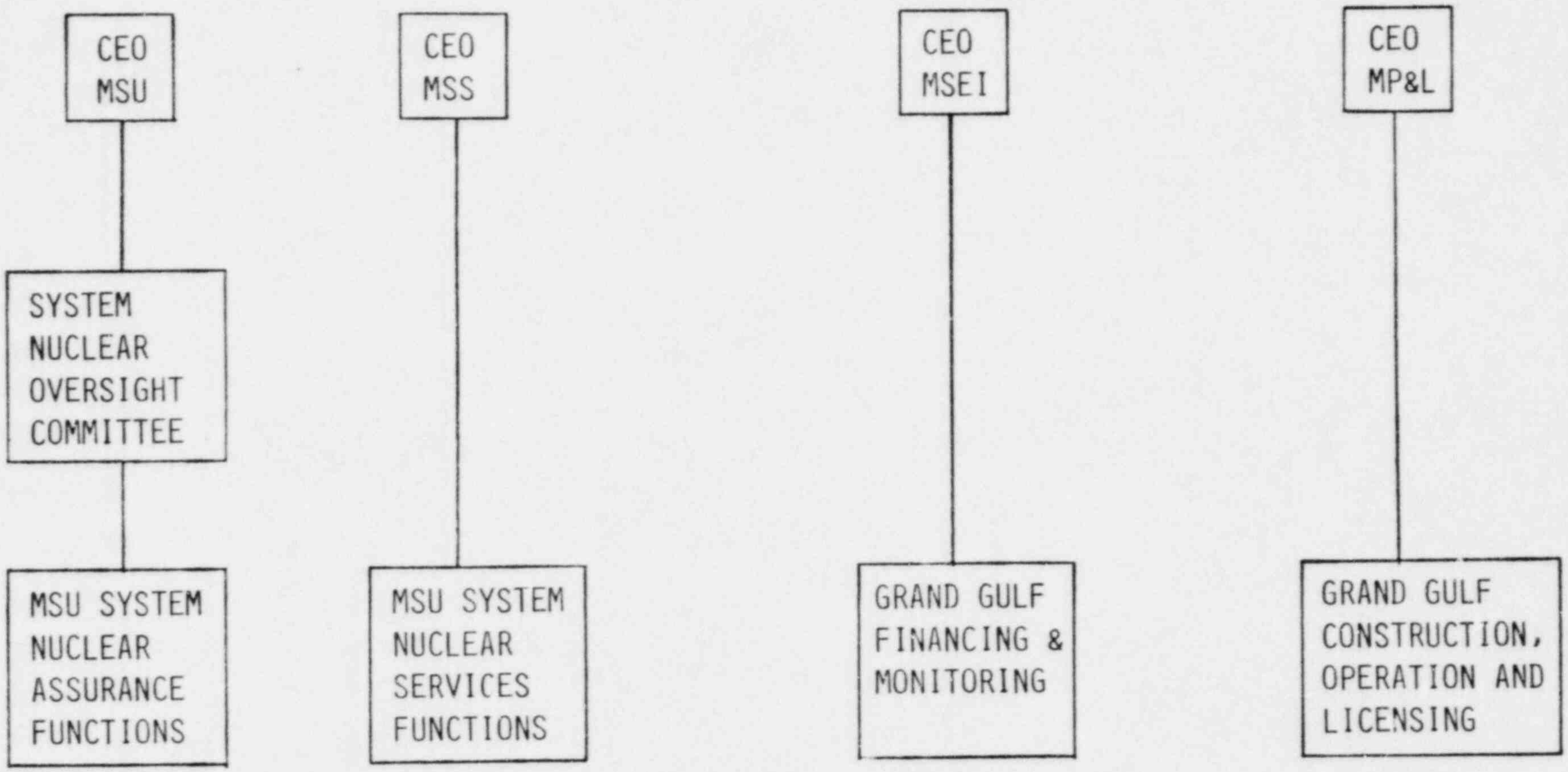
PROVIDES

- STAFF SUPPORT
- REPRESENTATION ON SAFETY REVIEW COMMITTEES
- NUCLEAR SAFETY OVERSIGHT
- ASSESSMENT OF RISK CONTROL ACTIVITIES
- SYSTEM CONTACT WITH INPO

NUCLEAR SERVICES ORGANIZATION FUNCTIONS

- NUCLEAR FUEL CYCLE SUPPORT
- FUEL AND REACTOR ENGINEERING/ANALYSIS SUPPORT
- QUALITY ASSURANCE
- SPECIALIZED TECHNICAL SERVICES
- REGULATORY RESPONSE ON GENERIC/SYSTEM ISSUES

GRAND GULF ORGANIZATION RELATIONSHIPS



MISSISSIPPI POWER & LIGHT COMPANY

CORPORATE SAFETY REVIEW COMMITTEE

- FIRST MEETING IN JUNE, 1981
- MONTHLY MEETING FREQUENCY
- EXPANDED FROM 7 TO 11 MEMBERS
- REPORTS TO SENIOR VICE PRESIDENT - NUCLEAR
- OPERATION PER NUCLEAR PRODUCTION DEPARTMENT PROCEDURE 9.2
- FORMAL TRAINING SESSIONS ON DUTIES AND RESPONSIBILITIES
- CHARTER, DUTIES AND AREAS OF REVIEW AS PER TECH SPEC SECTION 6.5.2
- PLANNING TO EXPAND CHARTER AND AREAS OF REVIEW

SRC COMPOSITION

- ASSISTANT VICE PRESIDENT - NUCLEAR PRODUCTION (CHAIRMAN)
- MANAGER OF NUCLEAR SERVICES (ALTERNATE CHAIRMAN)
- MANAGER OF SAFETY AND LICENSING (SECRETARY)
- MANAGER OF QUALITY ASSURANCE
- MANAGER OF NUCLEAR PLANT ENGINEERING
- NUCLEAR PLANT MANAGER
- CORPORATE HEALTH PHYSICIST
- PRINCIPAL ENGINEER - OPERATIONS ANALYSIS
- MANAGER OF SYSTEM NUCLEAR OPERATIONS, MIDDLE SOUTH SERVICES
- CONSULTANT (DR. J. M. HENDRIE)
- CONSULTANT (DR. D. W. JONES)
- CONSULTANT (J. F. GROVES)

  - ONLY 2 OF 12 ARE LINE MANAGEMENT FOR PLANT OPERATIONS
  - 25% CONSULTANTS / 33% NOT MP&L EMPLOYEES
  - INDIVIDUALLY AND COLLECTIVELY MEET EXPERIENCE AND EXPERTISE REQUIREMENTS OF TECH SPEC SECTION 6.5.2.
  - AVERAGE 19 YEARS PROFESSIONAL AND 17 YEARS NUCLEAR EXPERIENCE

SRC SPECIAL SUBCOMMITTEE ON

REVIEW OF PLANT OPERATIONAL READINESS

- ORGANIZED TO REVIEW READINESS FOR FUEL LOAD - FEBRUARY, 1982

- COMPOSITION - NO MP&L EMPLOYEES

- REPORT SUBMITTED JUNE 7, 1982

- SUBCOMMITTEE RE-CONVENED JUNE 12, 1982
  - EXPANDED SCOPE
  - ADDED INDUSTRIAL PSYCHOLOGIST TO MEMBERSHIP

- REPORT SUBMITTED JUNE 13, 1982

- LETTER FROM D. C. LUTKEN DI[illegible]G FURTHER REVIEWS - JUNE 13, 1982

- OPERATING LICENSE CONDITION SPECIFYING FURTHER REVIEWS AND SCOPE OF EACH
  - PRIOR TO 5% OF FULL POWER
  - PRIOR TO 50% OF FULL POWER
  - WITHIN 30 DAYS OF COMPLETION OF WARRANTY RUN

- WILL PERFORM PERIODIC GENERAL ASSESSMENTS OF UNIT OPERATIONS

PLANT SAFETY REVIEW COMMITTEE (PSRC)

- FUNCTION - ADVISE PLANT MANAGER ON ALL MATTERS RELATED TO NUCLEAR SAFETY

- COMPOSITION - 7 MEMBERS

  - CHAIRMAN - ASSISTANT PLANT MANAGER
  - VICE CHAIRMAN - NUCLEAR SUPPORT MANAGER
  - MEMBER - OPERATIONS SUPERINTENDENT
  - MEMBER - TECHNICAL SUPPORT SUPERINTENDENT
  - MEMBER - QUALITY SUPERINTENDENT
  - MEMBER - CHEMISTRY AND RADIATION PROTECTION SUPERINTENDENT
  - MEMBER - MAINTENANCE SUPERINTENDENT

- MEETING FREQUENCY - MONTHLY OR AS CONVENED

- RESPONSIBILITIES - REVIEW OF:

  - STATION ADMINISTRATIVE PROCEDURES & CHANGES
  - SAFETY EVALUATIONS
  - PROPOSED CHANGES WHICH MAY INVOLVE AN UNREVIEWED SAFETY QUESTION
  - TESTS WHICH MAY INVOLVE UNREVIEWED SAFETY QUESTIONS
  - PROPOSED CHANGES TO TECH SPECS OR OPERATING LICENSE
  - REPORTS OF VIOLATIONS OF CODES OR PROCEDURES HAVING NUCLEAR SAFETY SIGNIFICANCE
  - REPORTS OF DEFICIENT SYSTEMS CONTAINING RADIOACTIVE MATERIAL
  - REPORTS OF OPERATING ABNORMALITIES OR DEVIATIONS
  - EVENTS REQUIRING 24 HOUR COMMISSION NOTIFICATION
  - UNANTICIPATED DESIGN OR OPERATIONAL DEFICIENCIES OF SAFETY-RELATED STRCUTURES, SYSTEMS OR COMPONENTS
  - PLANT SECURITY & CHANGES
  - EMERGENCY PLAN & CHANGES
  - POTENTIAL NUCLEAR SAFETY HAZARDS
  - INVESTIGATIONS OR ANALYSES REQUESTED BY CHAIRMAN OF NSRC
  - UNEXPECTED OFFSITE RELEASES
  - CHANGES TO PROCESS CONTROL PROGRAM DOSE MANUAL AND RADWASTE SYSTEMS

- AUTHORITY - REPORT TO PLANT MANAGER

TECHNICAL REVIEW AND CONTROL

. ACTIVITIES AFFECTING NUCLEAR SAFETY SHALL BE CONDUCTED AS FOLLOWS:

- PROCEDURES WHICH AFFECT PLANT NUCLEAR SAFETY SHALL BE PREPARED, REVIEWED AND APPROVED
  - INDEPENDENT REVIEW
  - PROCEDURES MUST RECEIVE WRITTEN APPROVAL BY PLANT MANAGER
- PROPOSED CHANGES TO PLANT NUCLEAR SAFETY-RELATED SYSTEMS/STRUCTURES/COMPONENTS SHALL RECEIVE REVIEW DESIGNATED BY PLANT MANAGER
  - INDEPENDENT REVIEW
  - IMPLEMENTATION MUST BE APPROVED BY PLANT MANAGER
- PROPOSED TESTS AND EXPERIMENTS WHICH AFFECT PLANT NUCLEAR SAFETY SHALL RECEIVE INDEPENDENT REVIEW
- REPORTABLE OCCURRENCES AND TECH SPEC VIOLATIONS SHALL BE INVESTIGATED WITH RECOMMENDATION TO PLANT MANAGER
- INDIVIDUALS PERFORMING REVIEW SHALL MEET RELATED ANSI STANDARD (18.1-1971)
- REVIEW SHALL DETERMINE WHETHER AN UNRESOLVED SAFETY ISSUE IS INVOLVED
- RECORDS OF ABOVE ACTIVITIES SHALL BE PROVIDED TO STATION MANAGER, PSRC FOR REQUIRED REVIEWS

PERSPECTIVE

ON

HUMPHREY ISSUES

SUMMARY OF EVENTS

1. JOHN HUMPHREY LETTER DATED MAY 8, 1982 RECEIVED BY MP&L ON MAY 12, 1982.

2. INITIAL MEETING WITH GE, BECHTEL, MP&L AND JOHN HUMPHREY ON MAY 17, 1982.

3. MEETING WITH NRC, MP&L AND JOHN HUMPHREY TO DISCUSS THESE ISSUES AND MP&L'S RESPONSE ON MAY 27, 1982.

4. MP&L RESPONSES FORMALLY SUBMITTED ON MAY 28, 1982.

5. MP&L PROVIDED JUSTIFICATION BY LETTER JUNE 8, 1982 FOR FUEL LOADING PENDING FINAL RESOLUTION OF THESE ISSUES.

6. MP&L FORMALLY RECEIVED REQUESTS FOR ADDITIONAL INFORMATION FROM THE NRC TO RESOLVE THE ISSUES ON JULY 8, 1982.

7. MP&L RECEIVED INFORMALLY A COPY OF MR. HUMPHREY'S LETTER TO AL SCHWENCER DATED JUNE 17, 1982 ON JUNE 27, 1982.

8. MP&L MET WITH NRC ON JULY 14, 1982 TO REVIEW ACTIONS AND SCHEDULES FOR PROVIDING FINAL CLOSURE OF ISSUES.

9. ACTION PLANS FOR RESOLVING ISSUES AND RESPONDING TO NRC INFORMATION REQUEST SUBMITTED TO NRC ON JULY 15, 1982.

10. MEETING HELD WITH MARK III OWNERS, GENERAL ELECTRIC, PLANT ARCHITECT ENGINEERS AND JOHN HUMPHREY ON JULY 22, 1982.

11. FORMED A MARK III OWNERS' GROUP FOR PERFORMING GENERIC WORK ON JULY 22, 1982.

12. ACRS FLUID DYNAMICS SUBCOMMITTEE MEETING JULY 29 & 30.

MP&L APPROACH TO RESOLUTION OF THESE CONCERNS

1. INITIAL EVALUATION DETERMINED THAT THE CONCERNS DO NOT IMPACT PLANT SAFETY AND ARE DETAILED DESIGN ISSUES

   - INITIAL REVIEW CONCLUDED THAT ALL TECHNICAL QUESTIONS ADEQUATELY ADDRESSED BY GGNS DESIGN
   - ISSUES RAISED DUE TO SELECTIVE OR UNREALISTIC COMBINATIONS OF ANALYTICAL ASSUMPTIONS, BOUNDARY CONDITIONS, TEST DATA AND SYSTEM PERFORMANCE
   - ISSUES DO NOT CONSIDER THE OVERALL LEVEL OF CONSERVATISM AND MARGIN INHERENT IN THE CONTAINMENT DESIGN
   - ANY EFFECTS WITHIN DESIGN MARGINS

2. TO QUANTIFY THE EFFECTS, A COMPREHENSIVE PROGRAM UNDERTAKEN

   - CONDUCTING PLANT SPECIFIC ANALYSES
   - PROCEDURE AND TECHNICAL SPECIFICATION REVIEWS

3. SCHEDULE FOR COMPLETING PROGRAM TO ADDRESS ISSUES

   - ACTION PLAN SUBMITTED JULY 15, 1982
   - INITIAL REPORT WITH JUSTIFICATION FOR FULL POWER OPERATION PENDING FINAL RESOLUTION ON AUGUST 19, 1982
     - DETAILED DESCRIPTION OF ANALYSIS, ASSUMPTIONS, EXPECTED RESULTS IF NOT COMPLETED PRIOR TO FULL POWER LICENSE
     - DETAILED DESCRIPTION OF ANALYSIS AND RESULTS IF COMPLETE
   - SUPPLEMENTARY INFORMATION SUBMITTED ON OCTOBER 1, 1982.
   - FINAL PROGRAM REPORT ON NOVEMBER 1, 1982

4. ACTIVELY INVOLVED IN GENERIC EFFORT

CONTAINMENT ISSUES OWNERS GROUP

1. OWNERS GROUP INCLUDES:

   - MISSISSIPPI POWER & LIGHT COMPANY
   - CLEVELAND ELECTRIC ILLUMINATING COMPANY
   - ILLINOIS POWER COMPANY
   - GULF STATES UTILITIES
   - GENERAL ELECTRIC COMPANY

2. OWNERS GROUP EFFORTS INCLUDE:

   - REVIEW OF GGNS ACTION PLAN TO DEVELOP GENERIC ACTION PLAN
   - IDENTIFY AREAS REQUIRING PLANT UNIQUE ANALYSIS AND AGREE ON ACCEPTABLE PLAN FOR RESOLUTION
   - ESTABLISH REVIEW PANEL TO INDEPENDENTLY REVIEW ACTION PLANS AND RESULTS OF ANALYSIS

3. REVIEW PANEL COMPOSED OF GE/AE/UTILITY "EXPERTS" NOT ACTIVELY INVOLVED IN RESOLUTION OF THE ISSUES AND CHARGED WITH:

   - ASSURING ISSUES HAVE BEEN PROPERLY DEFINED.
   - REVIEWING GENERIC ACTION PLANS.
   - REVIEWING PLANT UNIQUE ACTION PLANS.
   - REVIEWING COMPLETED WORK AND VERIFYING ISSUES ARE CLOSED.

4. SCHEDULED COMPLETION IN EARLY 1983

ACTION PLAN OUTLINE

1. LOCAL ENCROACHMENTS

2. PERTURBATIONS IN LOAD DEFINITION CAUSED BY ANNULAR VENTS

3. UNACCOUNTED FOR RELIEF VALVE EFFECTS

4. SUPPRESSION POOL TEMPERATURE STRATIFICATION

5. DRYWELL TO CONTAINMENT BYPASS LEAKAGE EFFECTS

6. RHR PERMISSIVE ON CONTAINMENT SPRAY

7. CONTAINMENT PRESSURE RESPONSE

8. CONTAINMENT AIRMASS EFFECTS

9. DRYWELL AIRMASS EFFECTS

10. WEIR WALL OVERFLOW

11. OPERATIONAL CONTROL OF DRYWELL TO CONTAINMENT DIFFERENTIAL PRESSURE

12. CONTAINMENT SPRAY BACKFLOW

13. EFFECT OF SUPPRESSION POOL LEVEL ON TEMPERATURE MEASUREMENT

14. EFFECTS OF CHUGGING FROM LOCAL ENCROACHMENTS AND ADDITIONAL SUBMERGENCE

15. LATERAL LOADS DURING D/W NEGATIVE PRESSURE TRANSIENT

SUMMARY

- CONTAINMENT CONCERNS ARE DESIGN REFINEMENTS
- ACTION PLAN HAS BEEN DEVELOPED
- SUPPORTING ANALYSES BEING DEVELOPED
- SUBMITTALS TO NRC
  - AUGUST 19 - INFORMATION TO JUSTIFY FULL POWER LICENSE
  - OCTOBER 1 - ANALYSES
  - NOVEMBER 1 - FINAL DETAILS
- OWNERS GROUP ISSUE RESOLUTION BY EARLY 1983

GRAND GULF SQRT PROGRAM

- SQRT AUDIT ON JULY 28-30, 1981 AND TRIP REPORT ISSUED OCTOBER 22, 1981

- APRIL 5, 1982, MP&L NOTIFIED EQB THAT 62 PIECES OF NSSS AND 14 PIECES OF BOP EQUIPMENT NOT QUALIFIED TO THE SQRT CRITERIA
  - TOTAL EQUIPMENT QUALIFIED TO SQRT - 98.3%

- SINCE APRIL 5, 1982, MP&L HAS PROVIDED EQB WITH 6 SUBMITTALS JUSTIFYING INTERIM OPERATION OR DOCUMENTATION SHOWING QUALIFICATION

- SINCE APRIL 5, 1982, MP&L HAS MET WITH EQB TWICE

- AS OF AUGUST 1, 1982, 22 PIECES OF NSSS AND 4 PIECES OF BOP EQUIPMENT ARE NOT QUALIFIED TO THE SQRT CRITERIA

- NSSS EQUIPMENT NOT QUALIFIED TO SQRT CRITERIA AS OF AUGUST 1, 1982

  - FUEL HANDING AND AUXILIARY PLATFORMS
  - IN VESSEL RACK
  - DEFECTIVE FUEL STORAGE CONTAINER
  - BOP/PGCC PANELS (40 YEAR AGING)

- BOP EQUIPMENT NOT QUALIFIED TO SQRT CRITERIA AS OF AUGUST 1, 1982

  - SAFETY RELIEF VALVES (OPERABILITY)

OTHER OUTSTANDING ISSUES

- NRC CONTRACTOR PERFORMED AN INDEPENDENT ANALYSIS ON THE HPCS SERVICE WATER PUMP WHICH SHOWED IT TO BE OVERSTRESSED

  - MP&L MAINTAINED IT WAS QUALIFIED AND MET SQRT CRITERIA
  - EVALUATION OF EG&G ANALYSIS INDICATED ERRORS WHICH WHEN CORRECTED SHOWED STRESSES WITHIN ALLOWABLES

- VARIABLE FROTH IMPACT LOAD REQUIRED REVISED RESPONSE SPECTRA AND RE-EVALUATION OF EQUIPMENT QUALIFICATION

  - BOP EVALUATION SHOWED NEED FOR REQUALIFICATION OF PAM THERMOCOUPLES; REQUALIFICATION IS COMPLETE
  - HCU EVALUATION SHOWED QUALIFIED TO REVISED RRS WITH 25% ADDED FOR CONSERVATISM

ORIGINAL DESIGN

- GRAND GULF DESIGNED TO GESSAR II, APPENDIX 3B LOAD DEFINITIONS

  - FROTH IMPACT LOAD OF 15 PSI

  - FROTH IMPACT DURATION OF 100 MSEC

  - FROTH DRAG LOAD OF 11 PSI

- RESPONSE SPECTRA FOR EQUIPMENT QUALIFICATION WAS BASED ON THE GESSAR LOAD DEFINITION

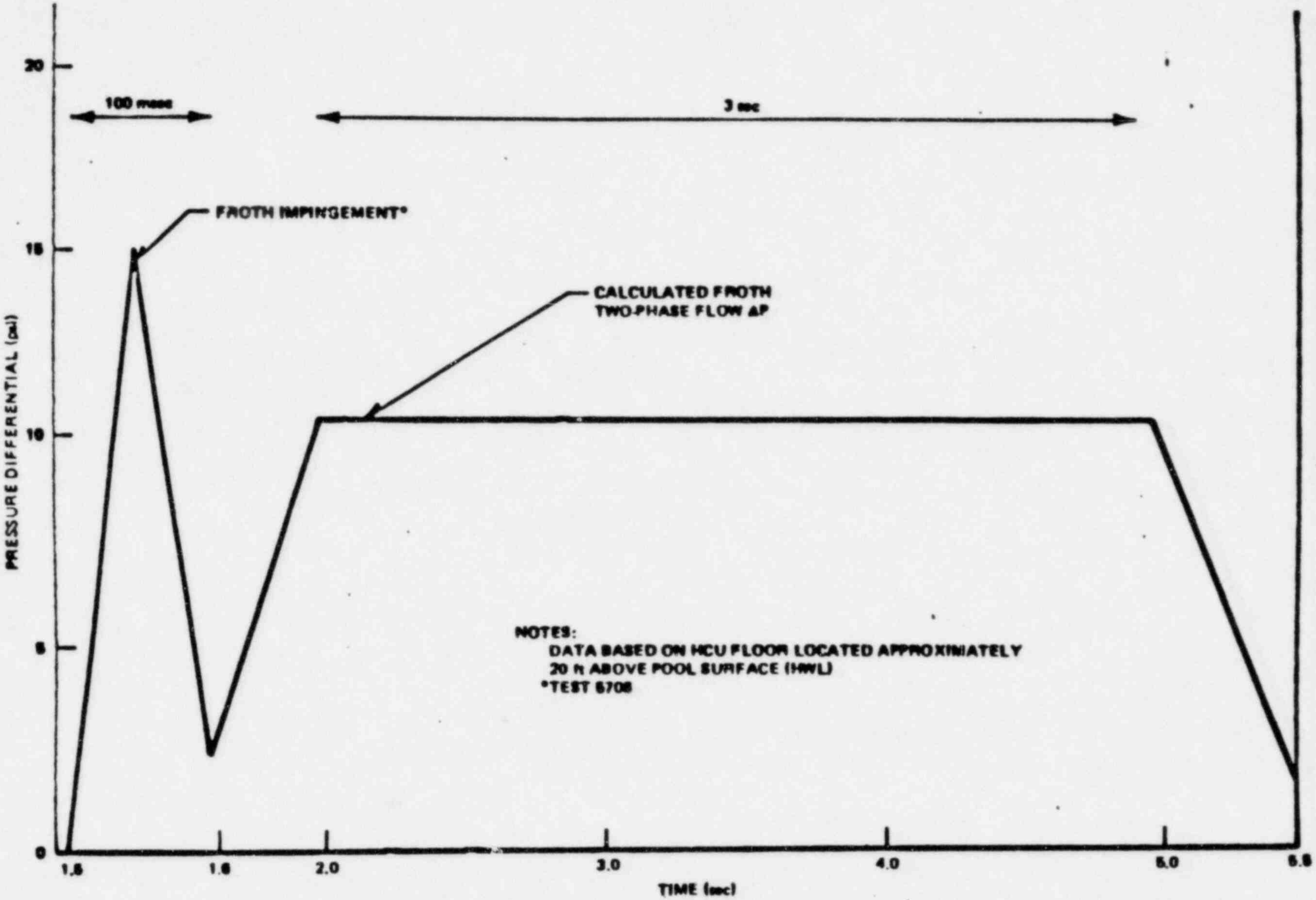


Figure 3B-73. Loads at HCU Floor Elevation Due to Pool-Swell Froth Impact and Two-Phase Flow

VARIABLE POOL SWELL - STRUCTURAL

IN A MEETING ON DECEMBER 16, 1982, MP&L WAS REQUESTED TO EVALUATE THE EFFECT ON THE HCU FLOOR'S STRUCTURAL CAPABILITY OF THE NRC BEST-ESTIMATE METHODOLOGY FOR FROTH IMPACT LOADS.

- FROTH IMPACT LOADS ARE VARIABLE.
  - DECREASE LINEARLY WITH INCREASING HEIGHT OF THE IMPACTED SURFACE ABOVE THE SUPPRESSION POOL.
  - DEVELOPED BY NRC CONSULTANT G. MAISE.
  - 3 PSI CONSERVATISM ADDED TO NRC REQUEST.
- FROTH IMPACT DURATION VARIES BETWEEN 20 AND 220 MSEC.
- FROTH IMPACT PRESSURE VARIES FROM 14.5 PSI TO 19 PSI.
- FROTH DRAG LOAD IS 11 PSI FOR SOLID FLOOR AREAS (CONCRETE) AND 5.5 PSI FOR GRATING.
- EVALUATION INDICATES THAT THE GRAND GULF HCU FLOOR IS CAPABLE OF WITHSTANDING THE REVISED FROTH IMPACT LOADS.

