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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON SEQUOYAH NUCLEAR PLANT

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Nuclear Regulatory Commission
Room 1046
1717 H Street, N.W.
Washington, D.C.
Wednesday, July 9, 1980

The Subcommittee met, pursuant to notice, at 8:35 a.m.

BEFORE:

J. CARSON MARK, Chairman
WILLIAM W. MATHIS
R. SAVIO
I. CATTON
W. LIPINSKI
Z. ZUDANS

NRC STAFF PRESENT:

P. STAHLE	LARRY MILLS
J. HALAPATZ	MARK BURZYNSKI
R. WESSMAN	ED MERRICK
W. CAMPBELL	GEORGE DILWORTH
JERRY BALLANTINE	

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PROCEEDINGS

MR. MARK: The meeting will come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards' Subcommittee on the Sequoyah Nuclear Plant.

I am Carson Mark, Subcommittee Chairman. Other ACRS members present are: William Mathis, on my left; and perhaps later Chester Siess; ACRS consultants Ivan Catton, Walter Lipinski. Dr. Zudans. Richard Savio of the ACRS Staff is also here.

The purpose of the meeting is to review the Tennessee Valley Authority's application for a license to operate Sequoyah Nuclear Plant Units 1 and 2 at full power.

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

Richard Savio is the designated Federal employee for the meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register of June 24, 1980.

A record of the meeting is being kept and will be made available as stated in the Federal Register notice. It is requested that each speaker first identify himself and speak with such clarity and volume so it can be readily heard, and if at all possible take advantage of a microphone.

We have received no written comments or requests for time to make oral statements from members of the public.

Before we start the meeting, I might remind you there was a subcommittee meeting on the Sequoyah status just about one month ago, the 2nd of June. A number of items which have previously been regarded as open were, I believe, settled at that subcommittee meeting.

There are still a few open items, both from the pre-TMI context and some from the post-TMI context; and it is not clear