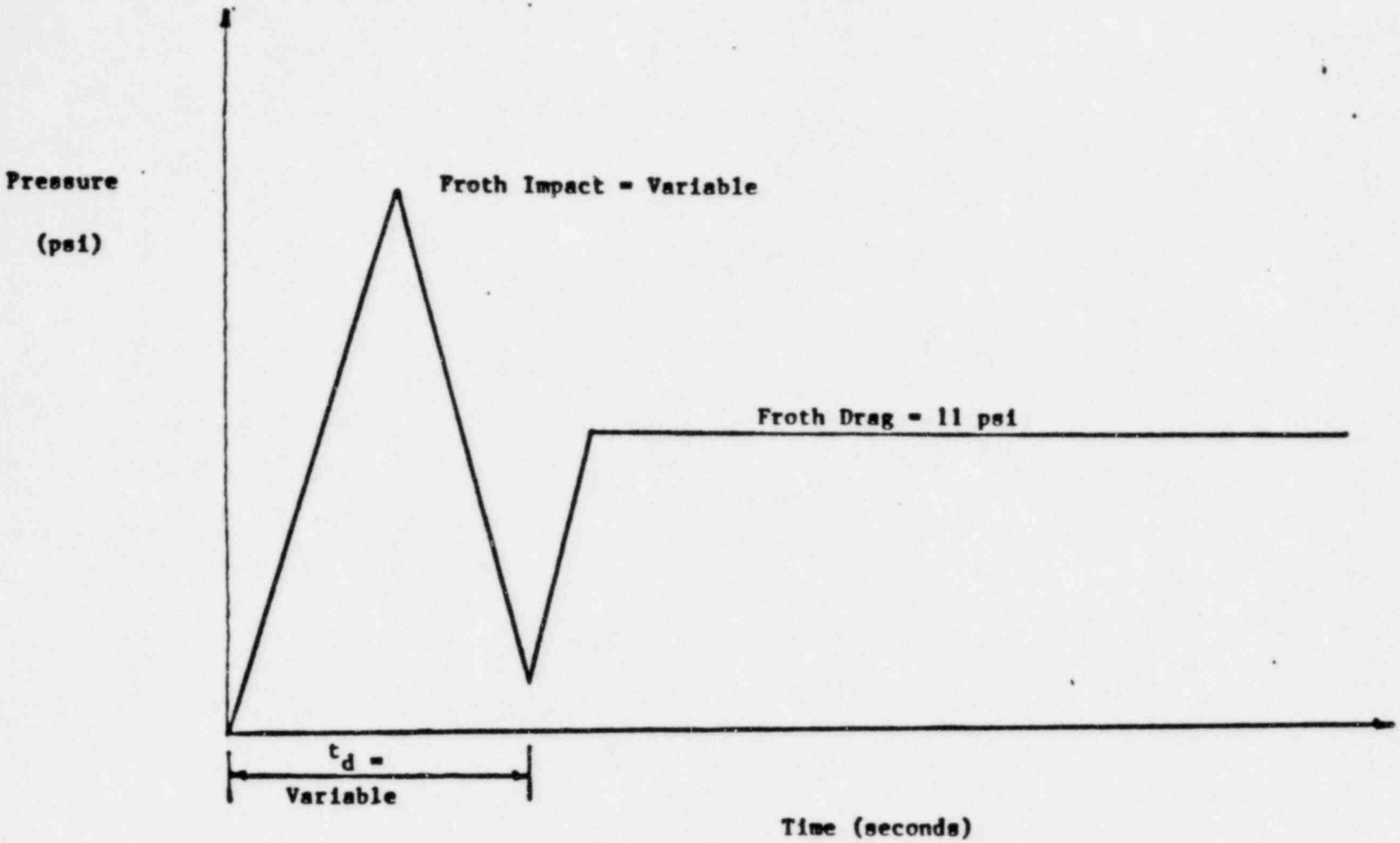

Pressure
(psi)
Froth Impact = Variable
Froth Drag = 11 psi
t_d =
Variable
Time (seconds)

Figure 1

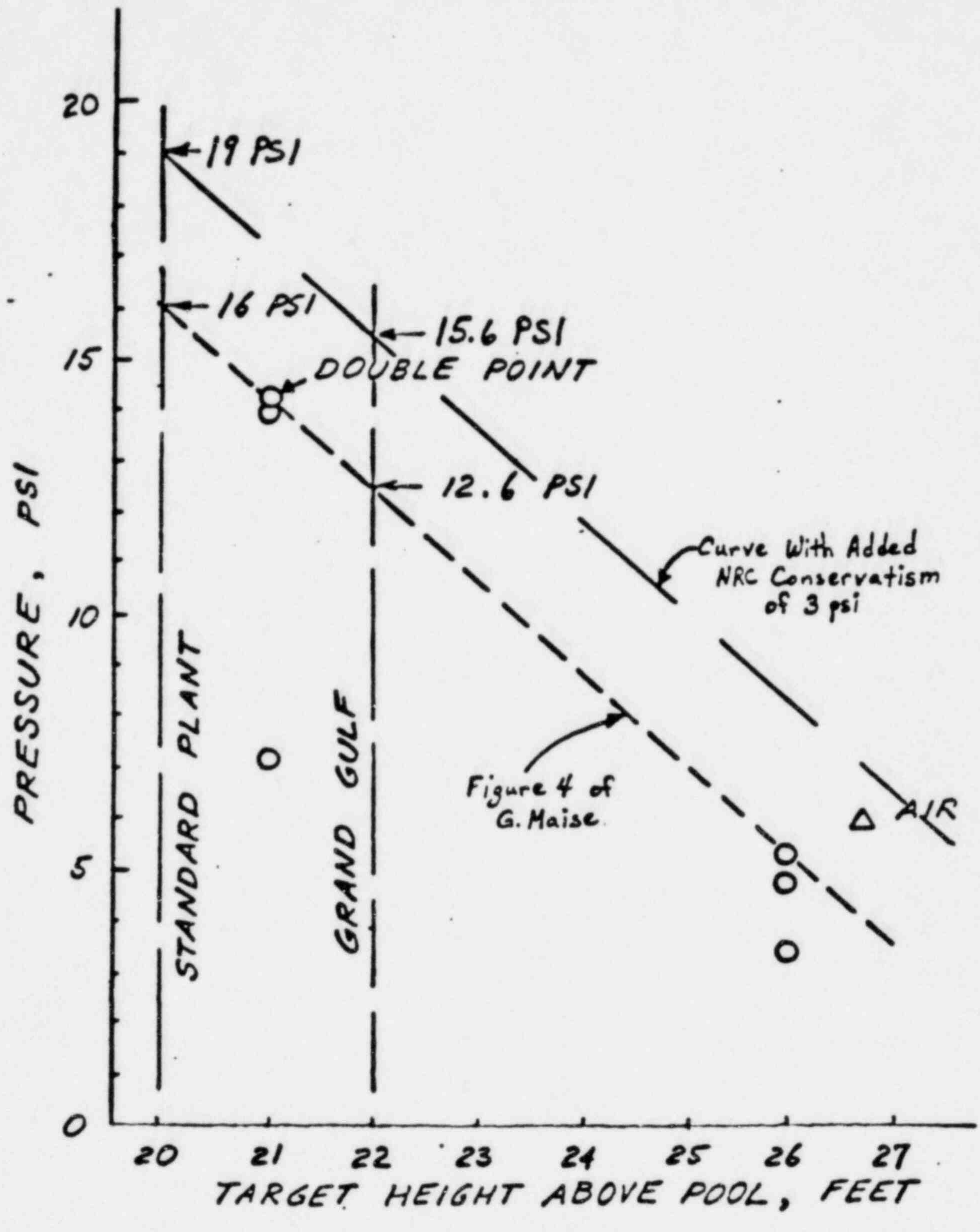


Figure 2. Froth Impact Data From Figure 3 Scaled Up to Full Size Using "Modified Froude Scaling"

VARIABLE POOL SWELL - EQUIPMENT

- FOLLOWING NRC FORMAL ACCEPTANCE OF THE GGNS LOAD DEFINITION, CURVES COMPARING THE NEW VARIABLE POOL SWELL RESPONSE SPECTRA WITH THE ORIGINAL DBA ENVELOPE RESPONSE SPECTRA WERE DEVELOPED.

  - POOL SWELL RESPONSE SPECTRA ENVELOPED WITH PREVIOUS DBA ENVELOPE RESPONSE SPECTRA TO PROVIDE REVISED DBA ENVELOPE RESPONSE SPECTRA FOR EQUIPMENT QUALIFICATION.

  - VARIABLE POOL SWELL RESPONSE SPECTRA DEVELOPED USING GLOBAL AVERAGE OF FROTH IMPACT AND DRAG LOADS.

  - DURING MEETING ON JUNE 7, 1982, NRC REQUESTED ENHANCEMENT BY 25% OF MID-SPAN RESPONSE SPECTRA FOR HCU QUALIFICATION.

- EVALUATION OF SQRT QUALIFICATION OF EQUIPMENT INSIDE CONTAINMENT INDICATED THAT:

  - POST ACCIDENT MONITORING THERMOCOUPLES MUST BE RE-QUALIFIED (COMPLETE).
  - MID-SPAN RESPONSE SPECTRA MUST BE DEVELOPED TO VERIFY HCU QUALIFICATION (COMPLETE).

CONTAINMENT PURGE

THE MARK III CONTAINMENT DESIGN DIFFERS FROM THE MARK I AND MARK II.

*ADVANTAGE	THE MAJORITY OF RELEASES FROM REACTOR COOLANT SUPPORT SYSTEMS ARE IN THE ISOLABLE PRIMARY CONTAINMENT
*DISADVANTAGE	INSPECTION AND MAINTENANCE REQUIREMENTS DURING NORMAL OPERATION REQUIRE MORE FREQUENT PERSONNEL ENTRY INTO CONTAINMENT.

AN EVALUATION PERFORMED BY MP&L CONCLUDES THAT CONTINUOUS FILTERED CONTAINMENT PURGING WILL BE REQUIRED TO KEEP PERSONNEL DOSES ALARA.

CONTAINMENT PURGE

THE CONTAINMENT VENTILATION AND FILTRATION SYSTEM PROVIDES FOR FILTERED RECIRCULATION AND TWO MODES OF PURGING THE CONTAINMENT ATMOSPHERE:

- *THE LOW VOLUME PURGE (LVP) - 500 CFM
- *THE HIGH VOLUME PURGE (HVP) - 6000 CFM

THE AMOUNT OF PURGING REQUIRED TO MEET ALARA GUIDELINES WAS ESTIMATED BASED ON THE FOLLOWING ASSUMPTIONS:

- *COOLANT LEAKAGE TO THE CONTAINMENT WOULD BE THE EXPECTED VALUES USED FOR DESIGN PURPOSES.

- *COOLANT RADIOACTIVITY CONCENTRATIONS ARE DESIGN VALUES BASED ON BWR OPERATING EXPERIENCE.

- *PERSONNEL DOSE LIMITS WOULD BE BASED ON THE GUIDELINES OF ICRP PUBLICATION 2 FOR WEEKLY ALLOWABLES AND 10 CFR 20.103.

THE EVALUATION CONCLUDED THAT CONTINUOUS OPERATION OF THE LVP MUST BE SUPPLEMENTED BY INTERMITTENT OPERATION OF THE HVP.

CONTAINMENT PURGE

NRC ACCEPTS UNRESTRICTED USE OF THE LVP BUT LIMITS USE OF THE HVP TO 1000 HOURS PER YEAR UNTIL ACTUAL OPERATING EXPERIENCE CAN BE EVALUATED.

DURING THE FIRST OPERATING CYCLE MP&L WILL COLLECT GRAND GULF SPECIFIC DATA ON THE FOLLOWING:

- *OPERATING DURATIONS OF RECIRCULATION AND PURGE MODES.
- *OPERATING COOLANT RADIOACTIVITY CONCENTRATIONS.
- *AIRBORNE RADIOACTIVITY LEVELS INSIDE CONTAINMENT.
- *TIME DURATIONS AND PERSONNEL EXPOSURES FOR INSPECTION AND MAINTENANCE ACTIVITIES DURING NORMAL OPERATION.

THE DATA WILL BE EVALUATED AND ANY PROPOSED REVISIONS FOR THE USE OF THE LVP AND HVP WILL BE SUBMITTED TO THE NRC PRIOR TO STARTUP FOLLOWING THE FIRST REGULARLY SCHEDULED REFUELING OUTAGE.

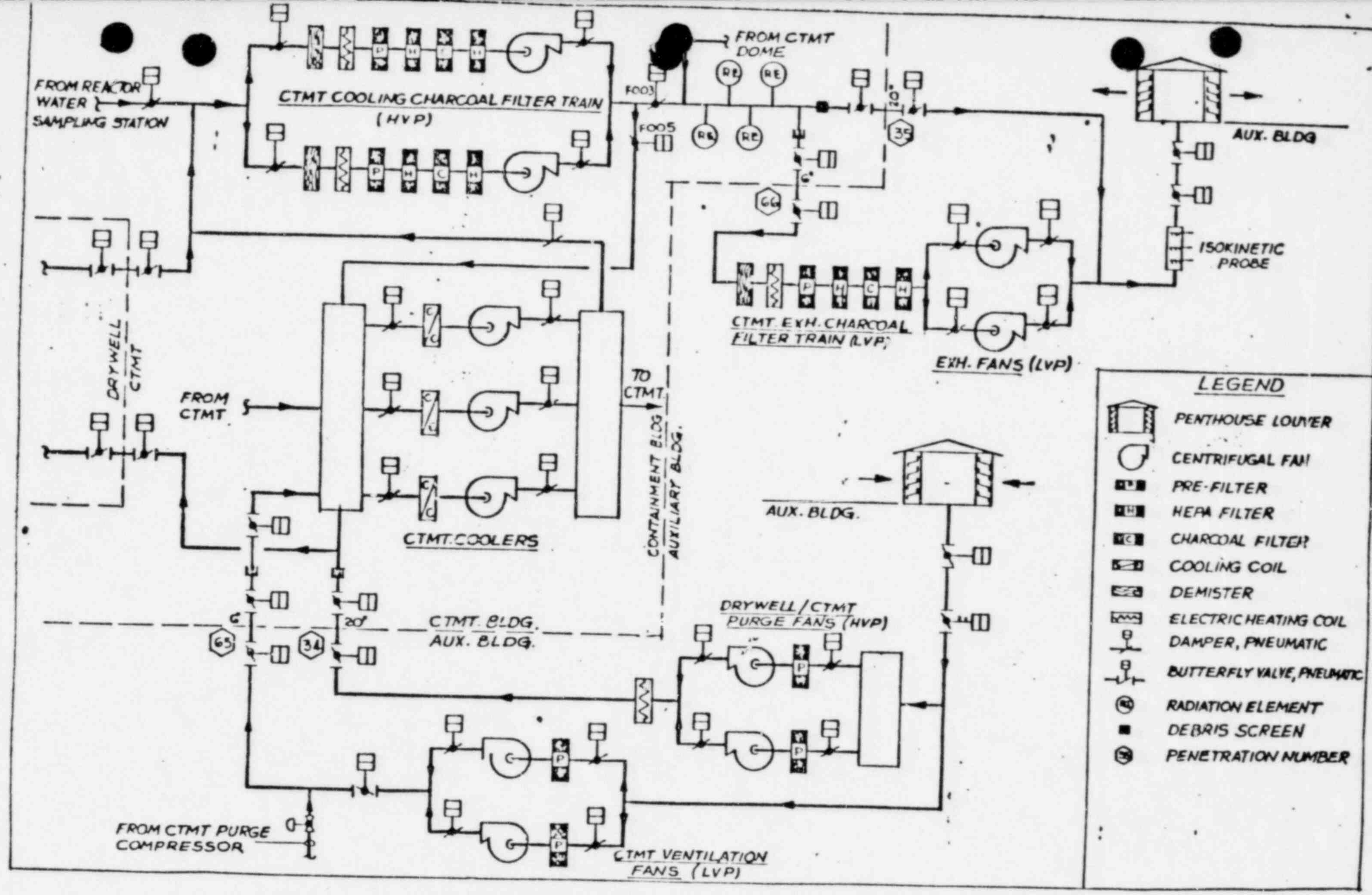


FIGURE 1.1-1

"Simplified Composite Drawing of the Containment Ventilation and Filtration System"

CONTAINMENT PURGE ASSUMPTIONS

- RADIATION DOSE RATE LIMITS:
  - WHOLE BODY/GONADS 1mRem/Hr
  - SKIN/THYROID 4mRem/hr
- AIRBORNE RADIOACTIVITY SOURCES
  - REACTOR STEAM (SAFETY RELIEF VALVES) 2000 lb/hr
  - REACTOR WATER TO CONTAINMENT ATMOSPHERE 50 lb/hr
  - REACTOR STEAM TO CONTAINMENT ATMOSPHERE 5 lb/hr

CONTAINMENT PURGE EVALUATION RESULTS

. EQUILIBRIUM DOSE RATES FOR VARIOUS PURGE RATES

Purge Rate (cfm)	Filtered Recirc. (cfm)	Equilibrium Does Rate (m/Rem/hr.) Thyroid	Beta-Skin	Whole Body
0	3000	8.2	2.1	0.31
0	6000	4.6	2.1	0.31
500	3000	6.8	1.4	0.24
500	6000	4.1	1.4	0.24
6000	N/A	3.0	0.63	0.11

CONTAINMENT PURGE RESTRICTIONS

- NOT USED FOR TEMPERATURE OR HUMIDITY CONTROL
- NO MORE THAN ONE SUPPLY LINE AND ONE EXHAUST LINE AT THE SAME TIME
- LVP UNRESTRICTED FOR OPERATIONAL CONDITIONS 1 THROUGH 5
- HVP UNRESTRICTED FOR OPERATIONAL CONDITIONS 4 AND 5 AND 1000 HOURS FOR CONDITIONS 1, 2 AND 3.

SURVEILLANCE TESTING

- PERIODIC TESTING FOR CLOSURE TIME AND LEAKAGE
- VALVES INCLUDED IN PUMP AND VALVE ISI PROGRAM
  - MINIMUM TESTING FREQUENCY OF EVERY 92 DAYS FOR LEAK TIGHTNESS
  - CLOSURE TIME EVERY 92 DAYS OR FOLLOWING ANY MAINTENANCE

PURGE VALVE OPERABILITY

- DURING A DESIGN BASIS LOCA IN DRYWELL, WILL DRYWELL PURGE/VENT ISOLATION VALVES CLOSE AGAINST DIFFERENTIAL PRESSURE

- VALVE SUPPLIER ANALYSIS WAS CONSERVATIVE AND CONSIDERED
  - WORST CASE (MSLB) POST-ACCIDENT DRYWELL PRESSURE
  - DELAY TIME FROM START OF DBA TO RECEIPT OF ISOLATION SIGNAL
  - WORST CASE CONFIGURATION (BOUNDING GRAND GULF CONFIGURATION)
  - WORST CASE FLOW AND CLOSURE DIRECTIONS

- ANALYSIS CONFIRMS THAT DRYWELL PURGE/VENT VALVES WILL CLOSE UNDER DBA CONDITIONS

- OPERABILITY ANALYSIS EXTENDED BY VENDOR TO CONTAINMENT HVPS (20") ISOLATION VALVES SINCE
  - CONTAINMENT AND DRYWELL VALVES ARE IDENTICAL
  - ANALYSIS CONFIGURATION BOUNDS CONFIGURATION OF HVPS VALVES

- PURGING DRYWELL DURING OPERATIONAL MODES 1 THROUGH 5 IS JUSTIFIED

SINGLE FAILURE DESIGN BASIS

- A DESIGN BASIS EVENT (ACCIDENT)

  PLUS

  - RESULTING FAILURES (CONSEQUENTIAL DAMAGES)

  PLUS

  - A SINGLE FAILURE
    - ACTIVE MECHANICAL FAILURE, OR
    - ACTIVE OR PASSIVE ELECTRICAL FAILURE (IEEE-379)

- SHALL NOT PREVENT REQUIRED SAFETY FUNCTIONS

CONFORMANCE TO GDC-24 AND IEEE-279

FAILURE OF A CONTROL GRADE SYSTEM CAUSING NEED FOR PROTECTIVE ACTION AND ALSO DISABLING A PROTECTION INSTRUMENT CHANNEL REQUIRES THAT AN ADDITIONAL FAILURE WILL NOT DISABLE ANY REQUIRED PROTECTIVE FUNCTIONS.

PROTECTION SYSTEMS ARE DESIGNED TO BE SEPARATE FROM CONTROL SYSTEM TO THE EXTENT THAT:

FAILURE OF ANY COMMON ELEMENT WILL LEAVE A PROTECTION SYSTEM SATISFYING:

RELIABILITY
REDUNDANCY OR DIVERSITY
INDEPENDENCE

FOR PURPOSES OF ASSURING:

- INTEGRITY OF RCPB
- CAPABILITY TO ACHIEVE & MAINTAIN SHUTDOWN
- CAPABILITY TO PREVENT OR MITIGATE CONSEQUENCES OF ACCIDENTS WHICH COULD RESULT IN POTENTIAL OFFSITE EXPOSURES COMPARABLE TO THOSE REFERENCED IN 10 CFR 100.11

DESIGN IMPLEMENTATION

- INSTRUMENT SYSTEMS AND PIPING SYSTEMS ARE ARRANGED SO THAT CONSEQUENTIAL FAILURES IN REDUNDANT OR OTHER INSTRUMENT SYSTEMS AS A RESULT OF ACCIDENTS ARE MINIMIZED

- INSTRUMENT SYSTEMS ARE DESIGNED TO MEET THE SINGLE FAILURE CRITERION, ASSUMING THE FAILURE OF THE SENSING LINE IS INDEPENDENT OF THE EVENT REQUIRING PROTECTIVE ACTION, UNLESS FOUND TO BE OTHERWISE DURING EVALUATIONS.

- EVALUATIONS ARE PERFORMED WHICH REVEAL THOSE PROTECTION SYSTEM SENSING LINES WHICH ARE AFFECTED BY EVENTS THEY MUST MITIGATE (I.E., JET IMPINGEMENT STUDY, ERT, ETC.)

  - EACH CASE IN WHICH A SENSING LINE IS AFFECTED BY THE EVENT IS EVALUATED AND IS ACCEPTABLE IF:

    - THE SENSING LINE CAN SURVIVE THE EVENT, OR
    - DIVERSE BACKUP PROTECTION IS PROVIDED, OR
    - ALL POSSIBLE FAILURE MODES DO NOT PREVENT THE REQUIRED SAFETY ACTION

  - UNACCEPTABLE CASES ARE CORRECTED BY INSTALLATION OF BARRIERS OR OTHER CHANGES TO MEET ACCEPTANCE CRITERIA

CASES OF SHARING OF COMMON INSTRUMENT TAPS

A REVIEW OF THE RPS AND ECCS WAS CONDUCTED TO EVALUATE CASES OF SHARING OF COMMON INSTRUMENT TAPS.

THE CASES FOUND AND EVALUATED WERE:

- SDV LEVEL
- RPV LEVEL
- TURBINE FIRST STAGE PRESSURE
- ADS PERMISSIVE
- CST AND SUPPRESSION POOL LEVEL

THE EVALUATIONS DETERMINED THAT THE SINGLE FAILURE DESIGN BASIS WAS MET AND NO PLANT MODIFICATIONS ARE REQUIRED.

PIPE BREAK EVALUATION

- OVER 100 BREAKS AND 200 JET CONES IN THE DRYWELL WERE EVALUATED
- SELECTION OF BREAKS BASED ON MEB 3-1 STRESS CRITERIA
- SINCE FINAL STRESS DATA UNAVAILABLE DURING EARLY DESIGN STAGES, NUMEROUS WHIP RESTRAINTS INSTALLED FOR WORST CASE POSTULATED BREAKS
- JOBSITE WALKDOWNS PERFORMED TO REVIEW "AS-BUILT" SITE CONDITIONS AND IDENTIFY THE ESSENTIAL COMPONENTS POTENTIALLY AFFECTED BY PIPE WHIP AND JET IMPINGEMENT
- WHIP RESTRAINT ADEQUACY VERIFIED AND JET BARRIERS INSTALLED AS REQUIRED
- IN PARTICULAR CRD BUNDLE ANALYZED AS ONE OF MANY ESSENTIAL TARGETS EVALUATED FOR POSTULATED HELB
- PRIOR TO 9/81 ACRS WALKDOWN, DESIGN CHANGES ISSUED TO REDUCE DEFLECTION OF WHIP RESTRAINTS (BASED ON NOZZLE LOADS) AND MODIFY UPPER/LOWER CRD SUPPORTS
- DETAILED ANALYSIS TO ASSESS REAL NEED FOR SHIELDS BASED ON CONSIDERATION OF ISI ACCESSIBITY AND ALARA
- FINAL RECIRC STRESS DATA REVIEWED AGAINST MEB 3-1 CRITERIA
- AS A RESULT OF REVIEW, LONGITUDINAL BREAKS NOT REQUIRED TO BE POSTULATED HOWEVER WHIP RESTRAINTS INSTALLED
- ELASTO-PLASTIC ANALYSIS OF CRD TUBES AND SUPPORTS PERFORMED FOR RECIRC JET IMPINGEMENT
  - ANALYSIS FOR SINGLE TUBE
  - AS TUBE DEFORMS, TUBES IN BACK ACT AS GROUP
  - GROUP DECREASES DEFORMATION (NO CREDIT TAKEN FOR THIS EFFECT)
  - IMPACT EFFECT BETWEEN TUBES WAS EVALUATED AND FOUND ACCEPTABLE
  - 90% FLOW AREA MAINTAINED

PIPE BREAK EVALUATION (CONT'D)

- TO SUPPORT CONTROL ROD INSERTION PERFORMANCE, 35% FLOW AREA REQUIRED
- TO SUPPORT DEFORMATION ANALYSIS, BEND TEST PERFORMED AT M.S.U. FOR 1" SCHEDULE 80, STAINLESS STEEL PIPE
  - FLOW AREA MEASURED FOR VARIOUS BEND ANGLES
  - RESULTS CORRELATED WITH ELASTO-PLASTIC ANALYSIS TO DETERMINE FLOW AREA
- ISI PROGRAM CONTINUALLY UPGRADED TO INCORPORATE LATEST NRC CONCERNS

PIPE BREAK EVALUATION (CONT'D)

- EXAMPLES OF PREVENTATIVE MEASURES
  - FEEDWATER VESSEL NOZZLES MODIFIED TO MINIMIZE CRACKING
  - RECIRC PIPING MODIFIED TO REDUCE IGSCC
  - EXAMINATION FREQUENCY FOR FIELD WELDS INCREASED TO MINIMIZE IGSCC
  - 1977 SECTION XI ASME CODE USED (INSPECTION OF SAME WELDS EVERY 10 YEARS TO DETECT DEGRADATION)

PIPE BREAK EVALUATION (CONT'D)

- THESE CRD MECHANISMS WITH INSERT AND WITHDRAWAL LINES SEVERED WILL SCRAM LESS THAN FOUR SECONDS DUE TO REACTOR PRESSURE
- ON DBA, REACTOR AT GREATER THAN 1000 PSI FOR 5 SECONDS
- FOR CRD WITHDRAWAL LINE, SCRAM FUNCTION NOT IMPAIRED PROVIDING FLOW AREA AFTER CRIMPING IS $\geq$ 35%
- 50% OF RODS IN A RANDOM/SCATTERED ARRAY WILL ALLOW HOT STANDBY/ZERO POWER
- WORSE CASE 5 RODS FULL OUT IN A GROUP WILL ALLOW HOT STANDBY/ ZERO POWER

GRAND GULF

STATUS OF REVIEW

DEAN HOUSTON

PROJECT MANAGER

NRC

GRAND GULF

CHRONOLOGY

SAFETY EVALUATION REPORT	SEPTEMBER 9, 1981
ACRS SUBCOMMITTEE MEETING	SEPTEMBER 17-18, 1981
ACRS FULL COMMITTEE MEETING	OCTOBER 16, 1981
ACRS INTERIM REPORT	OCTOBER 20, 1981
SER SUPPLEMENT NO. 1	DECEMBER 16, 1981
SER SUPPLEMENT NO. 2	JUNE 16, 1982
OPERATING LICENSE (LOW POWER)	JUNE 16, 1982
SER SUPPLEMENT NO. 3	JULY 21, 1982

GRAND GULF STATUS OF OUTSTANDING ISSUES

	ISSUE	STATUS	SECTION
(1)	Damping value for cable tray design	Resolved	3.7.3 (SSER 1)
(2)	Ultimate containment capacity	Resolved	3.8.1 (SSER 1) II.B.7 (SSER 3)
(3)	Tangential shear - drywell	Resolved	3.8.1, 3.8.4 (SSER 1)
(4)	Hydrodynamic LOCA loads - MARK III	Resolved pending confirmation	3.10, 3.11 (SSER 3) 3.8.1, 6.2.1 (SSER 2)
(5)	Load combination equations	Resolved	3.8.3 (SSER 1) 3.9.3 (SSER 2)
(6)	Electrical equipment qualification	Resolved with license conditions	3.11, 3.10 (SSER 2)
(7)	ODYN Code calculations	Resolved	5.2.2, 15.1 (SSER 1) 4.4.1 (SSER 2)
(8)	Containment isolation	Resolved	6.2.4 (SSER 1)
(9)	Containment purge	Resolved with license conditions	6.2.4.1, II.E.4.2 (SSER 2)
(10)	Single failure in SRV low-low setpoint function	Resolved	7.8 (SSER 1)

Continued

GRAND GULF STATUS OF OUTSTANDING ISSUES

	ISSUE	STATUS	SECTION
(11)	Single sequencer reliability	Resolved	8.4.5 (SSER 2)
(12)	Nonsafety loads on emergency sources	Resolved	8.4.6 (SSER 2)
(13)	Management capability and organization	Resolved with license conditions	13.0, I.A.1.1, and I.A.1.2 (SSER 2)
(14)	Emergency preparedness plan	Resolved (low power)	13.3 (SSER 2)
(15)	Operating and emergency procedures	Resolved	I.C.2, I.C.3, I.C.5, and I.C.6 (SSER 2)
(16)	Control room access and instrumentation	Resolved	I.C.4 (SSER 2)
(17)	Hydrogen igniter system	Resolved for interim operation with licensee condition	II.B.7, II.B.8 (SSER 3)
(18)	Reactor vessel level instrumentation	Resolved	II.K.1.23 (SSER 1)
(19)	Common reference water level instrumentation	Resolved	II.K.3.27 (SSER 1)
(20)	Recent containment concerns	Awaiting information	6.2.9 (SSER 3)

ISSUES INTRODUCED SINCE LAST ACRS MEETING

- LPCI MODIFICATION
- PMP FLOOD ANALYSIS
- CONTAINMENT CONCERNS (HUMPHREY)
- INDEPENDENT DESIGN VERIFICATION
- STAFFING CHANGES - PLANT OPERATING STAFF AND CSRC CONSULTANTS

MISSISSIPPI POWER & LIGHT COMPANY
Helping Build Mississippi
P. O. BOX 1640, JACKSON, MISSISSIPPI 39205

June 25, 1982

NUCLEAR PRODUCTION DEPARTMENT

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Unit 1 and 2
Docket Nos. 50-416 and 50-417
File 0260/L-860.0/0756
Outstanding Information Reques
for Hydrogen Control
AECM-82/294

Enclosed are the Mississippi Power & Light Company (MP&L) responses to four NRC review questions received via telecopy June 23, 1982 from your Mr. Schwencer.

During the past few months MP&L has transmitted to the NRC numerous submittals on the Hydrogen Control Issue. This was performed on an informa question - formal answer basis. It is our understanding that these four concerns must be addressed prior to issuance of the interim approval of the Hydrogen Control Issue and MP&L has put forth significant efforts into the resolution or interim resolution of each of the items listed below.

1. Concern
An Evaluation of the response of the air-lock to a local detonation.

Response
This concern has been addressed in AECM-82/292 dated June 25, 198

2. Concern
An Evaluation of Pool dynamic impact loads and pool carry-over du to hydrogen combustion.

Response
See attachment I to this letter (AECM-82/294 dated June 25, 1982)

3. Concern
An Expanded evaluation of equipment survivability for pressure, especially for the drywell vacuum breakers and drywell purge compressors.

Response
This concern has been addressed in AECM-82/296 dated June 25, 19

MISSISSIPPI POWER & LIGHT COMPANY

4. Concern
Identification of the valve to be used in Emergence Procedure 05-S-01-EP-9 for venting and an evaluation of their operability at the expected pressure differentials.

Response
It is MP&L's intention to follow this item closely, as it is an owner's group (BWR) issue, and it pertains to the Generic Emergency Procedure Guidelines prepared for the use of all BWR's. As the NRC has requested, MP&L addressed the issue of venting in AECM-82/276 dated June 15, 1982. The subject of venting the containment has been a topic of discussion in the industry for some time and resolution of this issue is expected in the near future. The companion subject of valve operability will only follow (not lead) the venting resolution and will only be applicable if containment venting, for such purposes as pressure relief, is contained in the resolution.

It is our understanding that this completed the efforts in the Hydrogen Control area and that the SSER and the subsequent ACRS may take place forthwi

Yours truly,

L. F. Dale
Manager of Nuclear Services

RMS/SHH/JDR:de
Attachment

cc: Mr. N. L. Stampley (w/a)
Mr. R. B. McGehee (w/a)
Mr. T. B. Conner (w/a)
Mr. G. B. Taylor (w/a)

Mr. Richard C. DeYoung, Director (w/a)
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. J. P. O'Reilly, Regional Administrator (w/a)
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Region II
101 Marietta St., N.W., Suite 3100
Atlanta, Georgia 30303

bcc: Dr. D. C. Gibbs (w/o)
Mr. A. Zaccaria (w/o)
Mr. L. E. Ruhland (w/o)
Mr. R. S. Trickovic (w/a)
Mr. C. D. Wood (w/o)
Mr. J. F. Hudson, Jr. (w/o)
Mr. T. H. Cloninger (w/o)
Mr. J. P. McGaughy (w/o)
Mr. T. E. Reaves (w/o)
Mr. C. K. McCoy (w/o)
Mr. J. W. Yelverton (w/o)
Mr. A. R. Smith (w/o)
Mr. R. F. Phares (w/a)
Mr. A. G. Wagner (w/a)
Mr. C. C. Hayes (w/a)
Mr. M. D. Houston (w/a)
Mr. J. F. Pinto (w/o)
Mr. M. D. Archdeacon (w/o)
File (w/a)

ATTACHMENT I

Pool Dynamic Impact Loads and Pool Carry-Over

due to Hydrogen Combustion

The evaluation of pool dynamic loads has been performed previously as it applies to Loss of Coolant Accidents (LOCA's). These loads have been evaluated by MP&L and the results show no degradation of Plant Safety. In concert with this, MP&L evaluated its drywell burn base case flame speed of 6 fps and found that the drywell pool swell velocity was of the same magnitude as the LOCA evaluation, indicating that Grand Gulf's initial LOCA evaluation remains valid.

In addition, it is MP&L's position that no global type burn will occur in the drywell, given such an event, but that an inverted flame will occur providing only a modest increase (less than 3 psi) in pressure, which will have no effect on suppression pool dynamics as discussed in AECM-82/25 dated March 2, 1982.

The following should also be considered: 1) that this event, Drywell/Small Break LOCA, is significantly lower in probability of occurrence (about 5×10^{-8}) than a transient induced Stuck Open Relief Valve (about 2×10^{-7}) as discussed in the Hydrogen Control Owner Group letter HGN-003 dated April 8, 1982, 2) that the use of a flame speed greater than 6 fps, i.e. 12 fps is quite unrealistic in the regime presented, 3) that utility sponsored testing to date has shown that flame speeds, in hydrogen concentrations around the 9 v/o level are less than anticipated (on the order of 4 fps *), and 4) planned tests will address the question of burn characteristics in such a regime and that the results will support an "inverted flame" as opposed to a "global burn" in the drywell.

* Preliminary results of EPRI sponsored tests at Whiteshell

It is MP&L's belief that the planned testing will show burn phenomenan and flame speed which are more closely characterized by continuous inverted flames or the drywell base case than by the 12 fps drywell burn case.

MP&L then believes that the information submitted to date provides an appropriate basis for the interim evaluation. Further evaluation of the 12 fps case for drywell pool swell will be carried out if test results indicate such evaluation is warranted. It is anticipated however, that other program on burn phenomena will demonstrate that this case is excessively conservative.

MISSISSIPPI POWER & LIGHT COMPANY
Helping Build Mississippi
P. O. BOX 1640, JACKSON, MISSISSIPPI 39205

NUCLEAR PRODUCTION DEPARTMENT

June 15, 1982

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
File 0260/0756/L-800.0
Emergency Procedures -
Containment Venting
AECM-82/276

Mississippi Power & Light Company (MP&L) has been requested by your staff to submit a change to the Grand Gulf Nuclear Station Emergency Procedures, specifically Procedure 05-S-01-EP-9 ("RPV Flooding"). This procedure addresses the venting of the primary containment at the design pressure of 15 psig and does not take into account the effects on containment ultimate capacity. The ultimate capacity evaluation has shown that the lower bound for the Grand Gulf Nuclear Station is 62 psig (AECM-81/221, dated June 19, 1981) and in light of that information MP&L is changing Step 3.5 to read:

If containment pressure exceeds 50 psig, vent the containment per (04-S-01-M51-1) to reduce pressure below 15 psig.

In addition to the above actions, it is our understanding that the NRC intends to address this subject on a generic basis with the BWROG in the near future. MP&L will evaluate any results that are achieved from the generic efforts and will initiate action on an as-needed basis, depending upon its applicability, schedule impact, cost benefit, etc.

MP&L believes this information should be sufficient to permit completion of the interim evaluation of the Hydrogen Ignition System.

Yours truly,

L. F. Dale
Manager of Nuclear Services

RMS/SHH/JDR:nll

cc: (See Next Page)

cc: Mr. N. L. Stampley (w/o)
Mr. R. B. McGehee (w/o)
Mr. T. B. Conner (w/o)
Mr. G. B. Taylor (w/o)

Mr. Richard C. DeYoung, Director (w/o)
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. J. P. O'Reilly, Regional Administrator (w/a)
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Region II
101 Marietta St., N.W., Suite 3100
Atlanta, Georgia 30303

REVIEW OF GRAND GULF
HYDROGEN IGNITION SYSTEM

STAFF APPROACH

- EVALUATION OF THE HIS PERFORMED TO DETERMINE ADEQUACY ON AN INTERIM BASIS
- LICENSE CONDITIONS IMPOSED TO REQUIRE DEMONSTRATION OF SAFETY MARGINS WITHIN APPROXIMATELY ONE YEAR
  - TESTING
  - ANALYSES
- FINAL EVALUATION BY THE STAFF TO CONFIRM ADEQUATE SAFETY MARGINS

INTERIM EVALUATION OF GRAND GULF HIS

- OBJECTIVE

  DETERMINE EFFECTIVENESS OF HIS IN CONTROLLING CONSEQUENCES OF HYDROGEN RELEASES FROM A TMI-TYPE DEGRADED CORE ACCIDENT IN ORDER TO PREVENT BREACH OF CONTAINMENT AND ALLOW SAFE SHUTDOWN

- BASES FOR EVALUATING HIS WAS THE TESTING AND ANALYSES PERFORMED (REFERENCED) BY MP&L AUGMENTED BY STAFF CONFIRMATORY ANALYSIS AND TESTING

  - PREVIOUS TESTING PERFORMED BY ICOG, LLNL AND SANDIA REFERENCED BY MP&L TO DEMONSTRATE IGNITER PERFORMANCE

  - INDEPENDENT EVALUATION OF HIS BY SANDIA NATIONAL LABORATORY

  - MP&L ENDORSEMENT OF HCOG RESEARCH PROGRAM

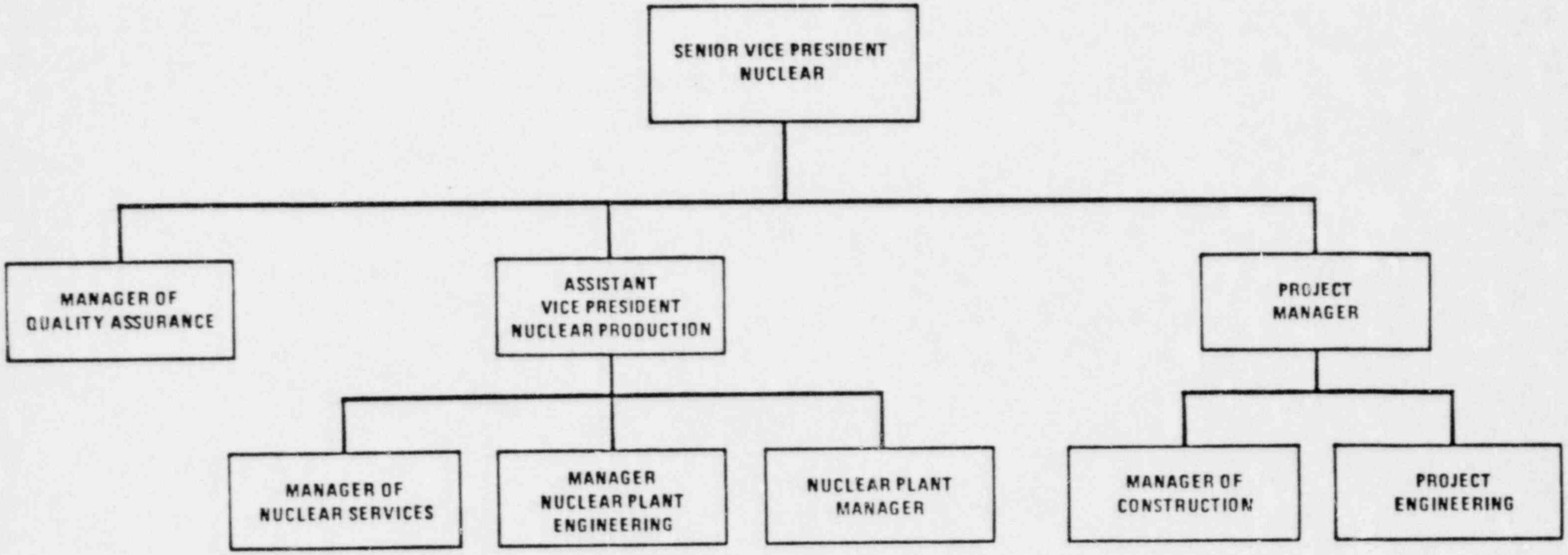
- CONCLUSION

  HIS FOUND ADEQUATE ON AN INTERIM BASIS CONDITIONAL TO SUCCESSFUL QUALIFICATION OF IGNITER ASSEMBLY. (SCHEDULED COMPLETION 8/82)

TOPICS FOR FINAL REVIEW

- MARK III COMBUSTION PHENOMENA
  - VERIFICATION OF WETWELL IGNITER PERFORMANCE
  - DRYWELL COMBUSTION
  - MIXING
- CLASIX - 3 VERIFICATION AND CONTAINMENT ANALYSIS
- ACCIDENT SCENARIOS
- HIS DESIGN
- EMERGENCY PROCEDURES
  - CONTAINMENT PURGE
  - SPRAY ACTUATION
- EQUIPMENT SURVIVABILITY

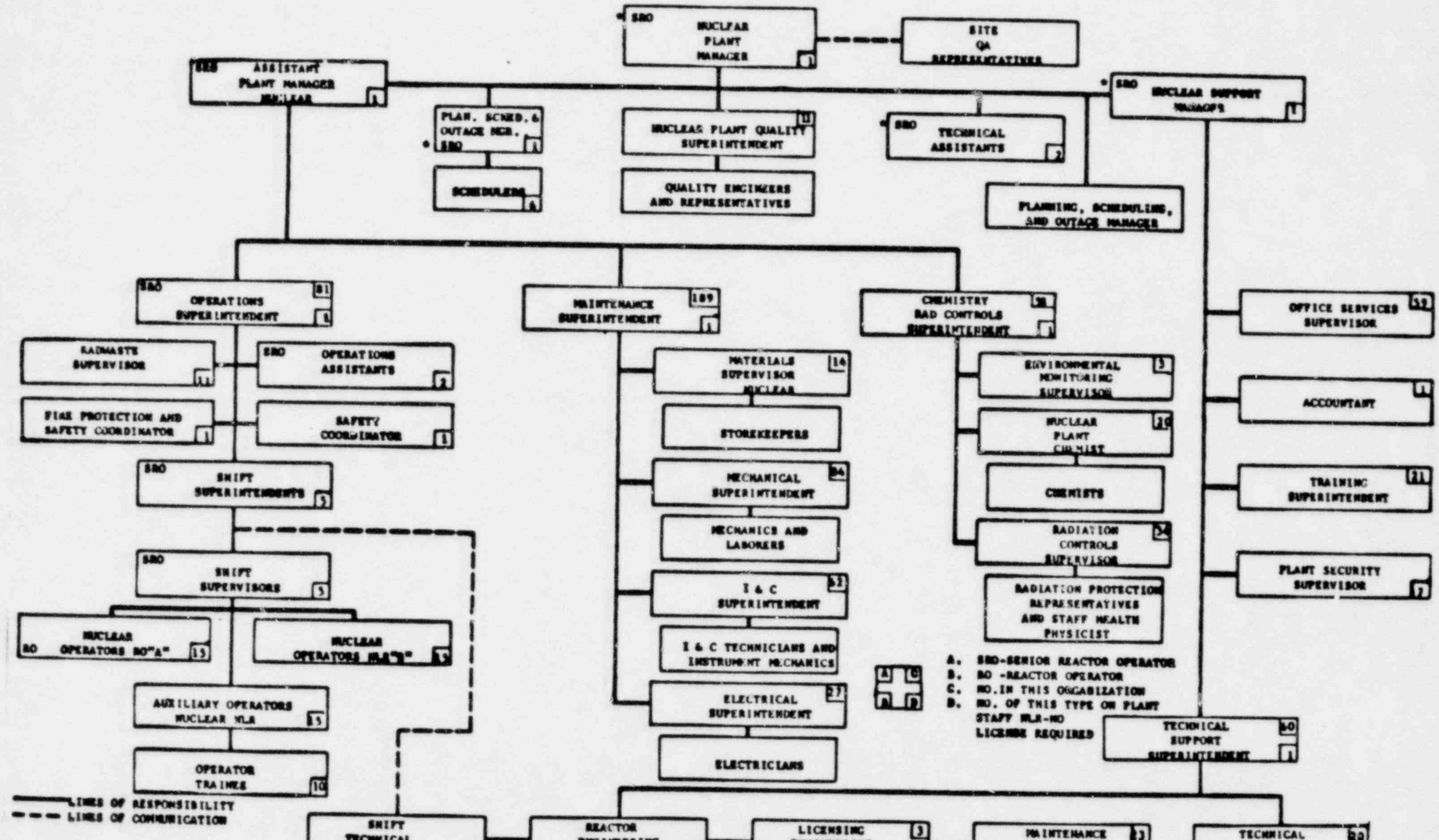
TECHNICAL SUPPORT ORGANIZATION
(UNIT TWO OPERATION)



EXPERIENCE OF
MP&L TECHNICAL SUPPORT PERSONNEL
FOR GRAND GULF

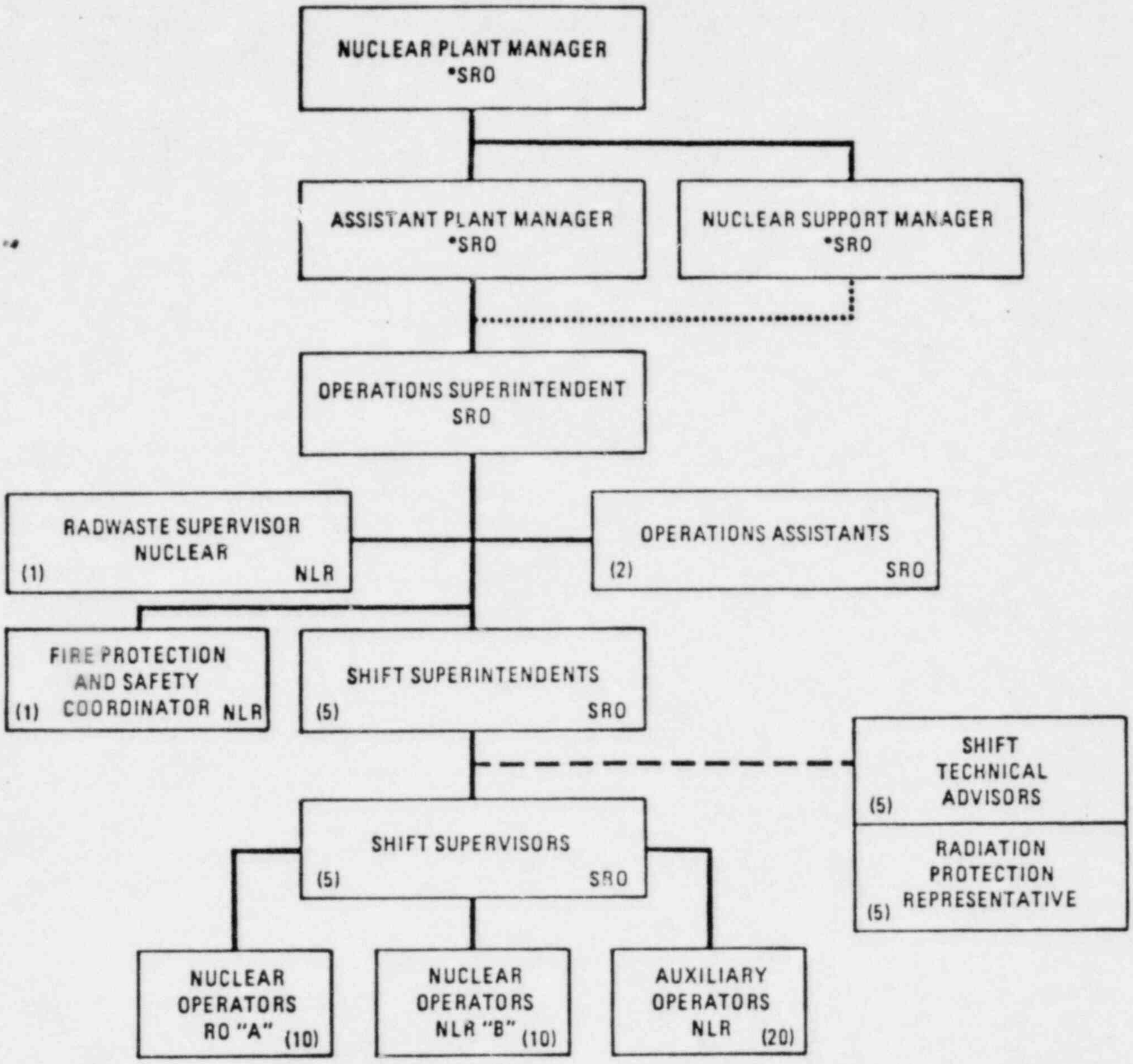
Organization	Engr. Degree	Related Science Degree	Other Degrees	Total Profs.	Professional Experience Total	Professional Experience Avg.	Nuclear Experience Total	Nuclear Experience Avg.
Construction	3	3	1	7	127	18	61	8.7
Startup	8	1	-	9	77.8	8.6	58.8	6.5
Project Engineering	2	-	-	2	12	6	9	4.5
NPE	38	4	3	47	354	7.5	225.5	4.8
Nuclear Support	21	9	4	32	284	8.9	162	5.1
Nuclear Services	17	13	7	21	228	10.9	145.5	7
QA	15	7	4	30	362	12	165	6
Subtotal	105	41	25	170	1778	10.5	1052	6.2
MSS	39	11	4	48	580	12	397	8.3
TOTAL	144	52	29	218	2358	11	1449	6.6

PLANT STAFF ORGANIZATION AND STAFFING LEVELS



* MUST HAVE BEEN SRO OR BEEN CERTIFIED SRO FOR A PLANT OF THIS TYPE.

PLANT OPERATIONS ORGANIZATION
(ONE UNIT OPERATION)



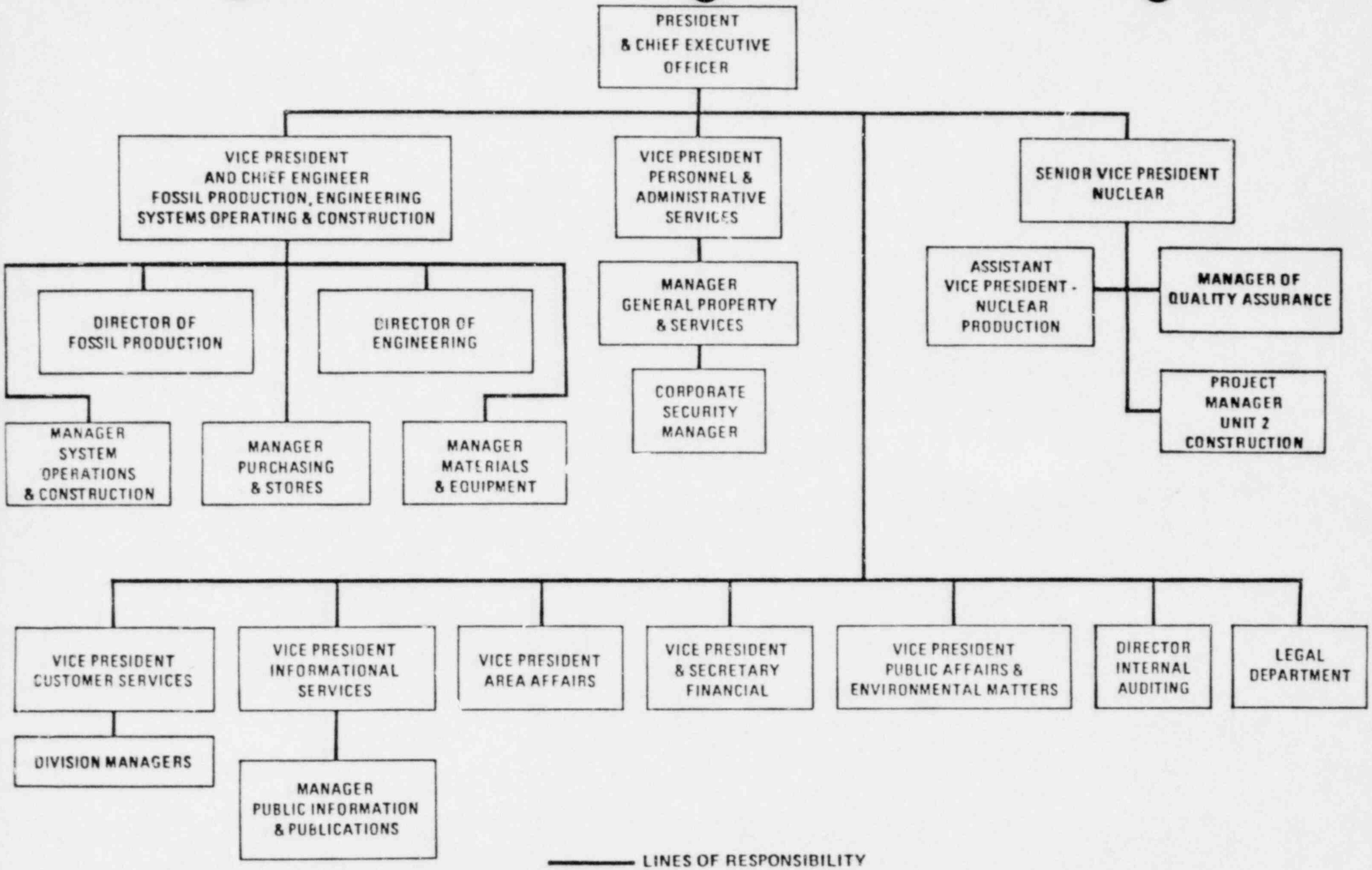
NOTES:
SRO - SENIOR REACTOR OPERATOR
RO - REACTOR OPERATOR
NLR - NO LICENSE REQUIRED
(X) - NUMBER OF PLANT PERSONNEL ASSIGNED TO THIS POSITION
— — - SHIFT TECHNICAL ADVISOR COMMUNICATES WITH SRO'S BUT REPORTS TO THE REACTOR ENGINEERING SUPERVISOR
* - THE PLANT MANAGER, THE ASSISTANT PLANT MANAGER, AND THE NUCLEAR SUPPORT MANAGER ARE TRAINED TO SRO LEVEL IN ADDITION TO THOSE IN THE OPERATIONS ORGANIZATION.
.... - TEMPORARY LINE OF SUCCESSION IN THE EVENT OF INCAPACITY OF BOTH THE PLANT MANAGER AND THE ASSISTANT PLANT MANAGER. SEE SUBSECTION 13.1.2.2.1

POST-ACCIDENT CONTAINMENT VENTING

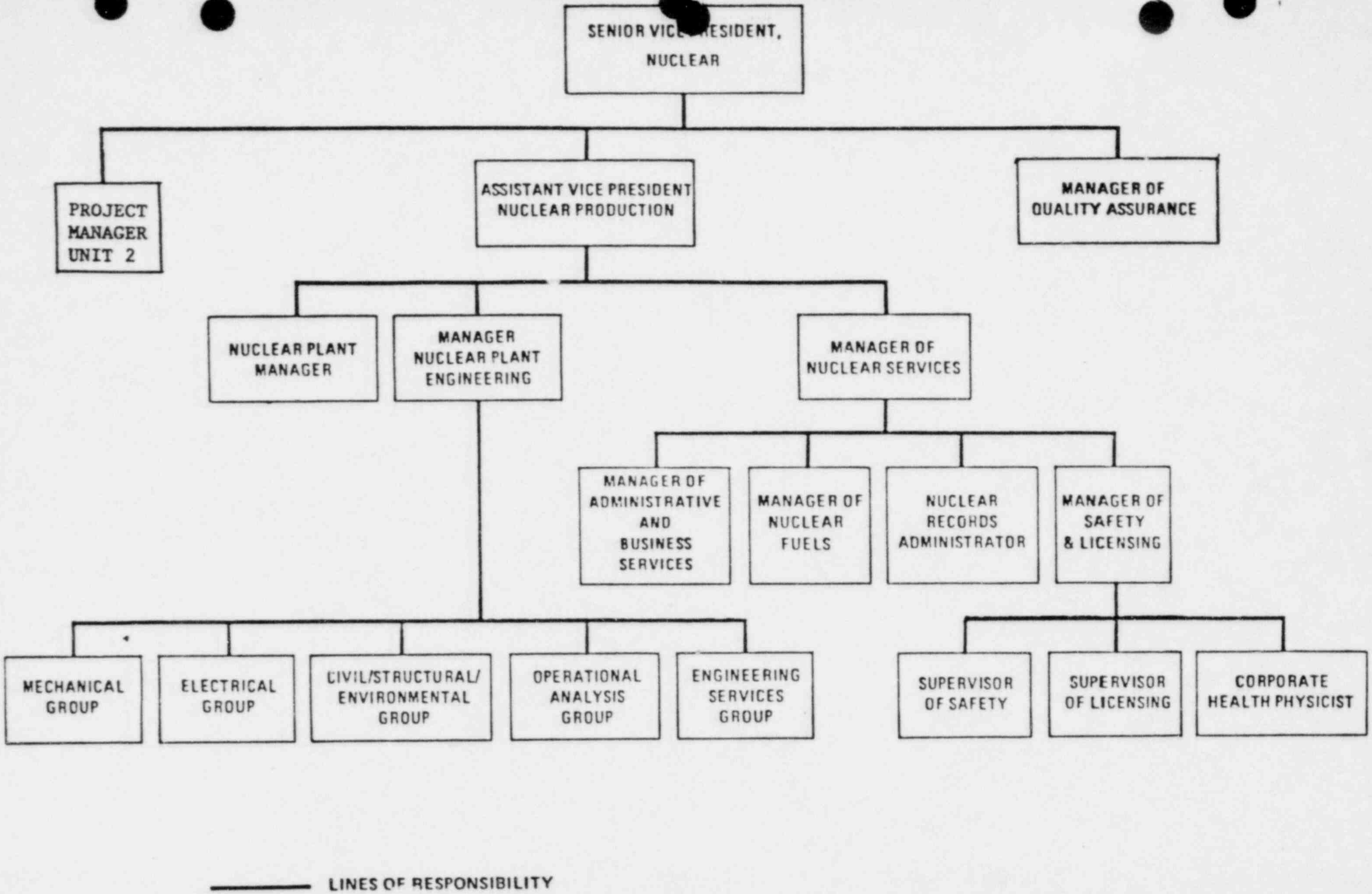
CHRONOLOGY

- EPGs LARGELY DEVELOPED PRIOR TO CONTAINMENT ULTIMATE CAPACITY ANALYSIS
- ALLOWED <u>OPTIONAL</u> CONTAINMENT VENTING AT DESIGN PRESSURE
- NRC CONCERNED THAT VENTING DURING HYDROGEN GENERATION EVENT IS UNANALYZED
- MP&L JUDGEMENT IS THAT VENTING WILL MITIGATE
- TO RESOLVE CONCERN, MP&L COMMITTED TO RAISE VENT PRESSURE TO 50 PSIG (HIGHER THAN PEAK BURN PRESSURE, LOWER THAN ULTIMATE CAPACITY)
- CONCERN IS VENT OPERABILITY
- MP&L WORKING WITH TMI BWROG REGARDING EPGs

MP&L CORPO[illegible] ORGANIZATION



TECHNICAL SUPPORT ORGANIZATION



POST ACCIDENT CONTAINMENT VENTING (CONT'D)

CONCERNS

- CONCERN RELATED ONLY TO DEGRADED CORE/SEVERE ACCIDENT
- CONTAINMENT PROTECTION (PRESSURE RELIEF)
- CURRENT CONTAINMENT VENT/PURGE SYSTEM NON-SAFETY GRADE (EXCEPT ISOLATION VALVES)
- RADIOLOGICAL/SYSTEM PRESSURE CONCERNS (FILTERED EFFLUENT, BUT DUCTWORK, FILTER TRAINS, ETC. NOT INTENDED FOR PRESSURES OF THIS MAGNITUDE)

POST-ACCIDENT CONTAINMENT VENTING (CONT'D)

RESOLUTION

- PURSUE ISSUE GENERICALLY WITH BWROG
- FEASIBILITY/DESIRABILITY STUDIES ON SYSTEM REQUIREMENTS AND OPERATIONAL CONSEQUENCES HAVE BEEN INITIATED
- VENTING NOT NEEDED FOR CONTAINMENT INTEGRITY PROTECTION FOR DEGRADED CORE/HYDROGEN CONCERNS

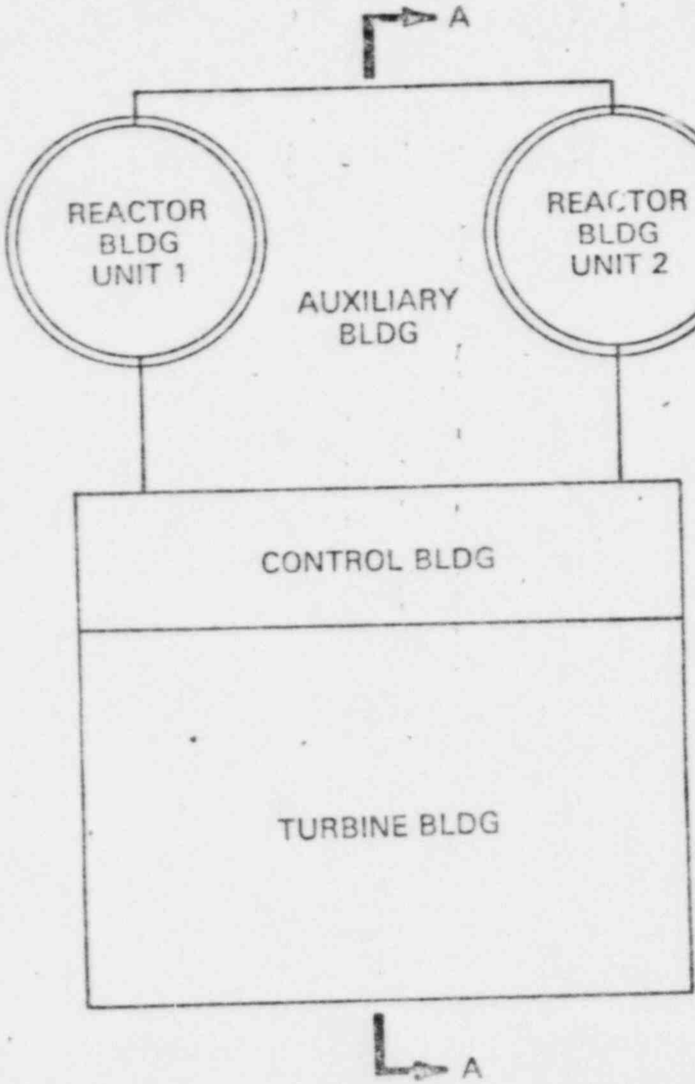
REPORT TO THE ACRS 268TH MEETING

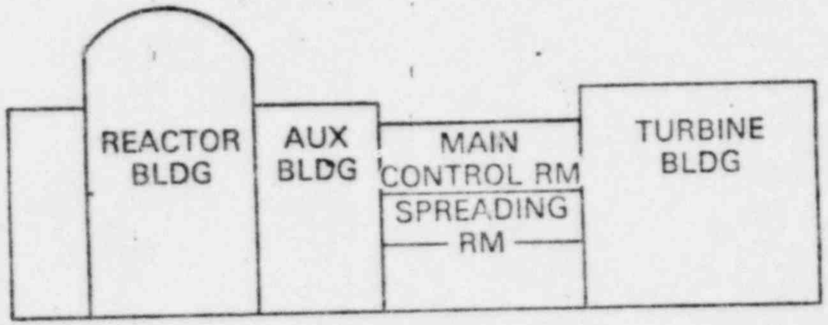
AUGUST 12, 1982

TVA EXPERIENCE IN CONTROL ROOM HABITABILITY DESIGN

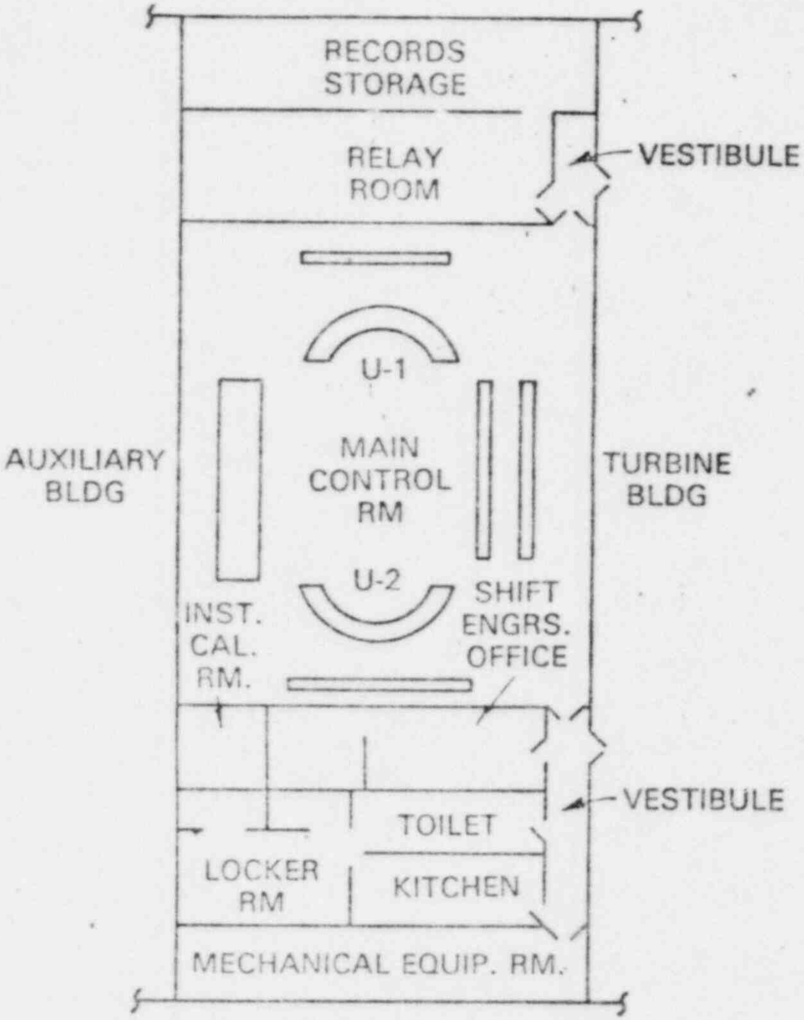
MR. L. J. KLAES,
SENIOR MECHANICAL ENGINEER
ENVIRONMENTAL CONTROL SYSTEMS
TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

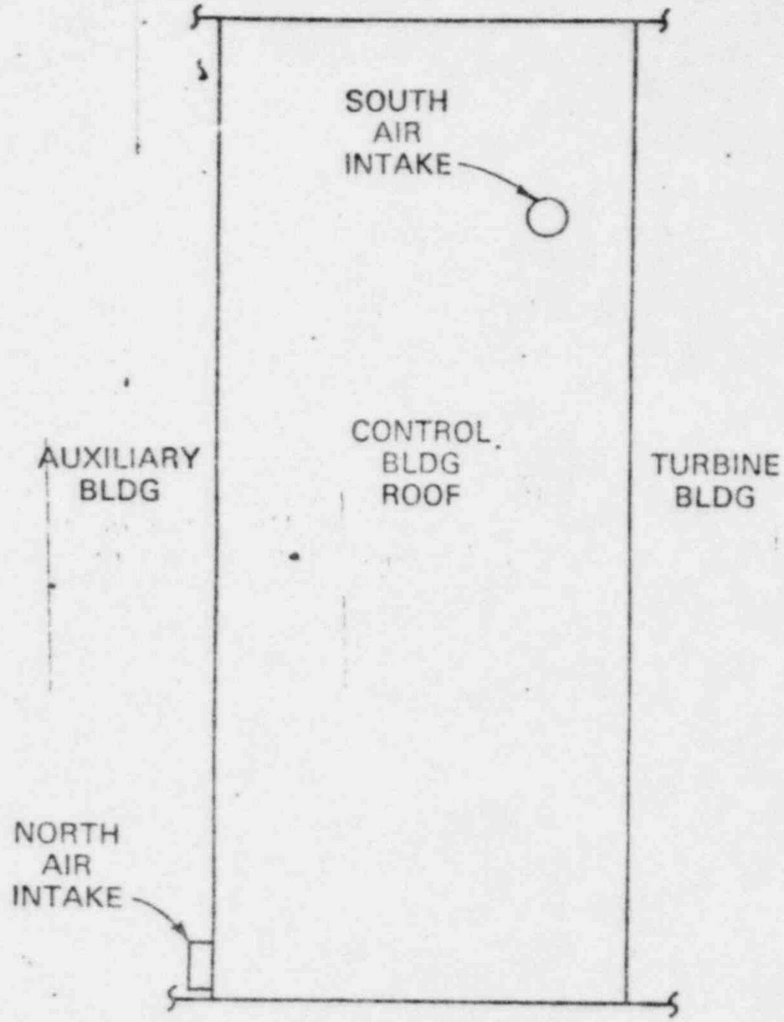




A-A



HABITABILITY ENCLOSURE



AIR INTAKE LOCATIONS

MAIN CONTROL ROOM HABITABILITY DESIGN CONSIDERATIONS

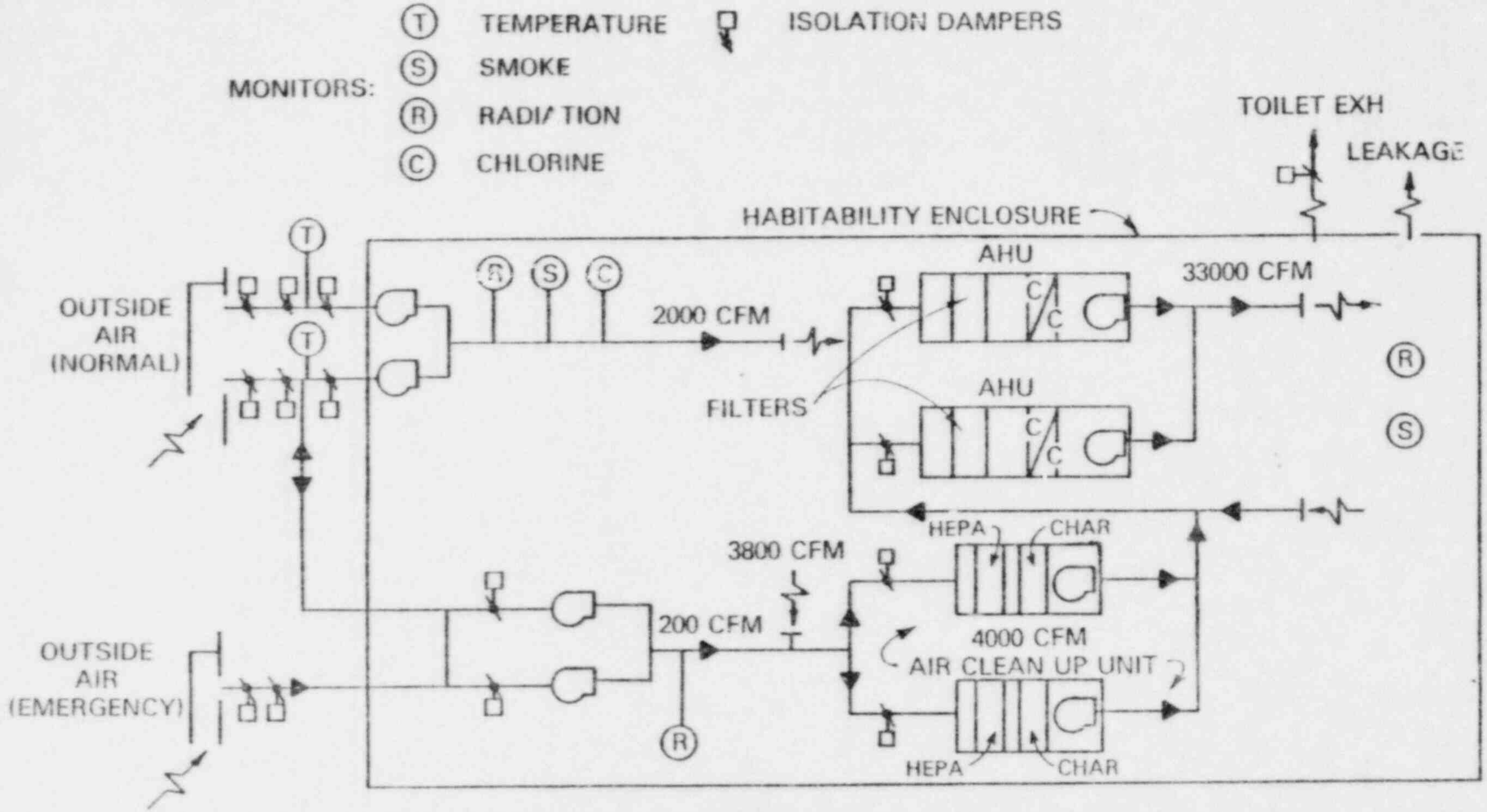
1. Radiation Hazards
   - A. Sources
   - B. Protection Features
   - C. Dose Analysis Results
2. Toxic Hazards
3. Natural Hazards
4. Environmental Control
5. Fire Protection
6. System Reliability

1.A. RADIATION-SOURCES

1. Gamma and Beta Sources Due to Radioactive Air That Enters Into the MCR From the Ventilation System, Personnel Access, and Leakages
2. Postaccident Gamma Source Surrounding the Main Control Room Due to Releases From the Containment
3. Postaccident Gamma Source From the Primary Containment Atmosphere
4. Postaccident Gamma Source in the Auxiliary Building Due to Inleakage From the Containment
5. Ingress and Egress Between MCR and Site Boundary

1.B. RADIATION-PROTECTION FEATURES

- Concrete Roof, Floor and Walls for Shine Protection
- Heavy Doors for Shine Protection
- Low Leakage Enclosure
- Radiation Monitors Activate Alarms and Initiate Emergency Operating Features
- Restricted Flow Emergency Pressurization
- Air Cleanup of Emergency Recirculated and Pressurization Air
- Portable Breathing Apparatus
- Protective Clothing



ENVIRONMENTAL CONTROL AND AIR CLEAN UP SYSTEM INCLUDING NORMAL AND EMERGENCY VENTILATION SYSTEMS FOR PRESSURIZATION

1.C. RADIATION-DOSE ANALYSIS RESULTS

CONTROL ROOM PERSONNEL DOSE FOR DBA POSTACCIDENT PERIOD

Source	Whole Body Gamma Dose (rem)*	Personnel Dose Beta Dose (rem)*	Thyroid (rem)
Control room airborne activity	0.114	5.73	19.28
External cloud shine	0.001	0	0
Containment shine	0	0	0
Floor, adjacent structures/shine	0.023	0	0
Ingress - Egress	0.042	0.097	1.18
Total	0.18	5.83	20.5

*Includes Occupancy Factor: 100 percent occupancy 0-24 hours
60 percent occupancy 1-4 days
40 percent occupancy 4-30 days

Acceptable Dose Limit: 5 rem Whole Body Gamma
30 rem Thyroid
30 rem Beta

7

2. TOXIC HAZARDS (OTHER THAN RADIATION)

- Analyses of Potential Hazards
  - RG 1.78
  - RG 1.95
  - NUREG-0737, Part III.D.3.4
  - High Temperature
- Identified Potential Hazards
  - High Temperature
  - Smoke
  - Chlorine
- Design Features for Detection of Hazards Identified
- Design Features for Protection of Personnel
  - High Temperature in Air Intake
  - Smoke in Air Intake
  - Chlorine in Air Intake

3. NATURAL HAZARDS

- **Seismic Qualification**
- **Tornado Analysis** RG 1.76
  - Pressure
  - Wind
  - Missiles
- **Flood**

9

4. ENVIRONMENTAL CONTROL

- Temperature Control Capability
    - Personnel Comfort
    - Equipment
- Slight Positive Pressure ($\approx$ .125" WG) Capability
- Isolation Capability for Accidents
- Air Cleanup Capability for Accidents

5. FIRE PROTECTION

- Use of Noncombustible Equipment
- Administrative Control Over the Use of Papers, Log Sheets, etc.
- Local Smoke Detectors
- Fire Dampers & Fire Doors
- Portable Fire Extinguishers
- Auxiliary Control Room

6. SYSTEM RELIABILITY

- ESF Components
  - Seismically Qualified
  - Environmentally Qualified
  - Class 1E Power Supplied
- Redundancy and Separation
  - Redundant Active Components
  - Separation of Active Components
  - Instrumentation and Controls
- Auxiliary Control Room Backup

CRITERIA APPLICABLE TO

AIR FILTRATION SYSTEMS

FOR CONTROL ROOMS

DR. RONALD R. BELLAMY

MAY 14, 1982

10 CFR Part 50, Appendix A
General Design Criterion 19

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

REGULATORY GUIDE 1.52, REVISION 2

DESIGN, TESTING AND MAINTENANCE CRITERIA FOR POST-ACCIDENT ESF ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS.

I. o INTRODUCTION REPEATS GDC 19 AS APPLICABLE, CLEARLY DEFINING THE CONTROL ROOM AIR FILTRATION SYSTEM AS ESF.

o ENVIRONMENTAL FACTORS DURING A DBA NEED TO BE CONSIDERED DURING THE DESIGN.

1. Δ P
2. DOSE RATE
3. RH
4. TEMPERATURE (MAXIMUM AND MINIMUM)
5. INTEGRATED RADIATION DOSE

o SYSTEM DESIGN CRITERIA

1. TYPICAL COMPONENTS

    DEMISTER
    HEATER
    PREFILTER
    HEPA
    CARBON ADSORBER
    HEPA
    FAN
    HOUSING

2. Redundant
3. Seismic Category I
4. Flow rate < 30,000 cfm per train
5. Instrumentation
   - Appropriate IEEE Criteria
   - Control Room Readout

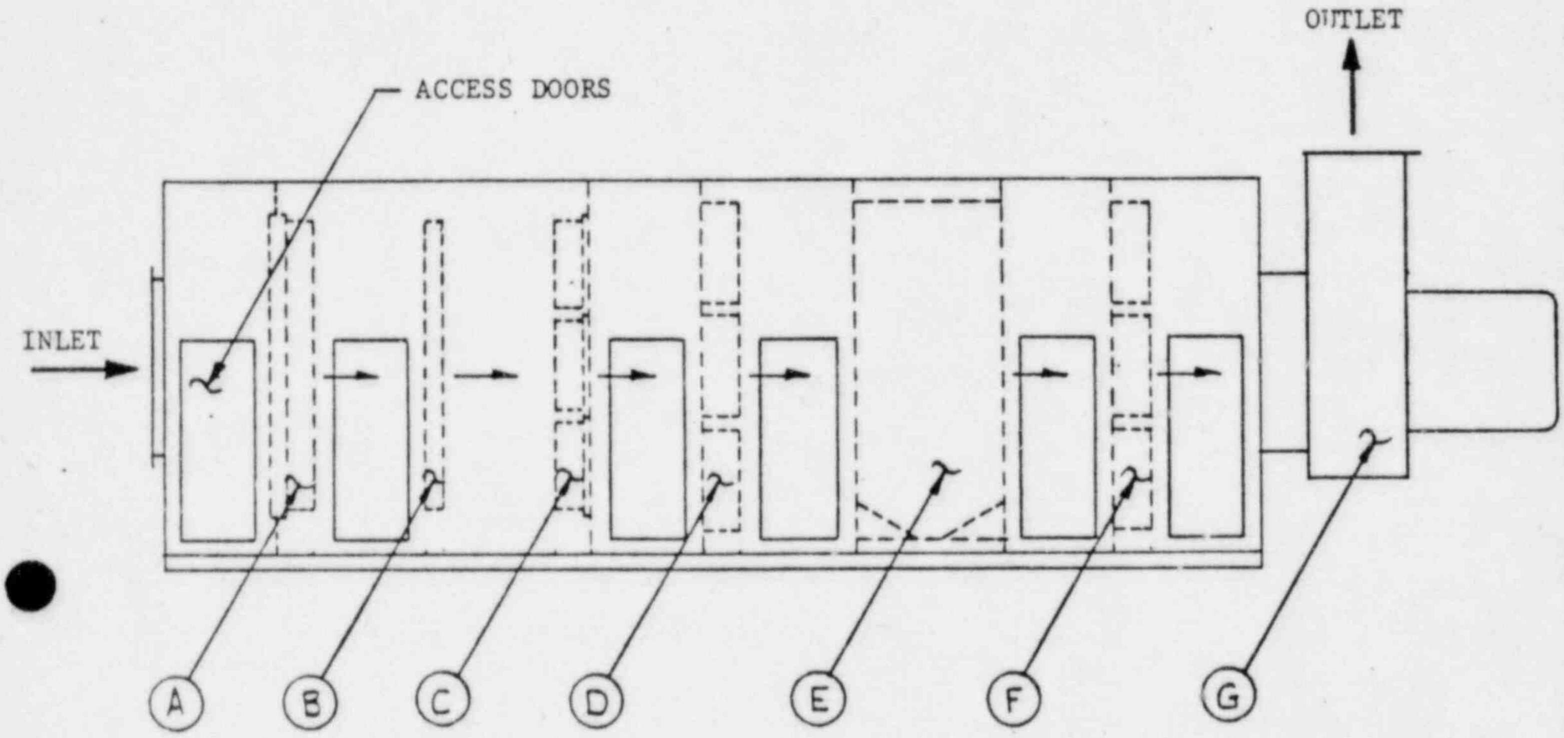
- Component Design Criteria and Qualification Testing
  1. ANSI N509 "Nuclear Power Plant Air Cleaning Units and Components" used as main reference.
  2. HEPA filters not sent to U.S.DOE filter test facilities.
  3. Activated carbon assumed as adsorbent.
- Maintenance
  1. Assessibility - three feet between components
  2. Permanent Test Probes
- In-Place Testing
  1. Visual
  2. Flow Distribution
  3. HEPA DOP Test-Leak Tightness > 99.95% warrants 99% particulate removal credit
  4. Carbon Freon Test-Leak Tightness > 99.95%
  5. Carbon Laboratory Testing-ANSI N509, bed depth determines decontamination efficiency
  6. Frequency

STANDARD REVIEW PLAN SECTION 6.5.1

- HIGHLIGHTS OF REGULATORY GUIDE 1.52
- BRANCH INTERFACES
- ASSIGNED DECONTAMINATION EFFICIENCIES ARE USED PER SRP 6.4

FIGURE 1-1

TYPICAL AIR CLEANING SYSTEM



Components

A Moisture Separator

B Electric Heater

C Roughing or Prefilter Bank

D HEPA Filter Bank

E Charcoal Adsorber Bed

F HEPA Filter Bank

G Fan/Motor

4.7.6.1.3 A Special Report shall be prepared and submitted to the Commission within 10 days if evidence of degradation is noted during an inspection. This report shall describe the extent and nature of the degradation and the plans and schedule for restoring the dike and errosion protection to a status equivalent to the original design provisions.

4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

4.7.7.1 The control room emergency air cleanup system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 100°F.

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c* and C.5.d* of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 14,350 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

   3. Verifying a system flow rate of 14,350 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

*The prerequisites of Section 10.3 and 12.3 of ANSI-N510-1975 do not apply.

CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filter and charcoal adsorber banks is less than 6 inches Water Guage while operating the system at a flow rate of 14,350 cfm ± 10%.

2. Verifying that on a control room air inlet radiation test signal or chlorine detection test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/10 inch W.G. relative to the outside atmosphere during system operation.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 14,350 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 14,350 cfm ± 10%.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room emergency ventilation systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4:

With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the control room emergency ventilation system in the recirculation mode.

b. With both control room emergency air ventilation systems inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

c. The provisions of Specification 3.0.3 are not applicable in MODE 6.

SURVEILLANCE REQUIREMENTS

4.7.7 Each control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 104°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 4000 cfm ± 10%.

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 4000 cfm ± 10%.

2. Verifying that on a safety injection signal or high radiation signal from the air intake stream, the system automatically diverts its inlet flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere at a system flow rate of 4000 cfm ± 10% (3800 cfm recirculation and 200 cfm fresh air).

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

OFFICE OF RESEARCH

SEVERE ACCIDENT RESEARCH PLAN

SECY-82-203 (May 19, 1982))

) = NUREG-0900 (DRAFT)

SECY-82-203A (August 1982))

RELATED ITEMS

SECY-82-1A, JULY 16, 1982

PROPOSED COMMISSION POLICY STATEMENT ON SEVERE ACCIDENTS AND RELATED VIEWS ON NUCLEAR REACTOR REGULATION

SAFETY GOAL AND ITS IMPLEMENTATION (NUREG-0880, ETC.)

STAFF REQUIREMENTS MEMO TO THE STAFF

1-19-82

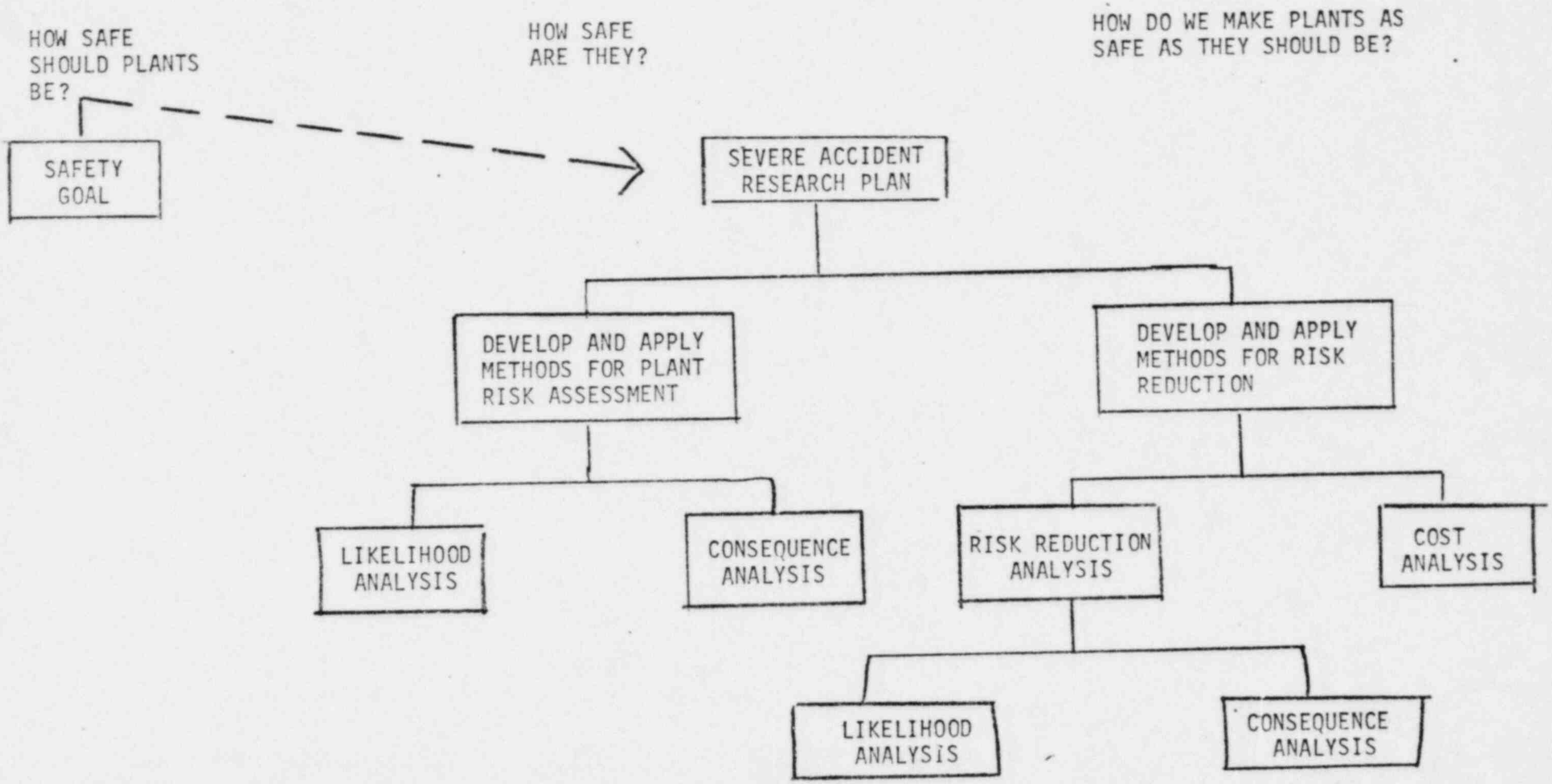
1. "THEY WILL ENSURE THAT IDCOR EFFORT CONTINUES."

2. "THEY WILL ENSURE THAT NRC RESEARCH AND OTHER PROGRAMS CRITICAL TO THIS APPROACH ARE CONTINUED."

GENERAL PURPOSE OF SARP:

TO DEVELOP GENERIC BASES TO DETERMINE HOW SAFE PLANTS ARE, AND WHERE AND HOW SAFETY OUGHT TO BE IMPROVED

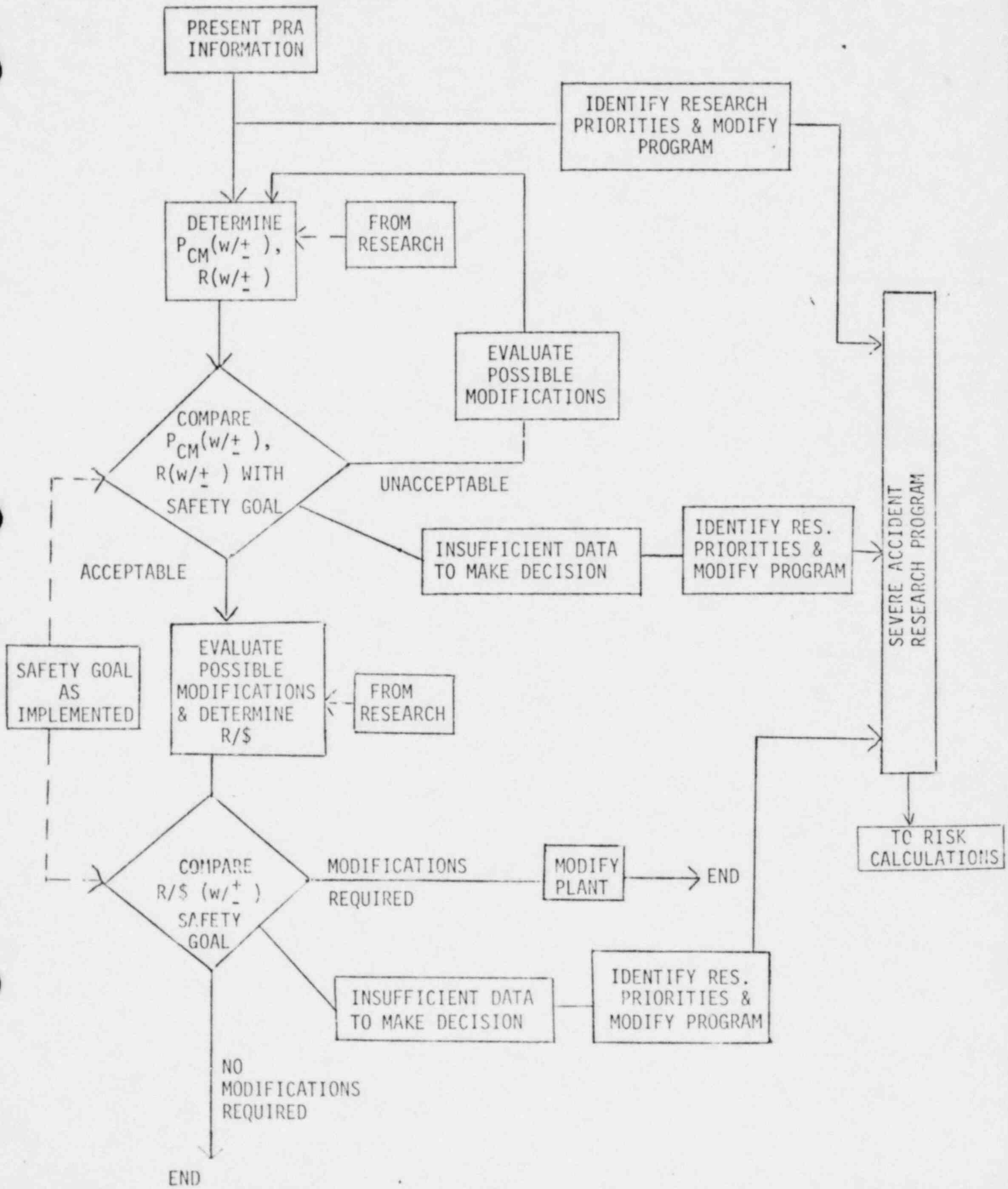
RESOLVING THE REGULATORY ISSUES RELATED TO SEVERE ACCIDENTS INVOLVES ANSWERING THREE FUNDAMENTAL QUESTIONS:



OBJECTIVES OF SARP

- USE COST-BENEFIT APPROACH, USING RISK-BASED ANALYSES, TO EXPLORE REGULATORY OPTIONS
- REDUCE UNCERTAINTIES
- INVOLVE QUESTIONS ON HOW SAFE PLANTS SHOULD BE, HOW SAFE ARE THEY, AND HOW COULD SAFETY BE IMPROVED
- DEVELOP METHODS AND DATA TO EVALUATE ACCIDENT SEQUENCE LIKELIHOOD
- DEVELOP BOTH DETAILED AND FAST-RUNNING METHODS
- DEVELOP COST-BENEFIT ALGORITHMS
- ASSESS CURRENT LEVEL OF RISK
- EVALUATE RISK-REDUCTION POTENTIAL OF VARIOUS DEVICES

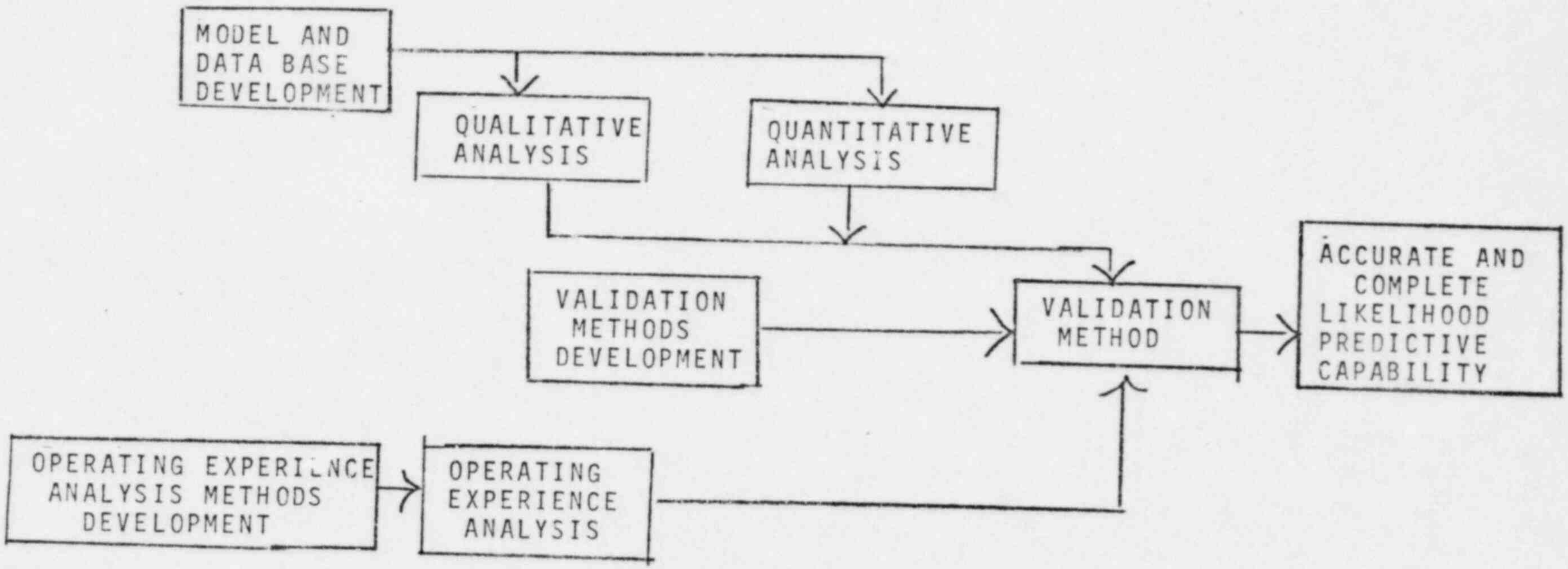
DECISION ANALYSIS PROCESS



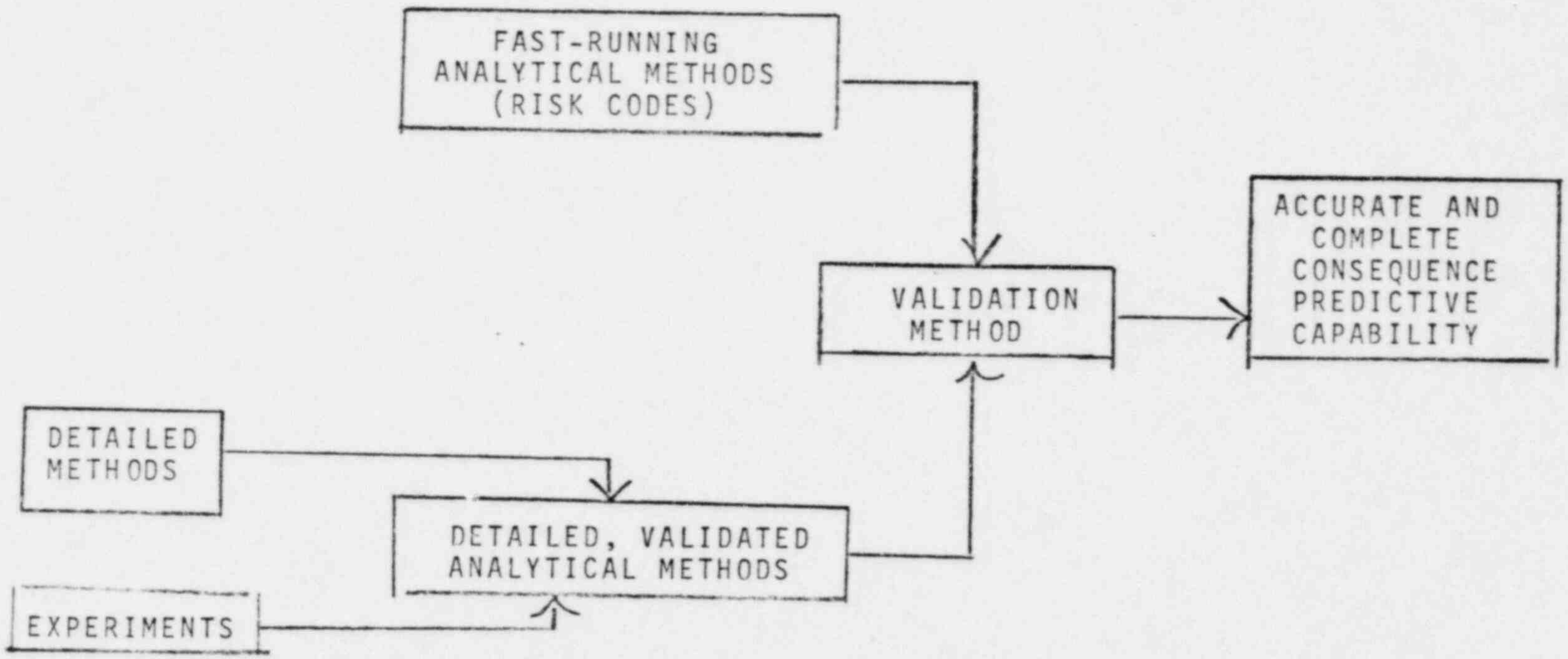
FOR THIS PROCESS TO WORK, RESEARCH AND ANALYSIS ARE NEEDED TO:

- SUPPORT A WORKABLE SAFETY GOAL
- PROVIDE REASONABLY ACCURATE AND COMPLETE PRA CALCULATIONS OF:
    - PRESENT LEVEL OF SAFETY
    - COST-EFFECTIVE RISK REDUCTION
- DEVELOP AND APPLY METHODS FOR DECISION-MAKING IN FACE OF UNCERTAINTIES

ACCURATE AND COMPLETE PRA LIKELIHOOD CALCULATIONS REQUIRE:



ACCURATE AND COMPLETE PRA <u>CONSEQUENCE</u> CALCULATIONS REQUIRE:



OUTLINE

SEVERE ACCIDENT RESEARCH PLAN

- INTRODUCTION
  - -- OBJECTIVES
  - -- BACKGROUND
- INFORMATION NEEDS AND REGULATORY ISSUES
- STATE OF THE ART
- PROGRAM LOGIC, SCHEDULE, AND INTERFACES
- PROGRAM ELEMENTS
  - -- DESCRIPTION
  - -- PLAN OF WORK

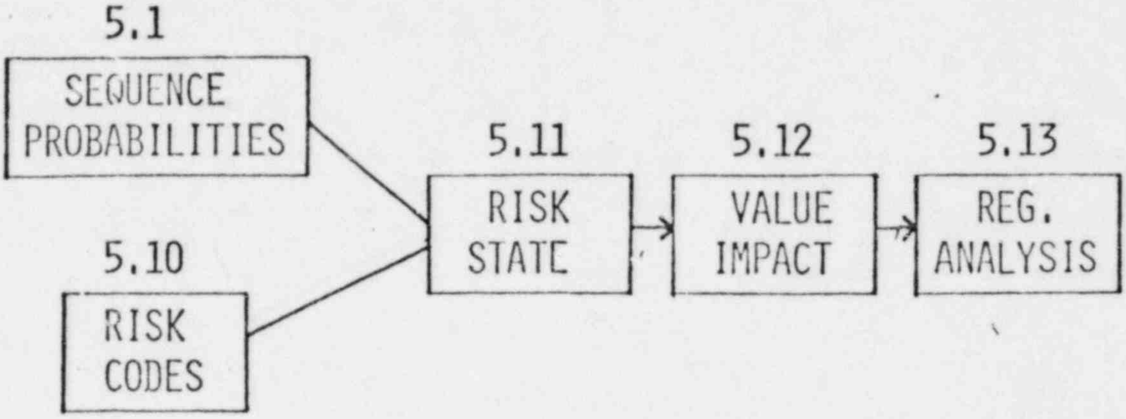
FORMAT OF CHAPTER 5

- ELEMENT DESCRIPTION
- TECHNICAL ISSUES RESOLVED BY THIS ELEMENT
- KEY INTERFACES WITH OTHER ELEMENTS
- BACKGROUND & STATUS
- PLAN OF WORK AS A FUNCTION OF TIME

CHAPTER 5
PROGRAM ELEMENTS

5.1 ACCIDENT LIKELIHOOD ANALYSIS
5.2 SEVERE ACCIDENT SEQUENCE ANALYSIS
5.3 ACCIDENT MANAGEMENT
5.4 BEHAVIOR OF DAMAGED FUEL
5.5 HYDROGEN GENERATION AND CONTROL
5.6 FUEL-STRUCTURE INTERACTION
5.7 CONTAINMENT ANALYSIS
5.8 CONTAINMENT FAILURE MODE
5.9 FISSION PRODUCT RELEASE AND TRANSPORT
5.10 RISK CODE DEVELOPMENT
5.11 ACCIDENT CONSEQUENCE AND RISK REEVALUATION
5.12 RISK REDUCTION AND COST ANALYSIS
5.13 REGULATORY ANALYSIS AND STANDARDS DEVELOPMENT

LOGIC FOR RISK ASSESSMENT SECTIONS



PRODUCTS FOR EACH CLASS OF PLANT

- CATALOG OF DOMINANT SEQUENCES
- CONSEQUENCE STATEMENT
- RISK STATEMENT
- VALUE-IMPACT REPORT

$\left.\begin{array}{l}\text{CONSEQUENCE STATEMENT}\\ \text{RISK STATEMENT}\end{array}\right\}$ RANGE OF SITES

SECTION 5.1

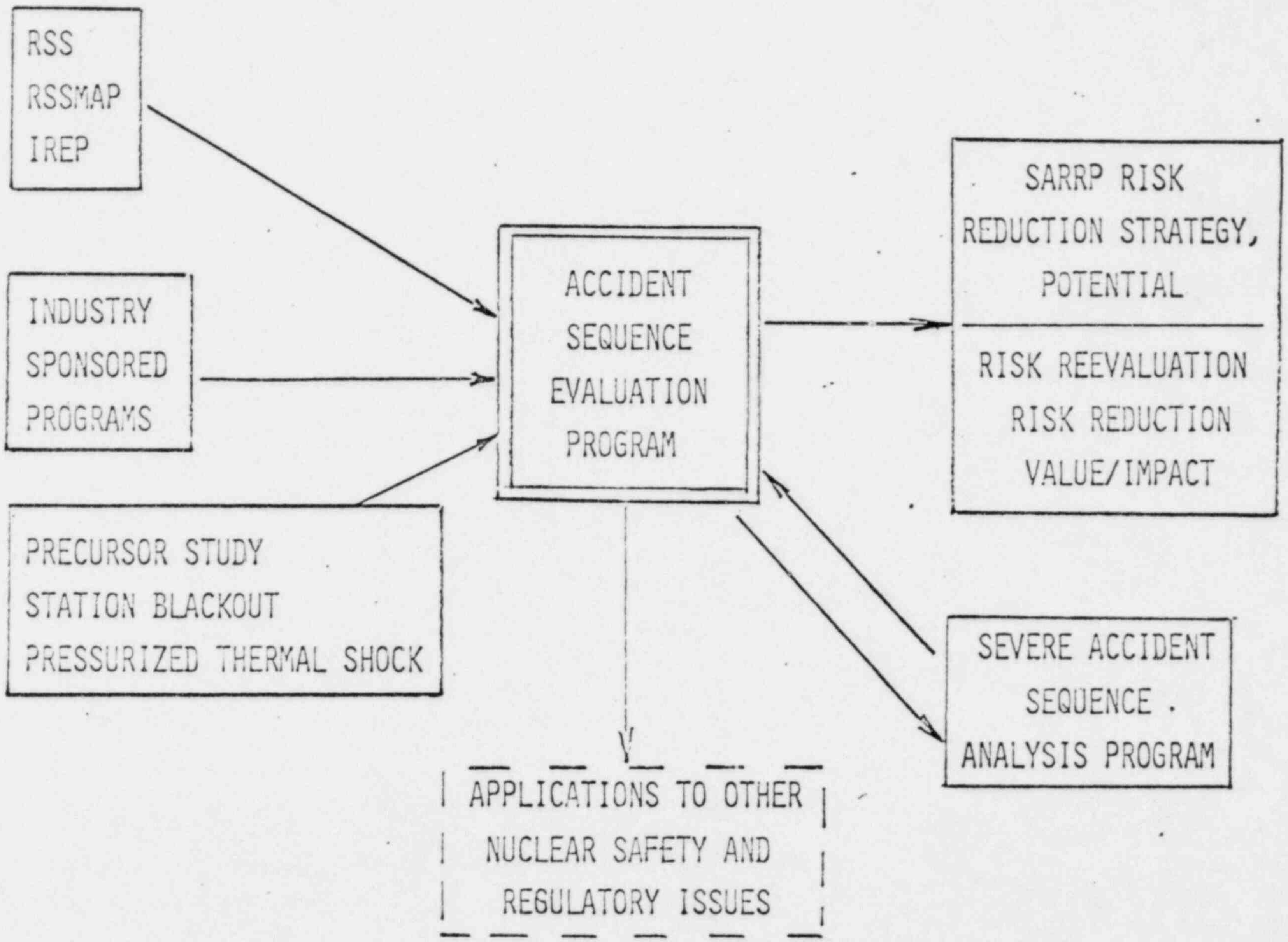
ACCIDENT SEQUENCE EVALUATION PROGRAM

OBJECTIVE

PROVIDE A COMPREHENSIVE AND GENERIC SET OF LWR ACCIDENT SEQUENCES (AND ASSESS THEIR LIKELIHOOD) WHICH WILL BE USED IN THE SEVERE ACCIDENT RISK REDUCTION PROGRAM (SARRP) AND IN THE EVALUATION OF OTHER NUCLEAR SAFETY AND REGULATORY ISSUES

THE ACCIDENT SEQUENCES WILL BE SUITABLY DELINEATED FOR USE IN THE EVALUATION OF CORE DAMAGE/MELT PREVENTION AND MITIGATION CONCEPTS, AND FOR PERFORMING VALUE-IMPACT ASSESSMENTS

PROGRAM INTERRELATIONSHIPS



ELEMENTS OF ASEP IN SUPPORT OF SARRP

- COMPILE AND EVALUATE ACCIDENT SEQUENCE LIKELIHOOD INFORMATION FROM EXISTING PRAs

- REVIEW OPERATING EXPERIENCE INFORMATION TO UPDATE CURRENT PRA ESTIMATES WHERE APPROPRIATE

- PERFORM SENSITIVITY ANALYSES AND SPECIAL EVALUATIONS CONSIDERING POTENTIAL COMMON CAUSE FAILURES AND SYSTEMS INTERACTIONS, SEISMIC EVENTS, AND SABOTAGE

- FORMULATE A GENERALIZED ACCIDENT SEQUENCE DELINEATION SCHEME FOR TRIAL USE IN RISK REDUCTION ANALYSES

- CONDUCT REVIEW BY MULTIDISCIPLINARY TEAM OF EXPERTS TO DETERMINE ASEP COMPLETENESS AND ACCURACY

- PREPARE FINAL ACCIDENT SEQUENCE DELINEATION REPORT FOR USE IN SARRP

SECTION 5.10

DEVELOPMENT OF IMPROVED PHYSICAL PROCESS COMPUTER CODES FOR RISK ASSESSMENT

PRINCIPAL PROGRAM OBJECTIVES:

1. TO PROVIDE SHORT-TERM UPGRADED VERSIONS OF THE MARCH, CORRAL (MATADOR) AND CRAC RISK ASSESSMENT CODES TO ACCOUNT FOR SPECIFIC AND IMPROVED DEFICIENCIES.

2. TO PROVIDE A LONGER-TERM SERIES OF RISK CODES (MELCOR) TO REPLACE THE MARCH, MATADOR AND CRAC CODES WHICH (I) HAVE A STRUCTURE READILY AMENABLE TO INCORPORATION OF NEW MODELS BASED ON THE ONGOING EXPERIMENTAL RESEARCH PROGRAM, AND (II) PERMIT THE QUANTITATIVE ANALYSIS OF BOTH "BEST ESTIMATE" SEVERE ACCIDENT CONSEQUENCES AND THE ASSOCIATED UNCERTAINTIES.

DEVELOPMENT OF IMPROVED PHYSICAL PROCESS COMPUTER

CODES FOR RISK ASSESSMENT

PRINCIPAL PROGRAM TASKS:

1. DEVELOPMENT, EVALUATION AND DOCUMENTATION OF MARCH-2 AND MATADOR COMPUTER CODES.

2. PLANNING PHASE FOR MELCOR CODE DEVELOPMENT (REVIEW OF EXISTING INFORMATION AND RECOMMENDATIONS FOR APPROACH).

3. FIRST-LEVEL MELCOR DEVELOPMENT, EVALUATION, AND DOCUMENTATION.

4. SENSITIVITY STUDIES USING FIRST-LEVEL CODE VERSION.

5. SECOND-LEVEL MELCOR DEVELOPMENT, EVALUATION AND DOCUMENTATION.

6. UNCERTAINTY ANALYSES.

INTERIM CODE DEVELOPMENT

<u>OBJECTIVE</u>: PROVIDE UPGRADED VERSIONS OF MARCH AND CORRAL (MATADOR) BY FALL, 1982, FOR USE IN SARP AND OTHER NRC PROGRAMS.

MARCH

- COLLECT AND EVALUATE MODIFICATIONS BY

  BATTELLE COLUMBUS
  SANDIA
  BROOKHAVEN
  OAK RIDGE
  TVA

- INCORPORATE MODIFICATIONS
- VALIDATION CALCULATIONS
- DOCUMENTATION
- SAMPLE PROBLEMS

MATADOR

- MODIFICATIONS IN RESPONSE TO PEER REVIEW
- VALIDATION CALCULATIONS
- DOCUMENTATION
- SAMPLE PROBLEMS

SECTIONS 5.11 AND 5.12

SEVERE ACCIDENT RISK RESEARCH PROGRAM

GENERAL OBJECTIVES

TO PROVIDE PERIODIC REASSESSMENTS OF:

- THE PREDICTED LEVEL OF RISK FOR A SPECTRUM OF MAJOR LWR DESIGN TYPES; AND
- THE RISK REDUCTION VALUE AND COSTS OF A SPECTRUM OF POSSIBLE DESIGN MODIFICATIONS FOR PREVENTION AND MITIGATION OF SEVERE ACCIDENTS

SEVERE ACCIDENT RISK RESEARCH PROGRAM

SECTION 5.11
RISK BENCHMARKING

INTEGRATION OF

- ACCIDENT LIKELIHOOD REANALYSIS (ASEP) AND
- ACCIDENT CONSEQUENCES REANALYSIS USING (SUCCESSIVELY):
  - MARCH-2/MATADOR
  - MELCOR MOD 1
  - MELCOR MOD 2

TO YIELD:

- PERIODIC "BENCHMARKING" OF PREVIOUSLY ASSESSED PRAs, E.G.,
  - RSS
  - RSSMAPs

SEVERE ACCIDENT RISK RESEARCH PROGRAM

SECTION 5.12

RISK REDUCTION BENEFIT/COST ANALYSIS

FOR: A SET OF LWR DESIGN TYPES, AND
A SPECTRUM OF POSSIBLE PLANT MODIFICATIONS TO PREVENT OR MITIGATE SEVERE ACCIDENTS

ANALYZE: THE REDUCTION IN CORE MELT PROBABILITY AND RISK BY THE ADDITION OF A MODIFICATION (OR COMBINATIONS), AND THE COST OF INSTALLING SUCH A MODIFICATION

YIELDING: RELATIVE MERIT OF POSSIBLE MODIFICATIONS

PREDECESSORS: FVCS, ADHR, MCRD PROGRAMS

CANDIDATE IMPROVEMENTS	OPTIONS
ADDITIONAL CONTAINMENT HEAT REMOVAL	ACTIVE VERSUS PASSIVE
CONTAINMENT ATMOSPHERE PARTICULATE CAPTURE	
CONTAINMENT ATMOSPHERE MASS REMOVAL	FILTERED VERSUS UNFILTERED LOW FLOW VERSUS HIGH FLOW
INCREASED CONTAINMENT MARGINS	INCREASED VOLUME INCREASED PRESSURE CAPABILITY PRESSURE SUPPRESSION FEATURES
COMBUSTIBLE GAS CONTROL	DELIBERATE IGNITION INERTING (PRIOR/POST ACCIDENT) FIRE SUPPRESSION (HALON/WATER FOGS)
CORE RETENTION DEVICES	DRY VERSUS WET ACTIVE VERSUS PASSIVE COOLING OR NO COOLING
MISSILE SHIELDS	IN-VESSEL STEAM EXPLOSIONS VESSEL THERMAL SHOCK VESSEL MELT-THROUGH AT HIGH PRESSURE EX-VESSEL STEAM EXPLOSIONS COMBUSTIBLE GAS EXPLOSIONS
BWR CONTAINMENT SPRAY SYSTEM	
PWR PRIMARY SYSTEM DEPRESSURIZATION	AUTOMATIC VERSUS MANUAL ADDITIONAL RELIEF CAPACITY PRESSURE SUPPRESSION FEATURES RADIOACTIVITY REMOVAL SYSTEMS
ADD-ON DECAY HEAT REMOVAL SYSTEMS	HIGH PRESSURE VERSUS LOW PRESSURE OPEN LOOP VERSUS CLOSED LOOP PRIMARY SYSTEM VERSUS SECONDARY SYSTEM
SPECIFIC PREVENTION CONCEPTS	IMPROVED DRAIN OR VALVE DESIGN IMPROVED MAINTENANCE PROCEDURES IMPROVED CONTROL LOGIC REDUCTION OF COMMON MODE DEPENDENCIES

PRELIMINARY RESULTS OF RISK REDUCTION ANALYSIS

CANDIDATE SAFETY APPROACH DESCRIPTION	FACTOR REDUCTION IN BWR CORE MELT RELEASE CATEGORY FREQUENCY				FACTOR REDUCTION IN TOTAL CORE MELT FREQUENCY	FACTOR REDUCTION IN RISK MEASURES		
	1	2	3	4		EAR	LAT	POP
1) HIGH VOLUME UN-FILTERED CONTAINMENT VENT (.01)	13	100	1.0	1.0	13	86	34	43
2) CONTAINMENT HEAT REMOVAL SYSTEM, NOT SIZED FOR ATWS (.01)					4.5			5.5
3) LOW VOLUME UN-FILTERED CONTAINMENT VENT (.01)					4.4			5.5
4) INCREASE CONTAIN-MENT DESIGN PRESSURE BY 100%					2.0			2.2
5) INCREASE RELIA-BILITY OF EMERGENCY AC POWER SYSTEM(.01)					1.4			1.3
6) INCREASE SAFETY/RELIEF VALVE RELIA-BILITY (.01)					1.2			1.2
7) INCREASE RELIA-BILITY OF RPS SYSTEM (0.1)					1.2			1.2

ELEMENT 5.2

SCOPE OF SASA PROGRAM

- o SEVERE ACCIDENT ANALYSIS FOR SPECIFIC PLANT DESIGNS
- o SAFETY CONCERNS GENERATED BY NRR OPERATOR GUIDELINES REVIEW
- o IREP DOMINANT SEQUENCE ANALYSIS
- o NRC UNRESOLVED SAFETY ISSUES
- o OPERATOR INSTRUMENTATION INFORMATION NEEDS
- o SANDIA NATIONAL LAB - PWR CONTAINMENT MANAGEMENT STUDY
- o LOS ALAMOS NATIONAL LAB - SEVERE ACCIDENT ANALYSES FOR THE OCONNE PLANT, A B&W PLANT - ANALYSIS OF DECAY HEAT REMOVAL USING FEED AND BLEED TECHNIQUES
- o OAK RIDGE NATIONAL LAB - ANALYSIS OF BWR DOMINANT SEQUENCES FOR BROWNS FERRY UNIT ONE (MARK I) AND LIMERICK (MARK II)
- o IDAHO NATIONAL ENGINEERING LAB - IN DEPTH SEVERE ACCIDENT ANALYSIS OF CE CESSAR-80 PLANT DESIGN

LOSS-OF-FEEDWATER TRANSIENTS AT ZION

PURPOSE: TO STUDY PLANT RESPONSE TO EQUIPMENT FAILURES AND OPERATOR ACTIONS DURING THE ACCIDENT.

SEQUENCE:

- o LOSS OF MAIN FEEDWATER FROM LOSS OF OFFSITE POWER, TURBINE TRIP, SYSTEM MALFUNCTION OR MECHANICAL FAILURES
- o FAILURE OF AFWS FROM CLOSED PUMP VALVES, BREAKS IN HEADER FAILURE OF TWO LOOPS W/ONE DOWN FOR MAINTENANCE OR LACK OF FLOW FROM PLUGGED VENTS

RESULTS:

MINIMUM CRITICAL SYSTEMS FOR RECOVERY

- o 15% AFW FLOW
- o 70% ECCS FLOW

ACCIDENT MANAGEMENT STRATEGIES

- o EARLY ECCS INITIATION
- o FEED AND BLEED RECOVERY WITH ECCS AND PORVs.
- o PRIMARY DEPRESSURIZATION USING ARVs RECOVERS IN 33 MIN.

REPORT PUBLISHED MAY 1982 NUREG/CR 2656

STATION BLACKOUT AT BROWN'S FERRY UNIT ONE

PURPOSE: TO ASSSIT IN THE RESOLUTION OF THE STATION BLACKOUT UNRESOLVED SAFETY ISSUE

SEQUENCE:
- o LOSS OF OFFSITE POWER
- o REACTOR SCRAM DUE TO TURBINE CONTROL VALVE FAST CLOSURE
- o MSIV CLOSURE
- o FAILURE OF DIESELS TO START AND LOAD

RESULTS:
- o OPERATOR CONTROLS PRESSURE BY REMOTE-MANUAL RELIEF VALVE ACTUATION (1075-900 PSIG)
  - TO DISTRIBUTE RELIEF VALVE DISCHARGE AROUND CIRCUMFERENCE OF PRESSURE SUPPRESSION POOL TO PREVENT LOCALIZED BOILING AND EVENTUAL CONTAINMENT FAILURE
  - TO REDUCE TOTAL NUMBER OF ACTUATIONS
- o OPERATOR SHOULD BEGIN DEPRESSURIZATION TO $\sim$100 PSIG WITHIN ONE HOUR
  - KEEP DRYWELL TEMPERATURE BELOW DESIGN LIMITS
  - LOW ENOUGH TO REDUCE RV SURFACE TEMPERATURE
  - HIGH ENOUGH FOR RCIC OPERATION
  - TIME FROM BATTERY EXHAUSTION TO CORE UNCOVERY INCREASES TO 3.8 HOURS
- o RV LEVEL AND PRESSURE CONTROL AND ADEQUATE INSTRUMENTATION AVAILABLE WHILE DC POWER REMAINS

REPORT: PUBLISHED AND DISTRIBUTED NOVEMBER 1981 NUREG/CR-2132

ELEMENT 5.3

PLAN OF WORK AS A FUNCTION OF TIME

THIS ELEMENT TIES TOGETHER SEVERAL OTHER RESEARCH ELEMENTS. WITHIN 2 YEARS, THERE WILL BE SUFFICIENT INPUT FROM THE OTHER ELEMENTS TO PROVIDE A PHASE 1 ACCIDENT MANAGEMENT REPORT. THIS INCLUDES:

1. IDENTIFICATION OF OPERATOR ERROR RATES.

2. ESTABLISHMENT OF MMI REQUIREMENTS FOR ACCIDENT MANAGEMENT.

3. DEVELOPMENT OF OPERATOR PROCEDURES FOR RECOVERY FROM POTENTIALLY DEGRADED COOLING ACCIDENTS.

4. ASSESSMENT OF RESPONSE OF CONTAINMENT AND ESF'S TO SEVERE ACCIDENT ENVIRONMENT.

5. DEVELOPMENT OF OPERATING PROCEDURES FOR ASSURING CONTAINMENT INTEGRITY DURING SEVERE ACCIDENTS.

ELEMENT 5.4

BEHAVIOR OF DAMAGED FUEL

WHAT QUESTIONS WILL IT ANSWER?

1. HOW ACCURATE/APPLICABLE ARE THE CURRENT FISSION PRODUCT SOURCE TERM DATA BASE/MODELS FOR DESCRIBING REALISTIC ACCIDENT BEHAVIOR?

2. HOW GOOD ARE OUR CURRENT ESTIMATES OF HYDROGEN SOURCE TERM AND TIMING OF RELEASE?

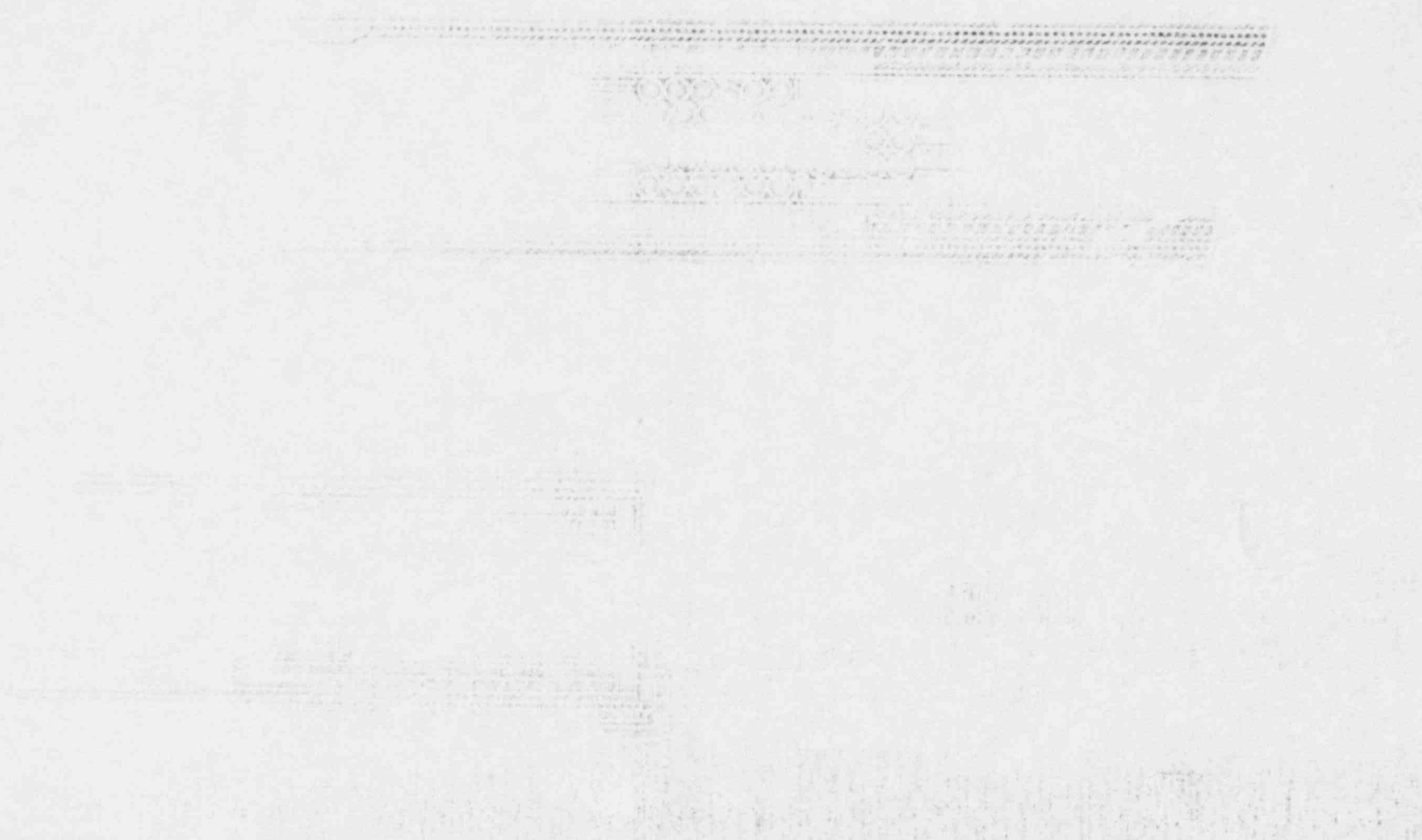
3. WHAT IS THE MAGNITUDE AND TIMING OF FUEL-BEHAVIOR-INDUCED LOADS ON CONTAINMENT?

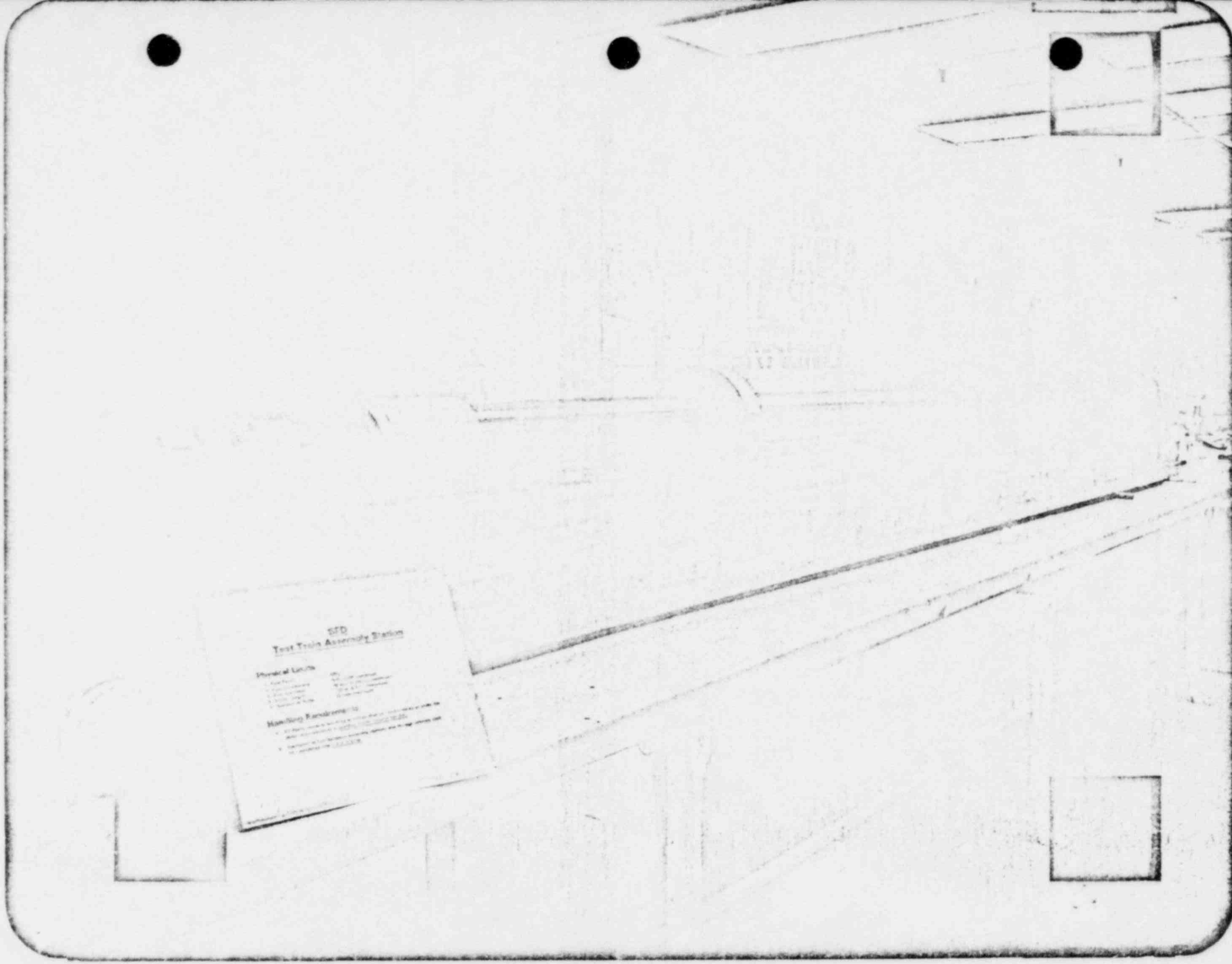
4. UNDER WHAT CONDITIONS IS DAMAGED CORE DEBRIS COOLABLE?

   - POST-ACCIDENT RECOVERY
   - LONG-TERM EX-VESSEL COOLABILITY

USE OF THREE IN-PILE FACILITIES: POWER BURST FACILITY (PBF), ANNULAR CORE RESEARCH REACTOR (ACRR), AND NATIONAL REACTOR UNIVERSAL (NRU).

SFD Test Train Schematic




SFD
Test Train Assembly Station
Physical Limits
Handling Requirements

33

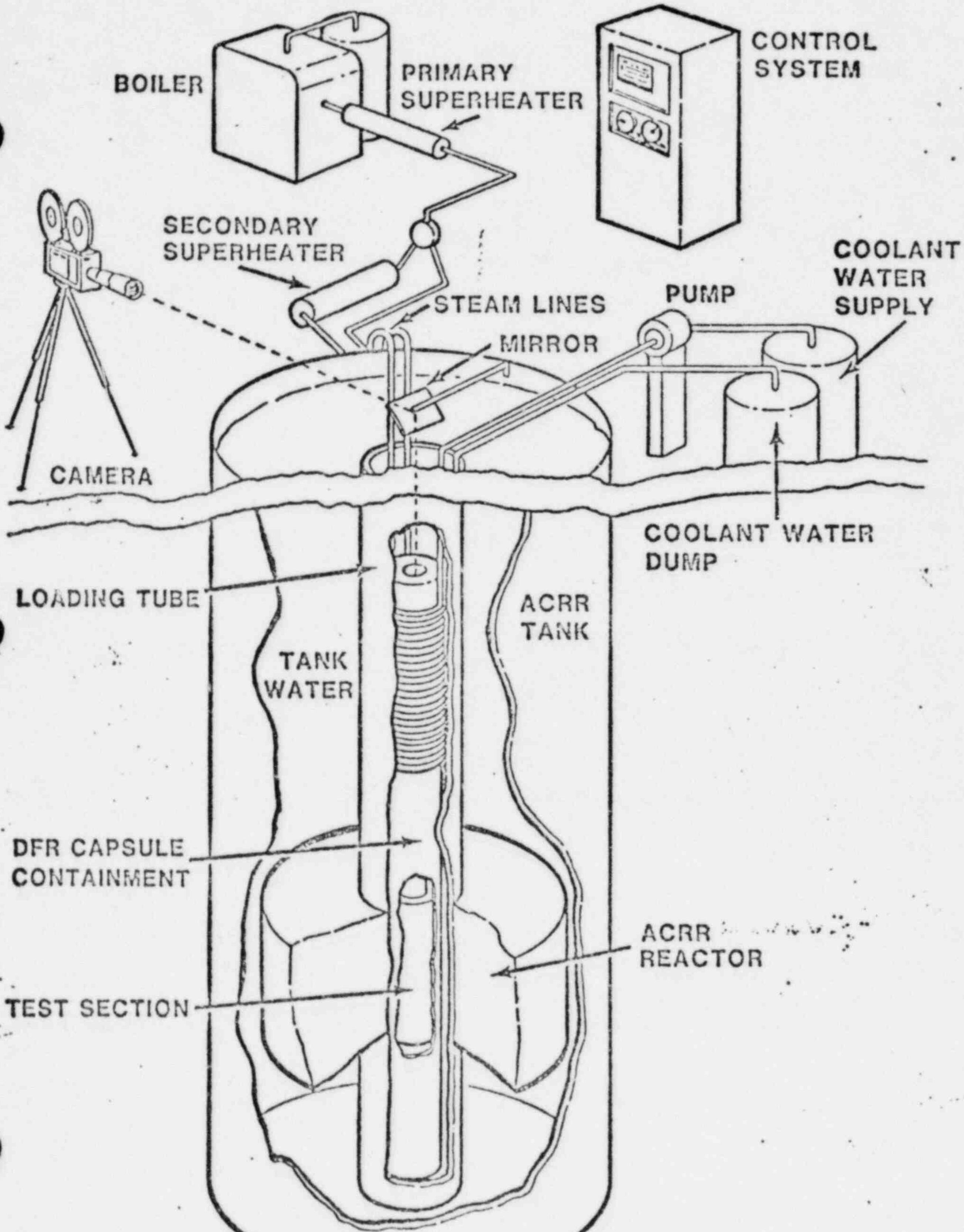
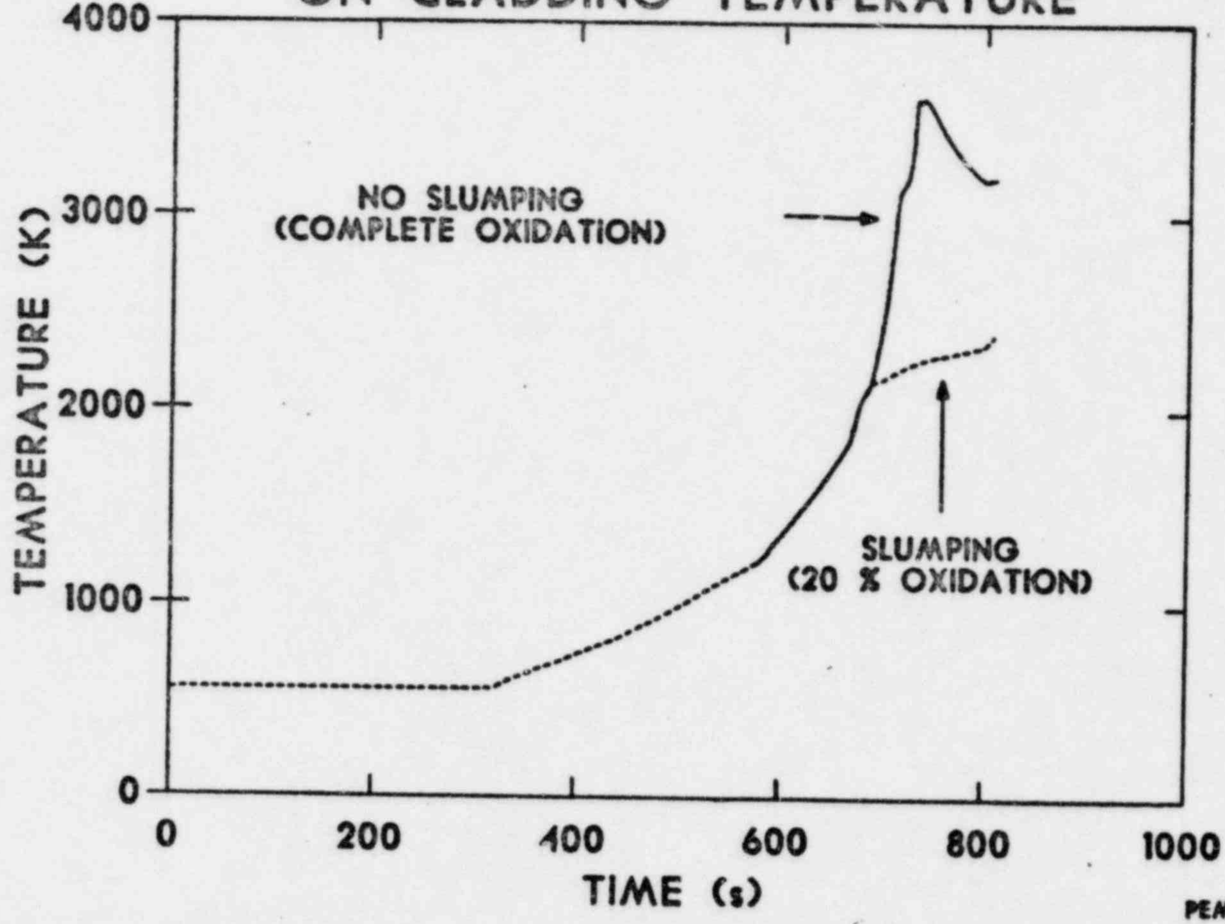


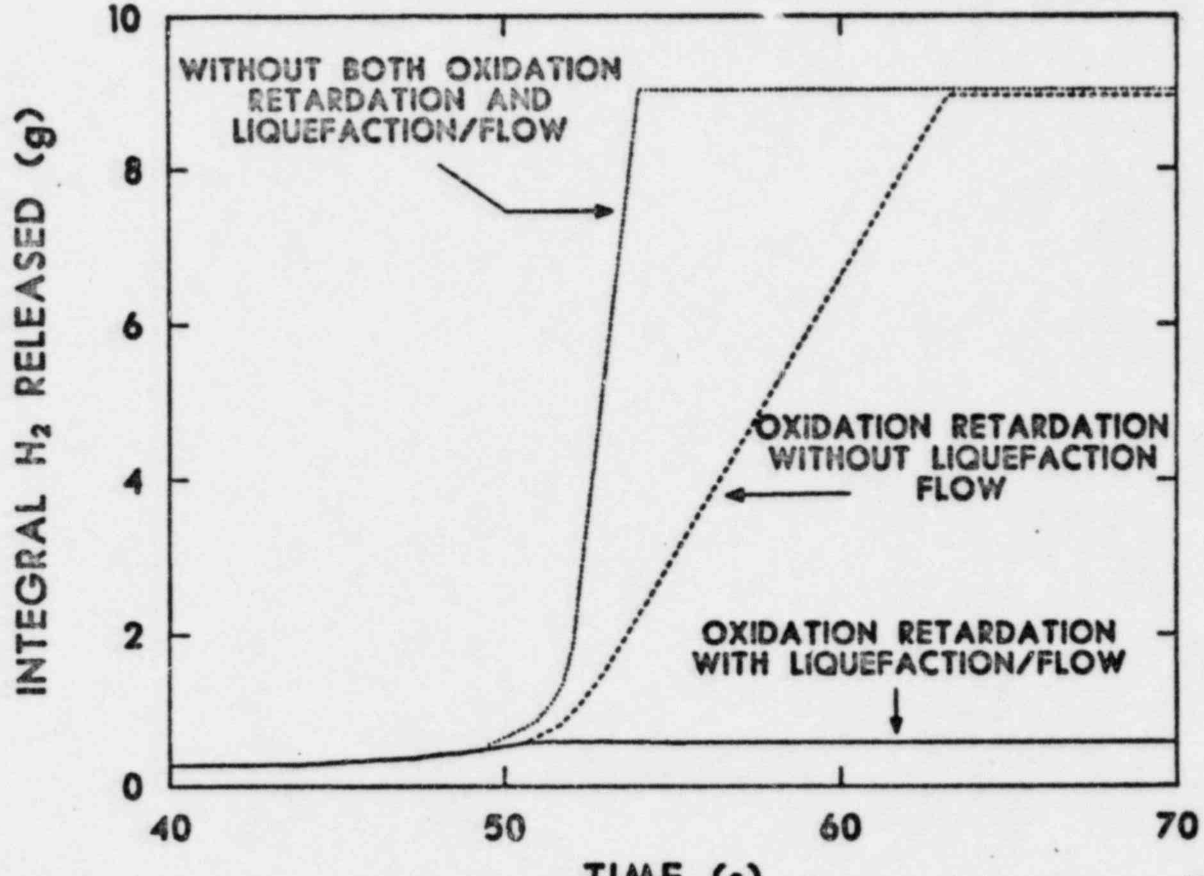
Figure 4-3. DFR In-pile Experiment Apparatus

EFFECT OF CLADDING SLUMPING ON CLADDING TEMPERATURE



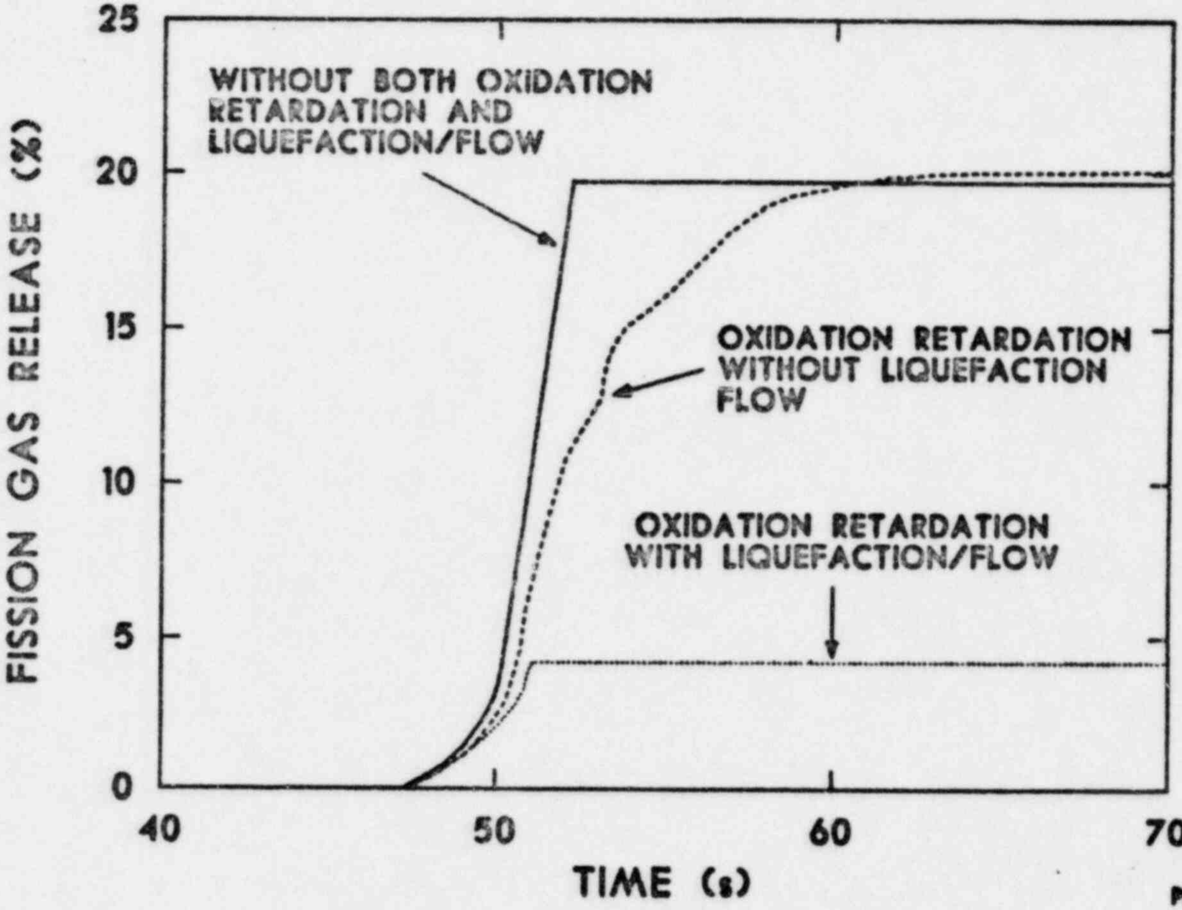
PEM-582-009

INTEGRAL HYDROGEN RELEASE



PEM-582-018

FISSION GAS RELEASE



PEM-582-020

ELEMENT 5.5

HYDROGEN GENERATION AND CONTROL

THE PROBLEM:

HYDROGEN GENERATED DURING HYPOTHETICAL LWR ACCIDENTS CAN UNDERGO COMBUSTION (DEFLAGRATION OR DETONATION) POSSIBLY RESULTING IN CONTAINMENT FAILURE OR DAMAGE TO SAFETY EQUIPMENT.

THE SOLUTION:

DETERMINE THE NATURE AND SEVERITY OF THIS THREAT. AS NECESSARY, ASSESS HYDROGEN CONTROL AND DISPOSAL METHODS AND MEANS FOR MITIGATING COMBUSTION-RELATED DAMAGE.

THE TOOLS:

THEORETICAL AND EXPERIMENTAL RESEARCH ENCOMPASSING:

- o LWR ACCIDENT ANALYSES
- o HYDROGEN TRANSPORT AND COMBUSTION ANALYSES
- o SAFETY EQUIPMENT TESTING AND MODEL DEVELOPMENT

LABORATORY-SCALE TESTS, INTERMEDIATE, AND LARGE-SCALE EXPERIMENTS (STEEL TANKS (VEGES, FITS), FLAME JETS, MCGILL UNIVERSITY EXPERIMENTS, PLASTIC BAGS, FLAME FACILITY)

HECTR CODE DEVELOPMENT AND APPLICATION

FY82 ACCOMPLISHMENTS

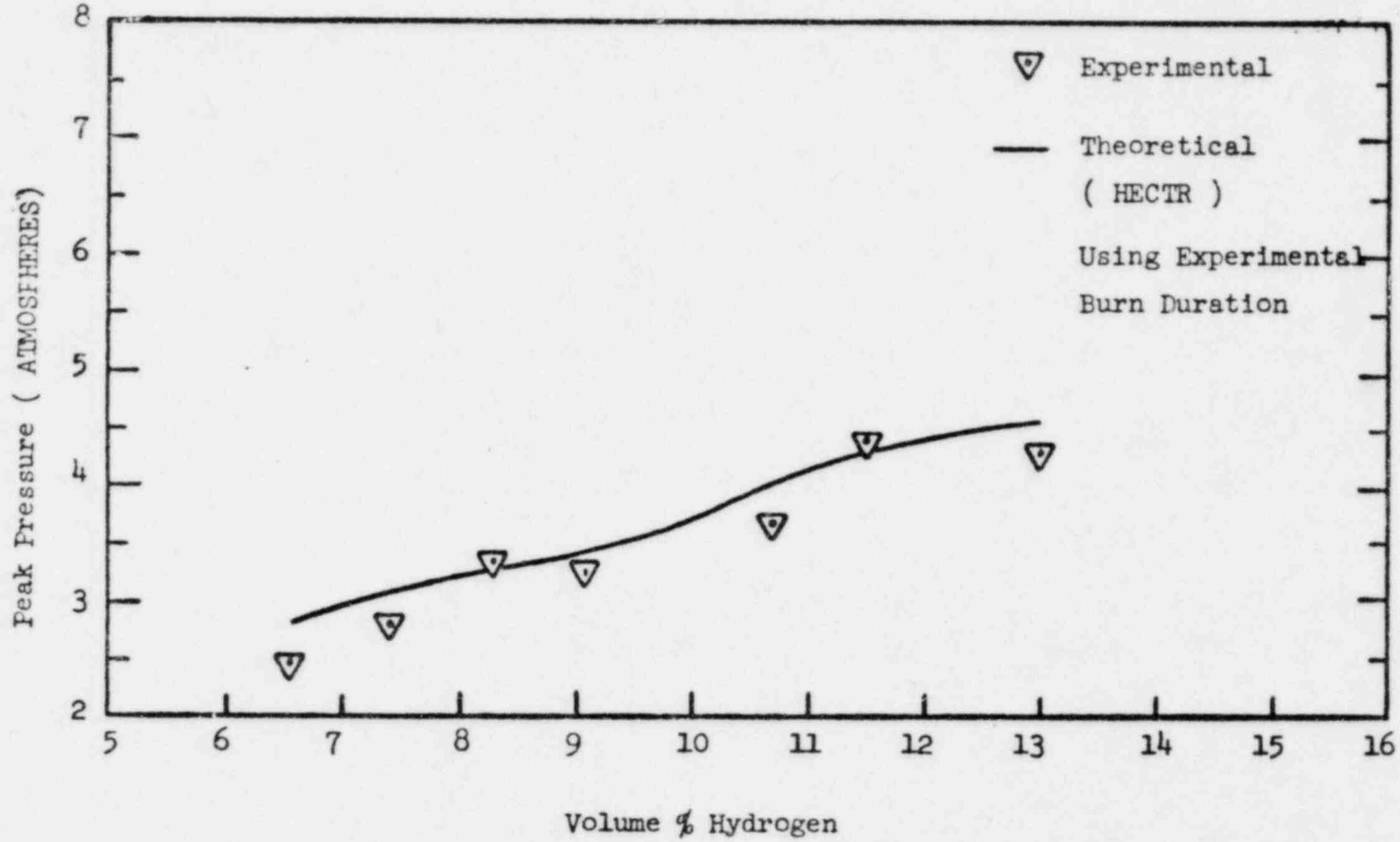
COMPLETED:

- VERSION 1 OF HECTR (HYDROGEN EVENT:CONTAINMENT TRANSIENT RESPONSE), INCLUDING MODELS FOR HYDROGEN BURNS, RADIATION, CONVECTION, AND SPRAYS
- ANALYSIS OF THE GRAND GULF MARK III BWR USING HECTR
- CALCULATIONS IN SUPPORT OF THE HBS PROGRAM AND VARIOUS EXPERIMENTAL TESTS

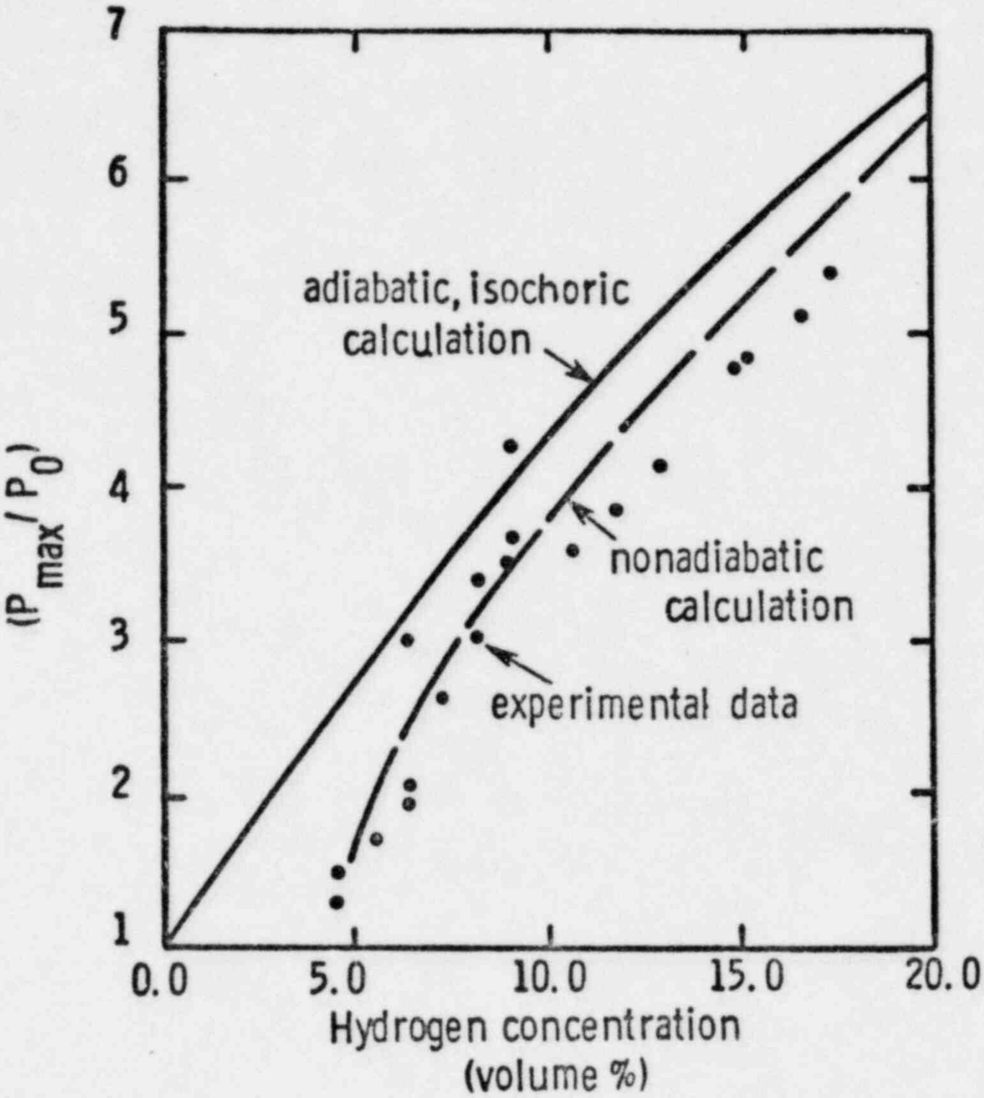
PLANNED:

- DEVELOP ADDITIONAL MODELS FOR CONVECTIVE MIXING, ICE CONDENSERS, FAN COOLERS, AND SUPPRESSION POOLS
- EVALUATE THE SEQUOYAH ICE CONDENSER PLANT AND THE ZION LARGE DRY PWR CONTAINMENT IN SUPPORT OF THE HBS AND SASA PROGRAMS
- INITIATE CODE CLEANUP, ASSESSMENT, AND SENSITIVITY STUDIES
- PRODUCE A DRAFT REPORT

Peak Pressure (Atmospheres) versus Volume % Hydrogen : VGES/HECTR Analysis



Peak Pressure versus Volume % Hydrogen for experimental (VGES) and calculated (HECTR) values.



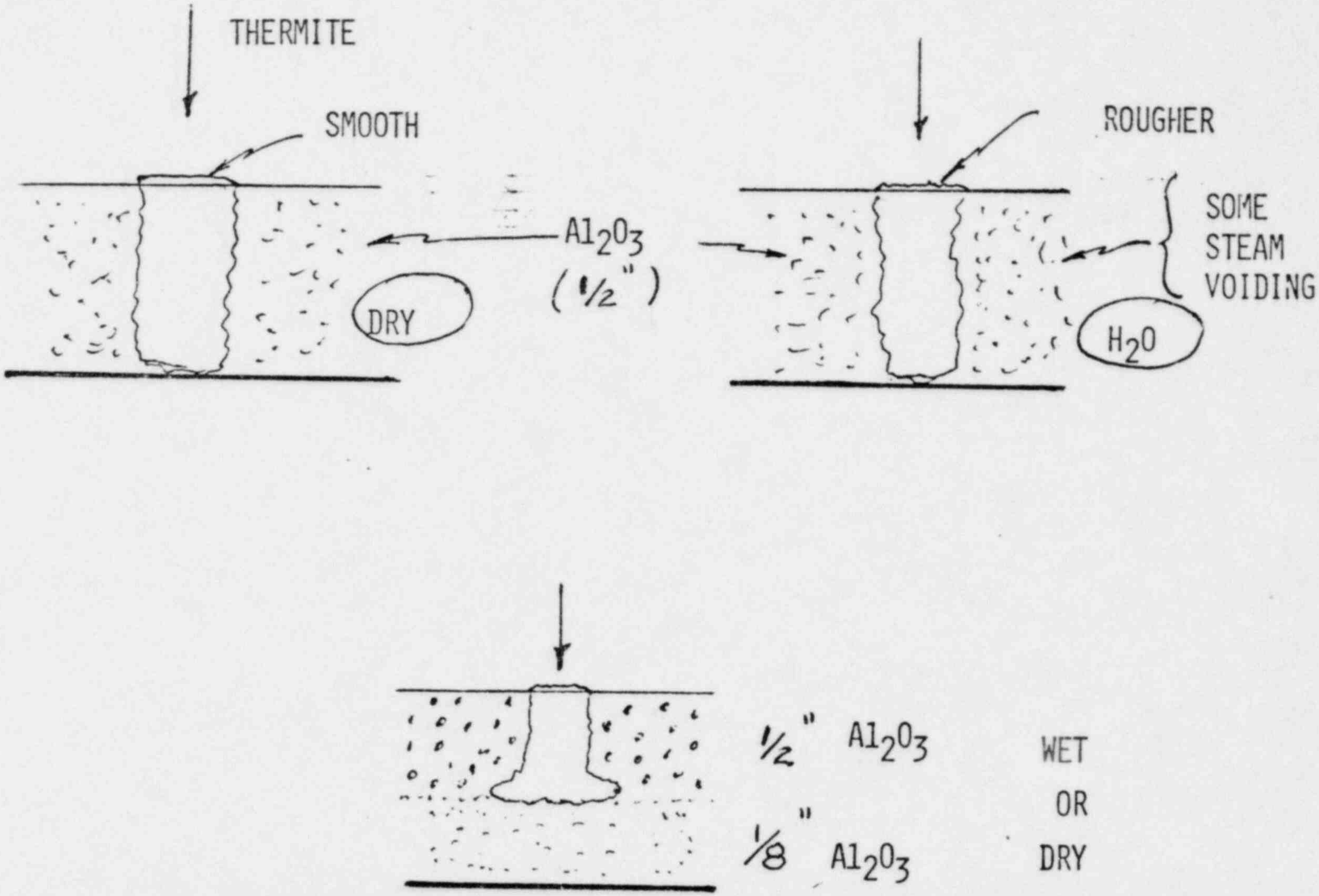
Comparison between adiabatic, isochoric calculations, HECTR-ES calculations, and VGES tank data for normalized peak pressure as a function of % hydrogen.

ELEMENT 5.6

FUEL/STRUCTURE INTERACTION

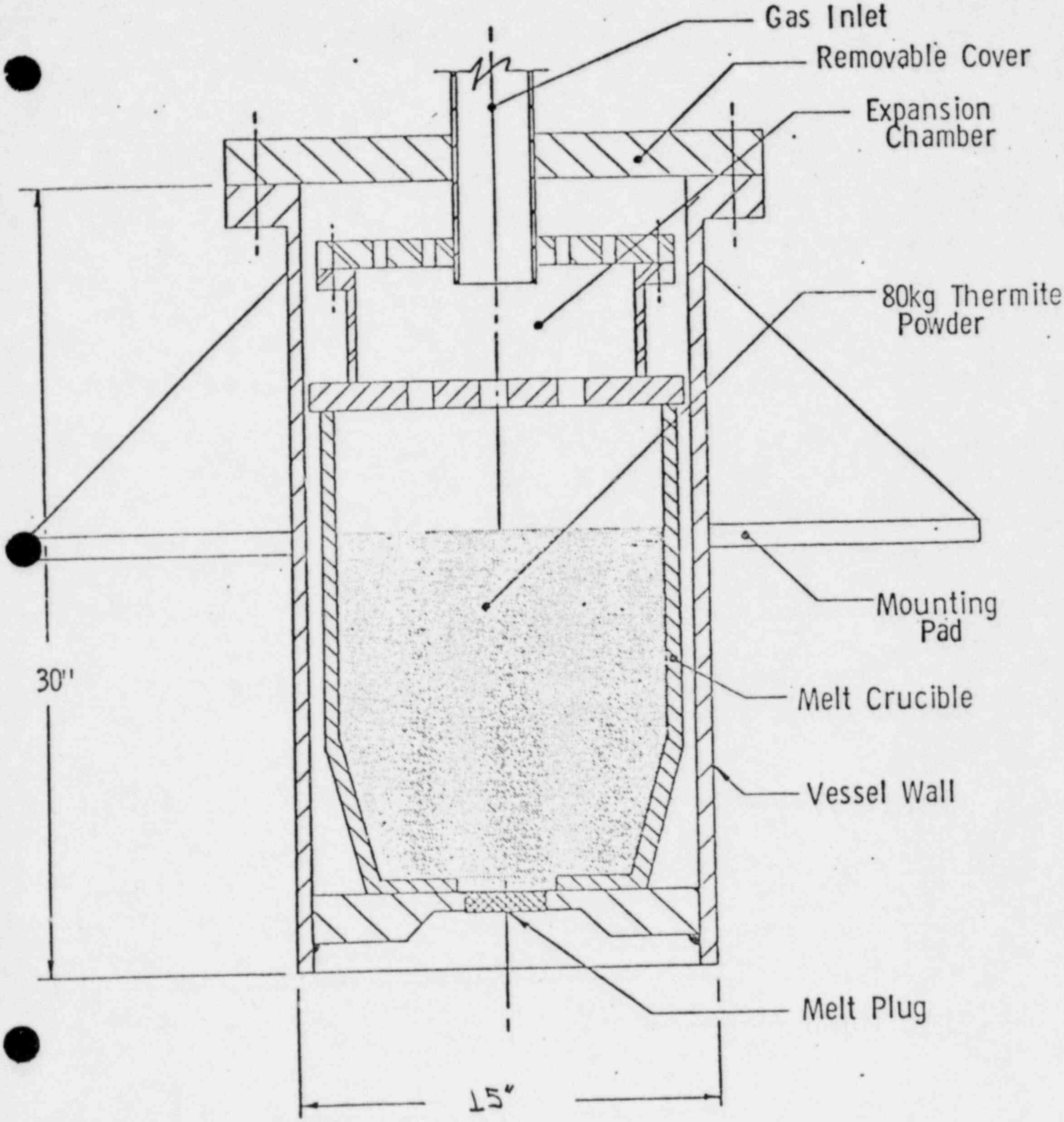
- PROGRAM TO PRODUCE LARGE (500 KG) UO_2 MELTS
  - + JUNE TEST FAILED THE MELT CRUCIBLE WITH ~80% OF A 273 KG CHARGE MELTED
  - + SEPTEMBER TEST
    - -- IMPROVED PROTECTION FOR ULTRASONIC THERMOMETRY
    - -- ARGON PURGE FOR THE MELT CRUCIBLE
    - -- TUNGSTEN INSERT IN THE CRUCIBLE
- SUSTAINED HEATING TESTS
    - -- GLASS TEST COMPLETE - RF SUSCEPTOR RINGS
    - -- UO_2 ON THO_2 - AL_2O_3 BEDS
- CORE RETENTION
  - + THERMITE ONTO AL_2O_3 GRAVEL w AND w/o WATER
  - + HIGH PRESSURE STREAMING TESTS
- BACKFITTING TO EXISTING PLANTS

  FINAL REPORT SEPTEMBER '82



CORE RETENTION BEDS

CONCEPTIONAL HIPS MELT GENERATOR



ELEMENT 5.7

CONTAINMENT SYSTEM INTEGRITY - CURRENT STATUS

- CONTAIN CODE - A GENERALIZED COMPUTER MODEL TO SIMULATE EXISTING AND PROPOSED CONTAINMENT SYSTEMS FOR BOTH LIGHT-WATER AND ADVANCED POWER REACTORS.

  PREDICT ABNORMAL CONTAINMENT LOADS RESULTING FROM SEVERE-ACCIDENT CONDITIONS AND ASSESS THE RADIOLOGICAL SOURCE TERM IN THE EVENT OF CONTAINMENT BREECH.

- CORCON CODE - MODELS MOLTEN-CORE CONCRETE INTERACTIONS - CAVITY DEBRIS POOL. PREDICTS CONCRETE PENETRATION, GAS GENERATION (H_2, CO, CO_2, ETC.), AND AEROSOL PRODUCTION.

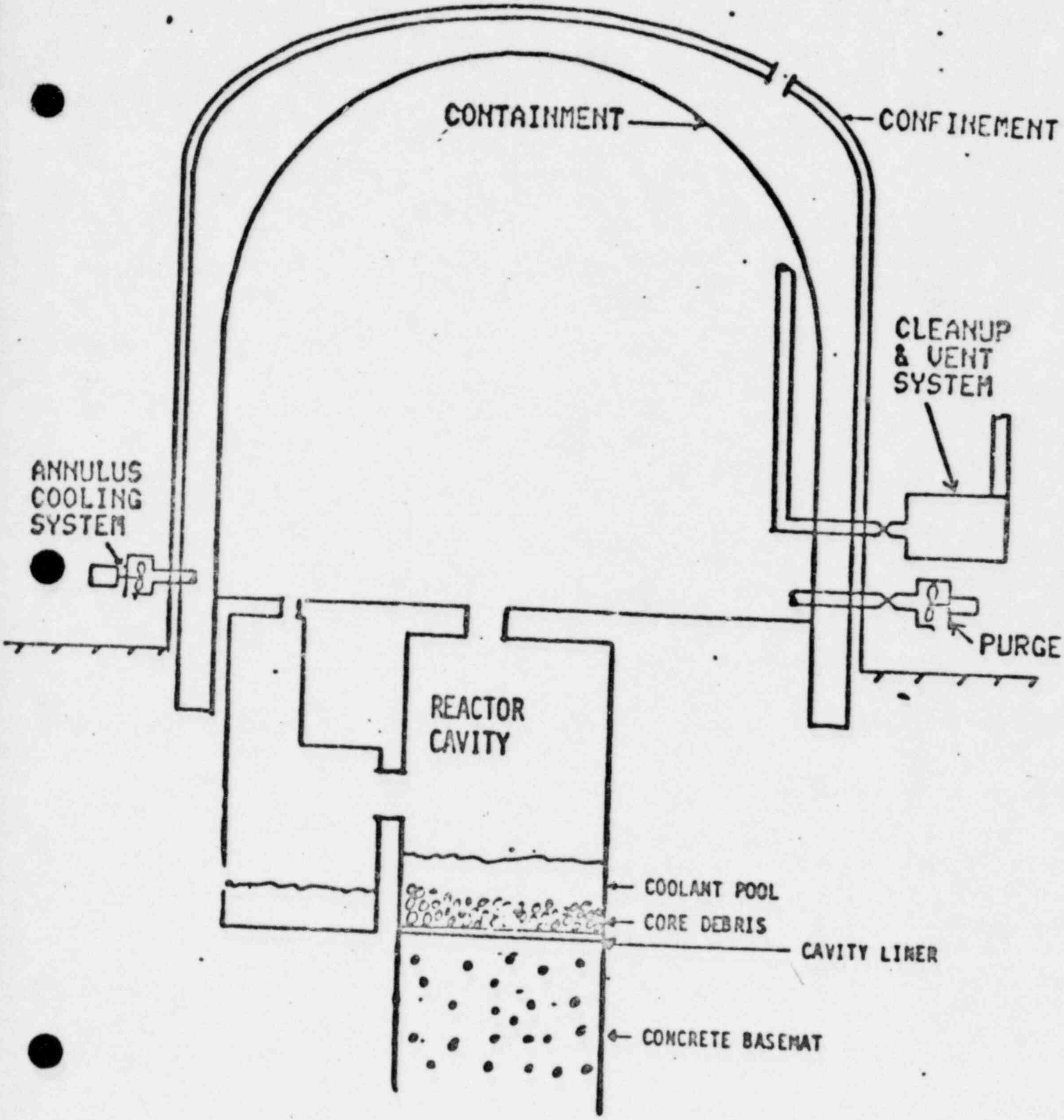
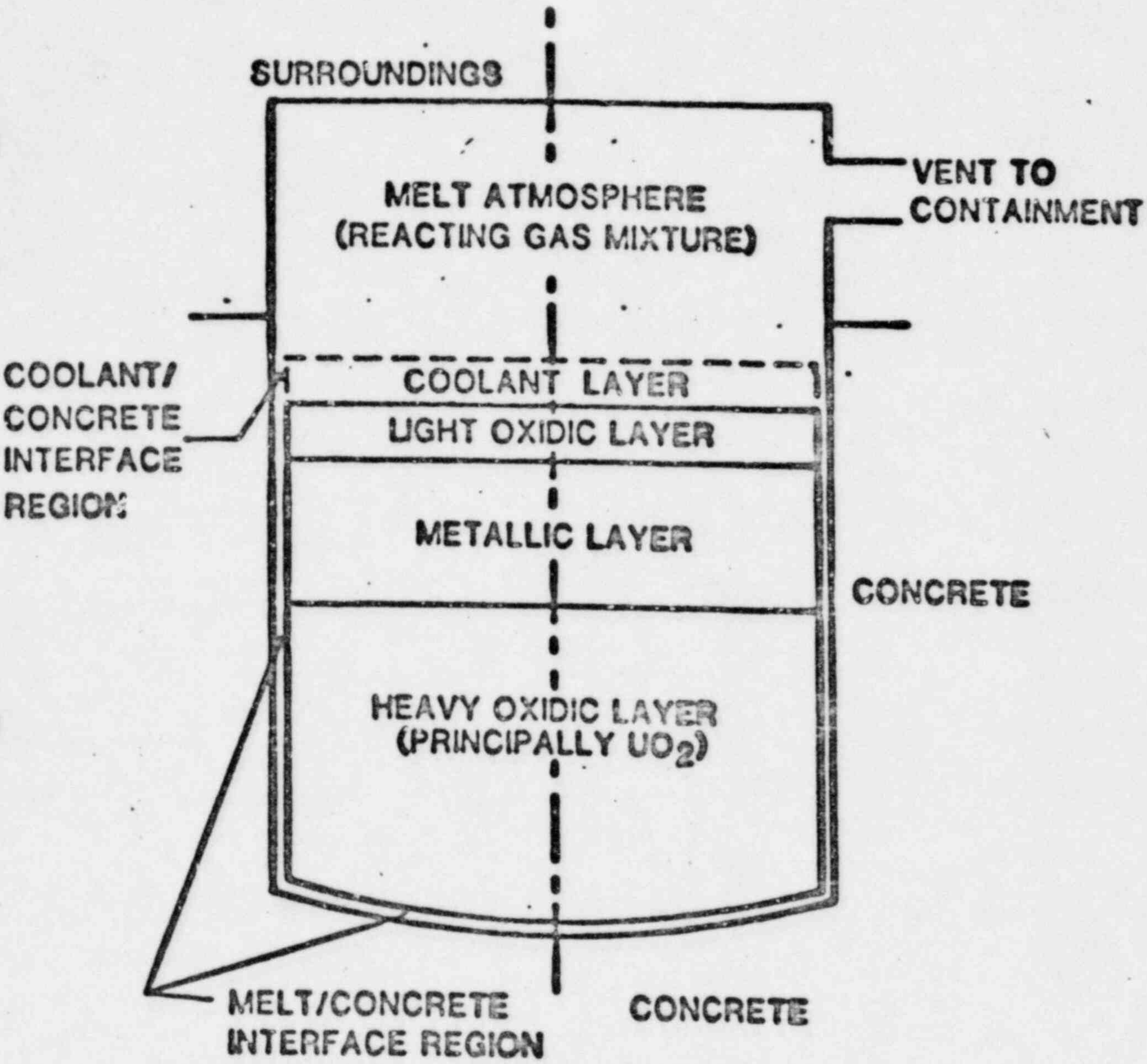


Figure I.2-2 CONTAIN's Modeling of Interconnected Compartments



CORCON CODE - DEBRIS POOL MODEL

CONTAINMENT ANALYSIS - MIDYEAR 1982 STATUS

- CONTAIN CODE: IS OPERATIONAL FOR TESTING AT SANDIA AND NRC

  DRAFT USERS' MANUAL RELEASED FOR DISTRIBUTION

  LIMITED DISTRIBUTION OF CODE FOR TESTING AND USER FEEDBACK

  LWR ENGINEERED-SAFETY-SYSTEM MODELING ONGOING

- CORCON CODE: MOD1 USERS MANUAL AND CODE RELEASED: LINK TO CONTAIN CODE OPERATIONAL. MOD2 DEVELOPMENT ONGOING, INCLUDES DEBRIS FREEZING AND COOLANT INTERACTIONS.

CONTAINMENT INTEGRITY

APPROACH:

- o THE GENERATION OF THE DATA BASE NEEDED TO ASSESS METHODS FOR PREDICTING THE BEHAVIOR OF LWR CONTAINMENTS UNDER ACCIDENT AND SEVERE ENVIRONMENTS BEYOND CURRENT DESIGN REQUIREMENTS
- o THE ASSESSMENT OF SELECTED PREDICTIVE NUMERICAL METHODS
- o THE IMPROVEMENT OF PREDICTIVE NUMERICAL METHODS AS NECESSARY

AREAS OF UTILIZATION

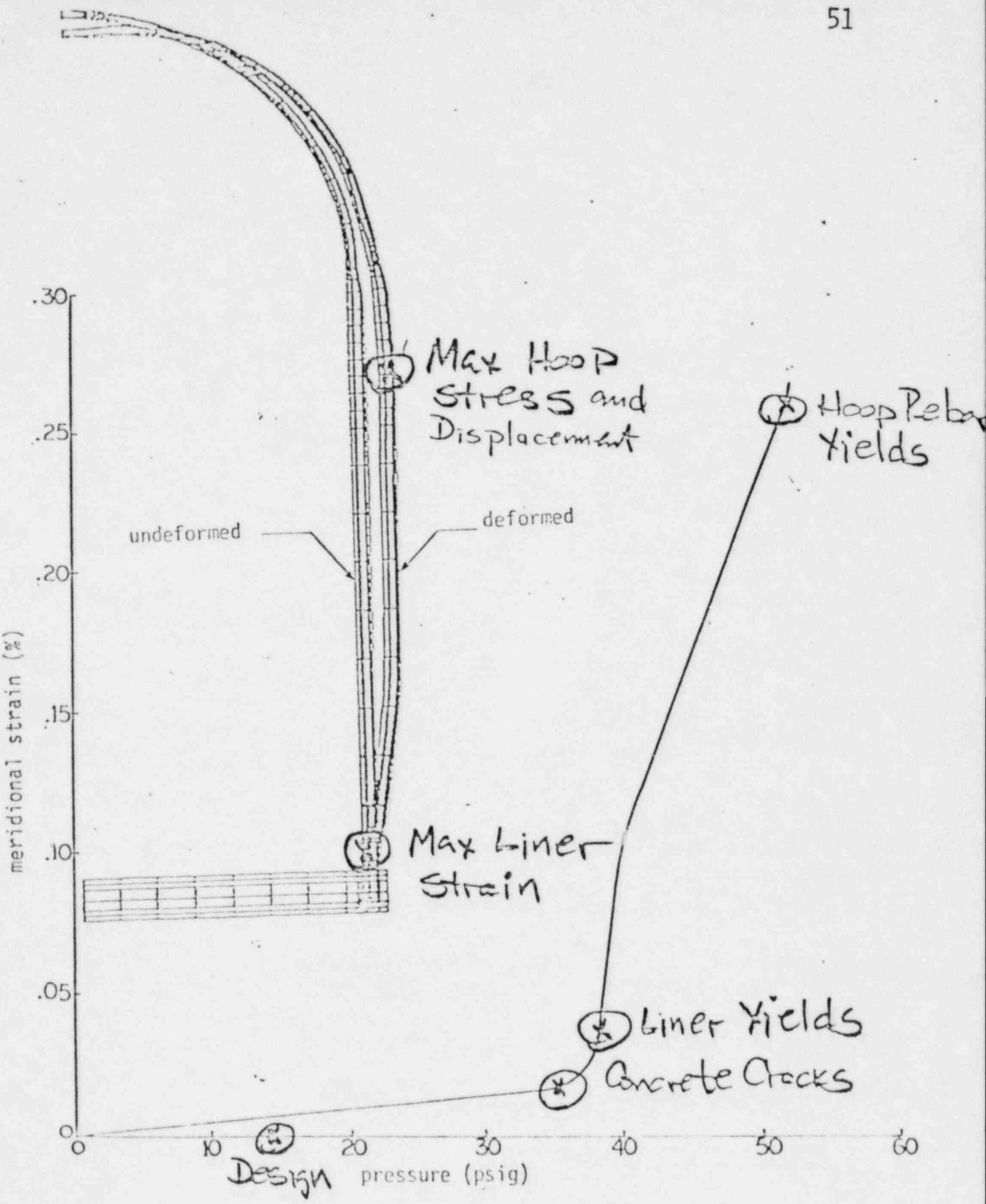
- o JUDGING CREDIBILITY OF CAPACITY ESTIMATES MADE ON BEHALF OF LICENSEES AND APPLICANTS
- o INPUT FOR RISK ANALYSES PERFORMED AS PART OF THE SEVERE ACCIDENT RESEARCH PLAN

CURRENT SCHEDULE

- o STATIC PRESSURE FY 82-84
- o UNSYMMETRIC PRESSURE FY 85-87
- o SEISMIC EFFECTS FY 88-90

STATE-OF-THE-ART-PREDICTIONS OF CONTAINMENT CAPACITY

- BASED ON SIMPLIFIED AXI-SYMMETRIC MODELS
- NO EXPLICIT CONSIDERATION OF PENETRATIONS
- ESTIMATES OF CAPACITY ARE SUBJECTIVE: WHEN DEFORMATIONS BEGIN TO INCREASE RAPIDLY WITH PRESSURE <u>OR</u> DEFORMATIONS ARE TOO LARGE FOR THE COMPUTER CODE, THE COMPUTATION IS STOPPED
- ESTIMATES SHOULD BE CONSERVATIVE, UNLESS PENETRATIONS FAIL
- ESTIMATES DO NOT GIVE A REALISTIC ESTIMATE OF FAILURE MODE (LOCATION)



Meridional Strain in the Liner at the Cylindrical Wall-Foundation Mat Intersection vs. Pressure

ELEMENT 5.9

FISSION PRODUCT RELEASE AND TRANSPORT

OBJECTIVES

TO DEVELOP EXPERIMENTALLY BASED MODELS FOR PREDICTING THE QUANTIFY, TYPE, AND TIMING OF RADIONUCLIDE RELEASE TO THE ENVIRONMENT AS A RESULT OF A SEVERE ACCIDENT AT A COMMERCIAL LWR.

FOR:

(1) DEVELOPING PLANT SITING POLICY AND REGULATIONS

(2) ASSESSING EMERGENCY PLANNING REQUIREMENTS

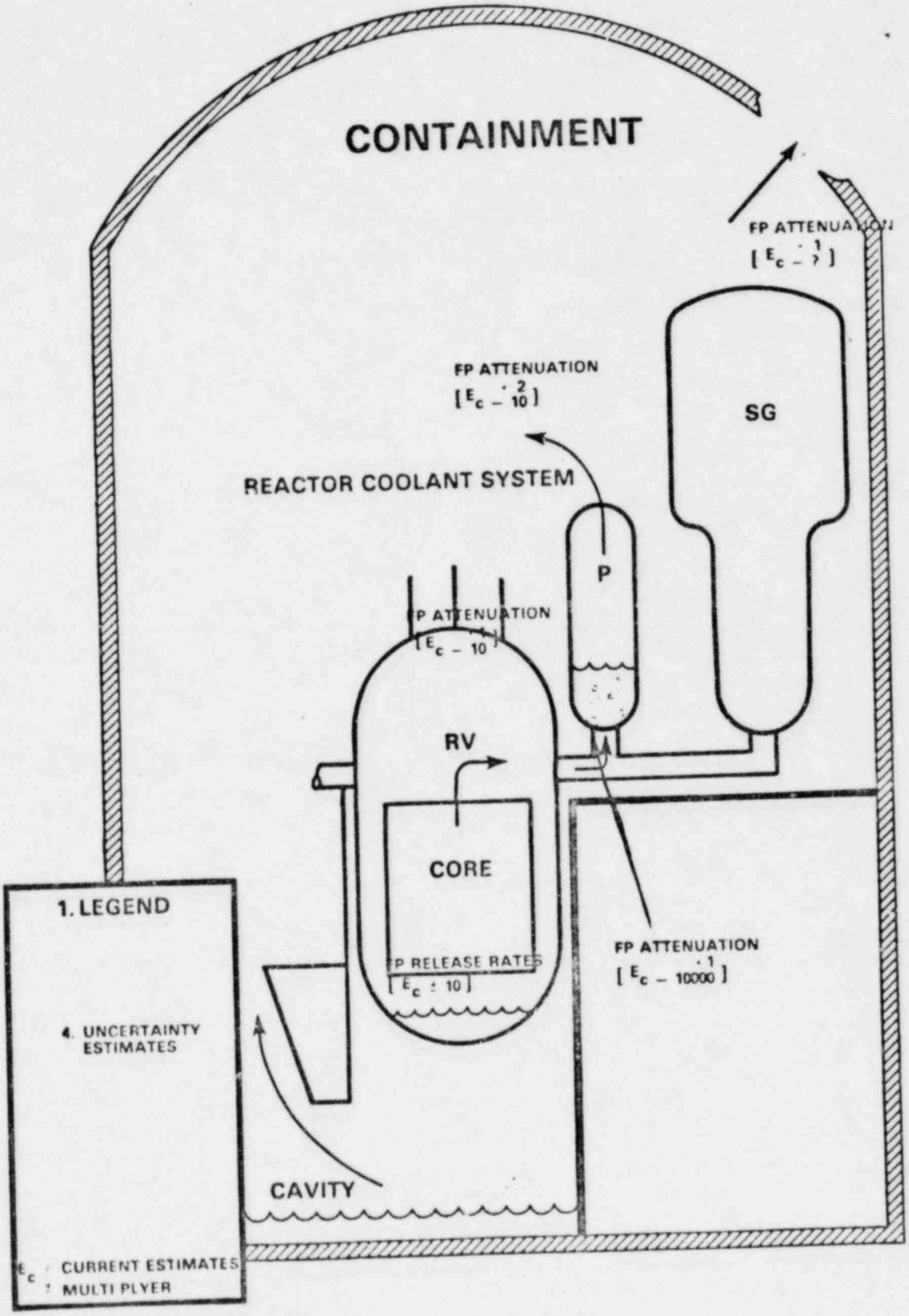
(3) IMPROVING PROBABILISTIC RISK ASSESSMENT TECHNIQUES

(4) DEFINING THE IN-PLANT RADIATION ENVIRONMENT FOR EQUIPMENT QUALIFICATION

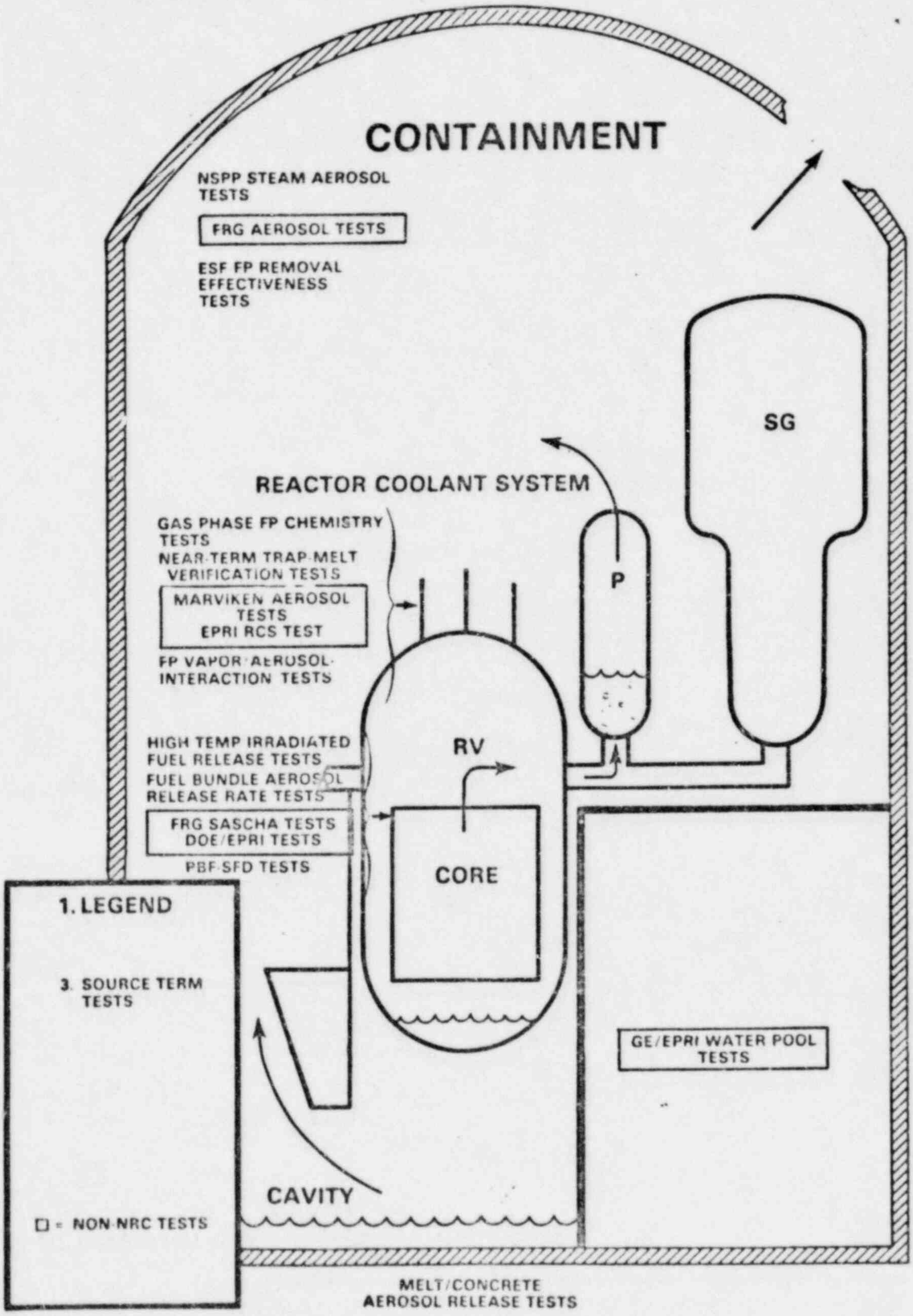
(5) DETERMINATION OF THE DESIGN REQUIREMENTS AND DESIGN ADEQUACY OF FISSION PRODUCT MITIGATION ESFs

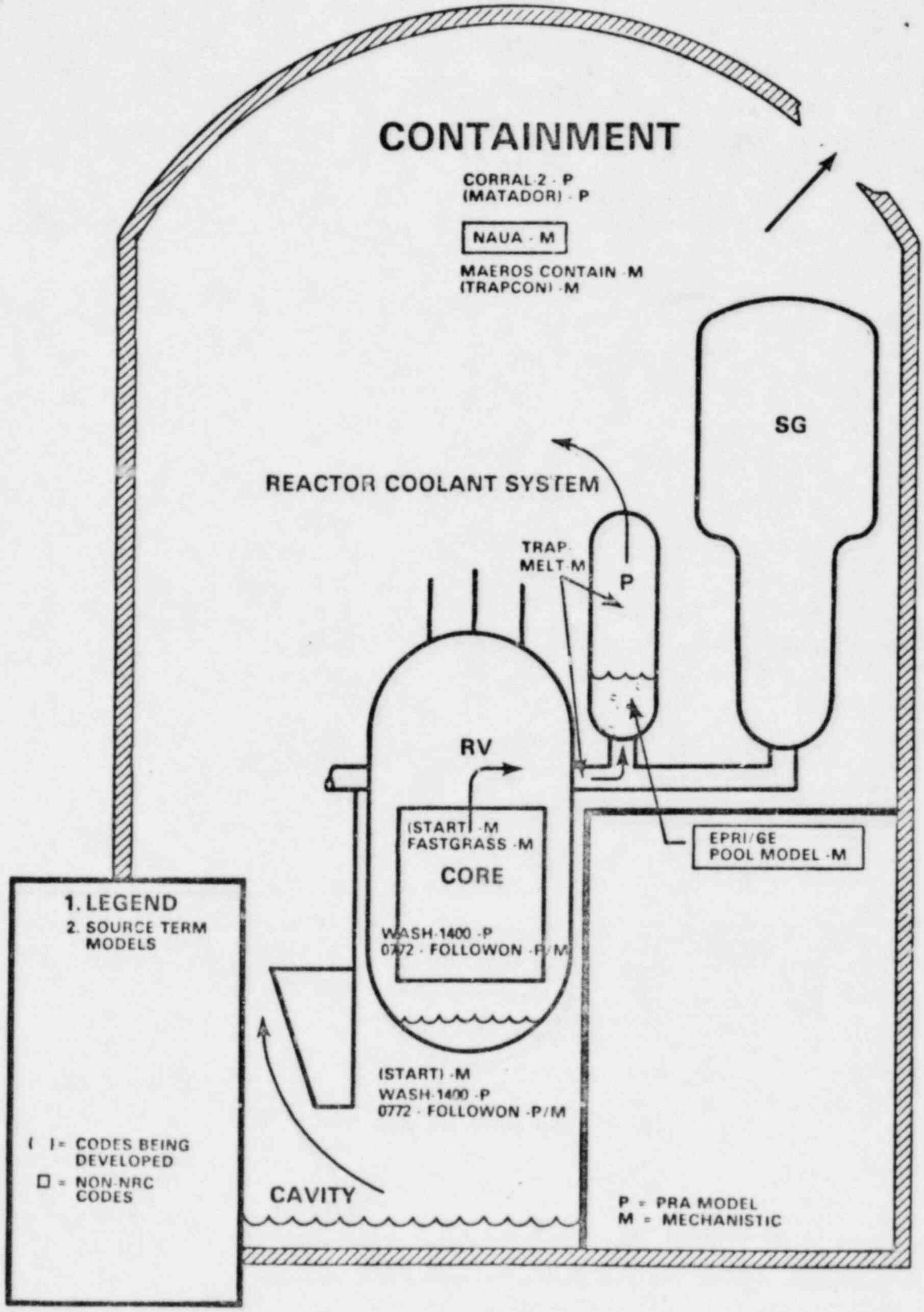
MAJOR UNCERTAINTIES AND DATA NEEDS IDENTIFIED IN NUREG-0772

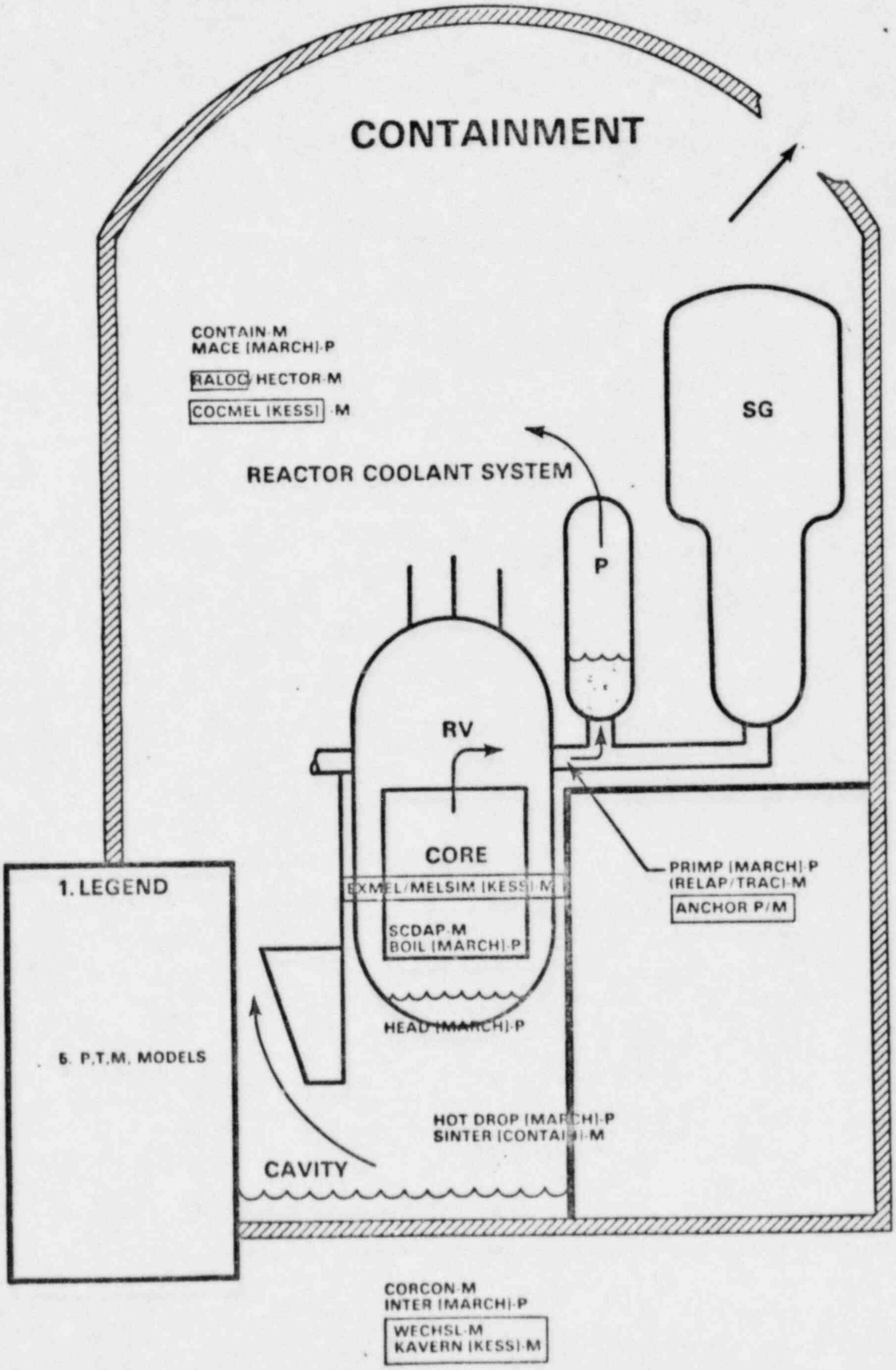
1. RCS AEROSOL BEHAVIOR (EXPERIMENTAL DATA FOR MODEL VERIFICATION).
2. RCS THERMAL/HYDRAULIC MODELS UNDER CORE MELT ACCIDENT CONDITIONS.
3. CONTAINMENT FAILURE TIME, MODE, LOCATION (EXPERIMENTAL DATA AND ANALYSIS).
4. FISSION PRODUCT CHEMISTRY (EXPERIMENTAL DATA).
5. LESS VOLATILE FISSION PRODUCT, CONTROL MATERIAL, AND STRUCTURAL MATERIAL AEROSOL FORMATION RATES (IN-VESSEL AND DURING INTERACTION WITH CONCRETE) - (EXPERIMENTAL DATA).
6. AEROSOL BEHAVIOR IN CONDENSING STEAM ATMOSPHERES (EXPERIMENTAL DATA).
7. REMOVAL OF PARTICULATE FISSION PRODUCTS IN WATER POOLS AND ICE BEDS (EXPERIMENTAL DATA AND MODELS).
8. THE EFFECT OF A HYDROGEN BURN ON FP PHYSICAL AND CHEMICAL FORMS (EXPERIMENTAL).
9. COUPLED MODELS OF CONTAINMENT FISSION PRODUCT VAPOR TRANSPORT, AEROSOL BEHAVIOR, STEAM EFFECTS, AND EFFECTS OF ESFs.



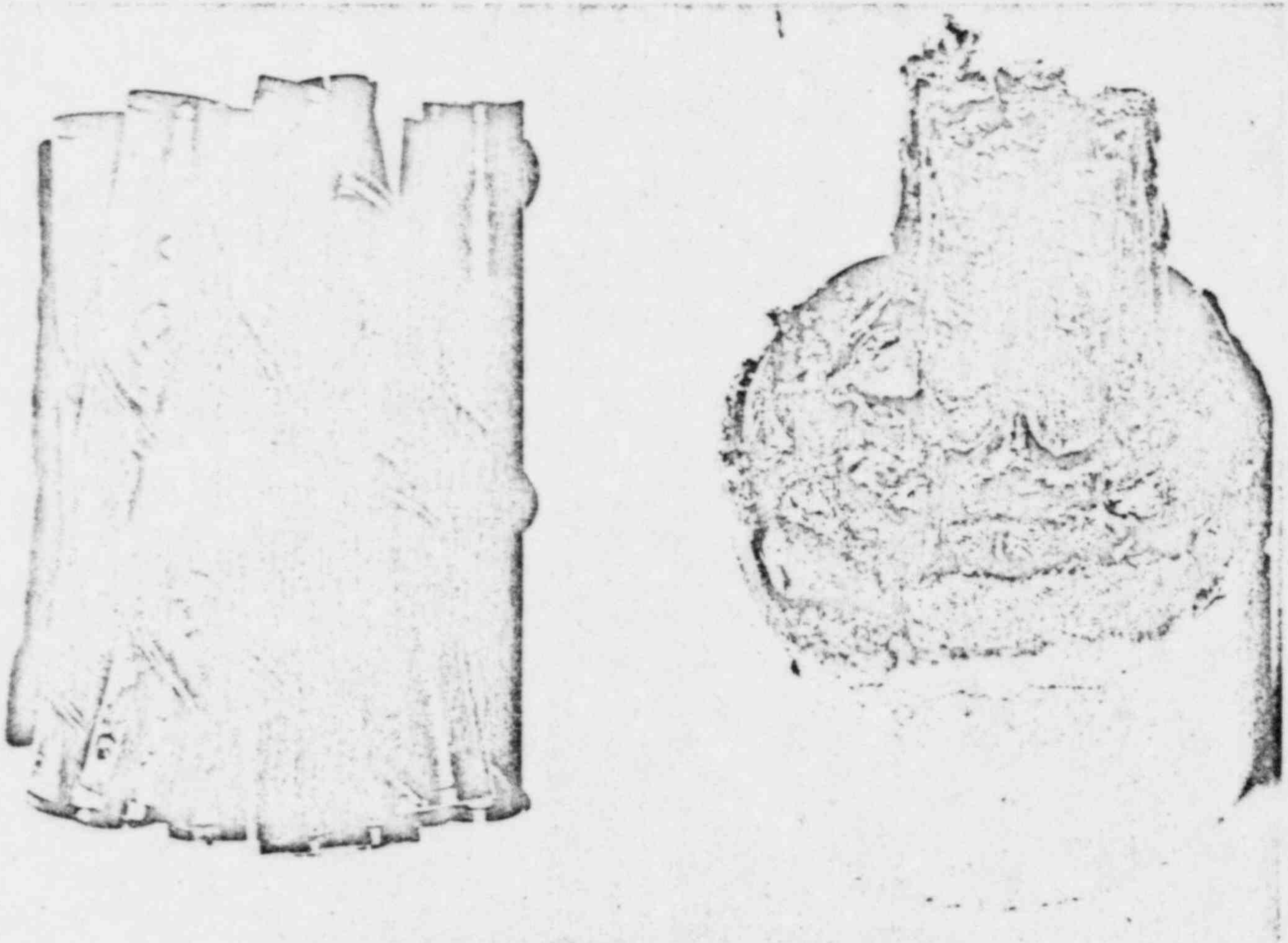
CONTAINMENT
FP ATTENUATION
[$E_c \stackrel{\cdot}{-} \, ^{1}_{?}$]
FP ATTENUATION
[$E_c \stackrel{\cdot}{-} \, ^{2}_{10}$]
SG
REACTOR COOLANT SYSTEM
P
FP ATTENUATION
[E_c – 10]
RV
CORE
FP ATTENUATION
[$E_c \stackrel{\cdot}{-} \, ^{1}_{10000}$]
FP RELEASE RATES
[E_c ± 10]
1. LEGEND
4. UNCERTAINTY ESTIMATES
E_c : CURRENT ESTIMATES
: MULTI PLYER
CAVITY


CONTAINMENT
NSPP STEAM AEROSOL TESTS
FRG AEROSOL TESTS
ESF FP REMOVAL EFFECTIVENESS TESTS
SG
REACTOR COOLANT SYSTEM
GAS PHASE FP CHEMISTRY TESTS
NEAR-TERM TRAP-MELT VERIFICATION TESTS
MARVIKEN AEROSOL TESTS
EPRI RCS TEST
FP VAPOR-AEROSOL-INTERACTION TESTS
P
HIGH TEMP IRRADIATED FUEL RELEASE TESTS
FUEL BUNDLE AEROSOL RELEASE RATE TESTS
RV
FRG SASCHA TESTS
DOE/EPRI TESTS
PBF-SFD TESTS
CORE
1. LEGEND
3. SOURCE TERM TESTS
GE/EPRI WATER POOL TESTS
CAVITY
☐ = NON-NRC TESTS
MELT/CONCRETE AEROSOL RELEASE TESTS


CONTAINMENT
CORRAL-2 - P
(MATADOR) - P
NAUA - M
MAEROS CONTAIN -M
(TRAPCON) -M
SG
REACTOR COOLANT SYSTEM
TRAP-
MELT-M
P
RV
(START) -M
FASTGRASS -M
CORE
WASH-1400 -P
0772 - FOLLOWON -P/M
EPRI/GE
POOL MODEL -M
1. LEGEND
2. SOURCE TERM
MODELS
( )= CODES BEING
DEVELOPED
☐ = NON-NRC
CODES
(START) -M
WASH-1400 -P
0772 - FOLLOWON -P/M
CAVITY
P = PRA MODEL
M = MECHANISTIC

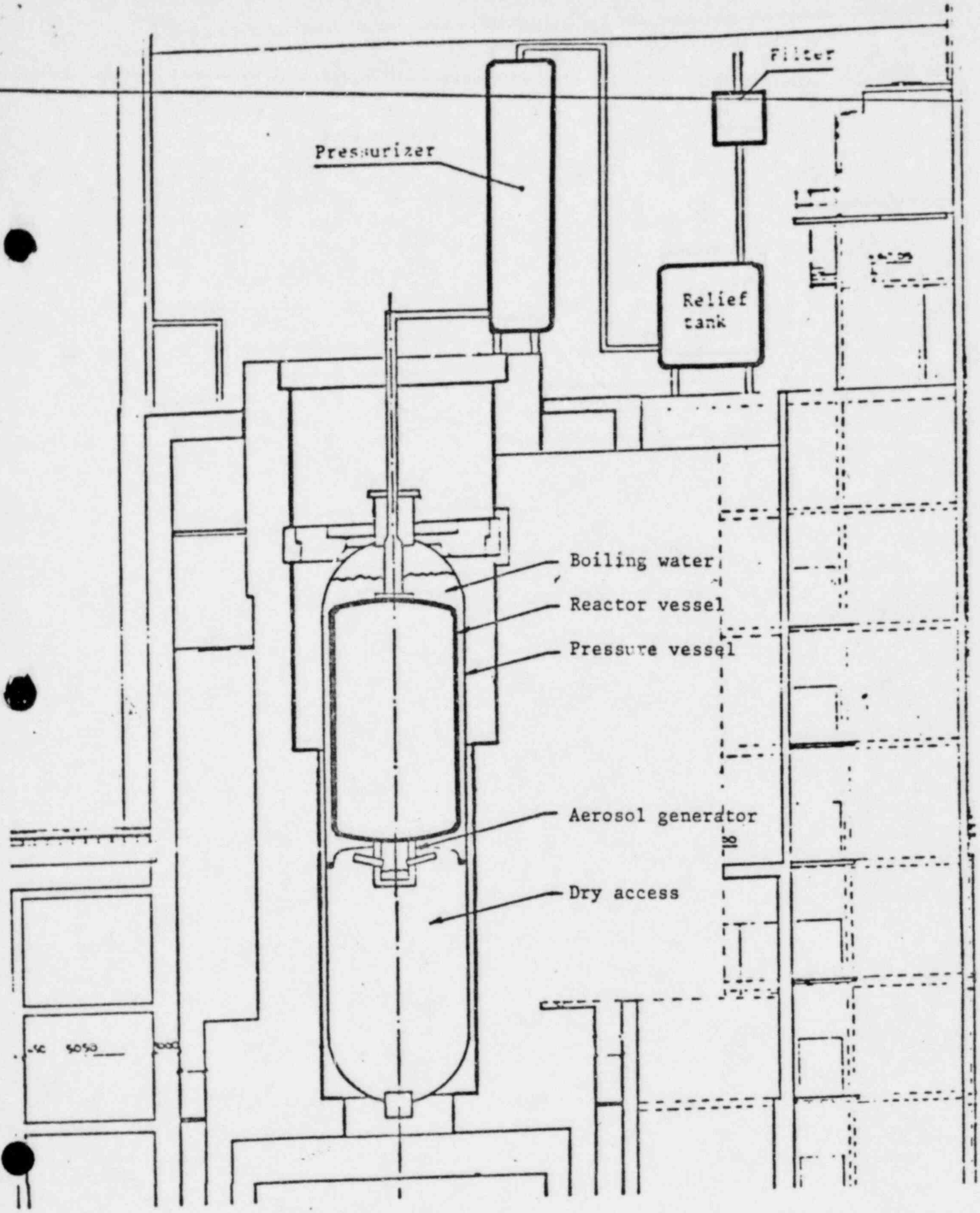

CONTAINMENT
CONTAIN-M
MACE (MARCH)-P
RALOC HECTOR-M
COCMEL (KESS) -M
REACTOR COOLANT SYSTEM
SG
P
RV
CORE
EXMEL/MELSIM (KESS)-M
SCDAP-M
BOIL (MARCH)-P
HEAD (MARCH)-P
PRIMP (MARCH)-P
(RELAP/TRAC)-M
ANCHOR P/M
1. LEGEND
6. P,T,M, MODELS
HOT DROP (MARCH)-P
SINTER (CONTAIN)-M
CAVITY
CORCON-M
INTER (MARCH)-P
WECHSL-M
KAVERN (KESS)-M

ORNL–PHOTO 6032–82



(a) (b)

Fig. 12. Steps in the silver–Zircaloy candling process at 1400°C (a) showing Zircaloy wetted by the silver alloy and cladding completely melted off and (b) at 1800°C.



MARVIKEN TEST FACILITY

STATUS OF THE MARVIKEN AEROSOL TRANSPORT TEST PROGRAM (MULTINATIONAL EFFORT)

PURPOSE: PROVIDE LARGE-SCALE INTEGRAL DATA TO VERIFY FISSION PRODUCT TRANSPORT AND PLATEOUT CODES (TRAP-MELT).

CHARACTERISTICS: FULL-SCALE COMPONENT GEOMETRIES. FULL-SCALE AEROSOL AND FISSION PRODUCT MASSES. REALISTIC TEMPERATURES.

STATUS: TECHNICAL ASPECTS DEFINED. PROJECT INITIATION AWAITING ADMINISTRATIVE ACTIVITIES (FORMAL AGREEMENT, FUNDING COMMITMENTS, ETC.)

SIGNIFICANT UNCERTAINTIES EXIST IN AREAS SUCH AS:

<u>COMPLETENESS</u>

<u>HUMAN BEHAVIOR</u>

<u>RELIABILITY DATA</u>

<u>NATURAL PHENOMENA</u>

<u>ACCIDENT PHYSICAL PROCESS PHENOMENA</u>

<u>CONSEQUENCE ANALYSIS</u>

AND THESE WILL BE THE SUBJECT OF RESEARCH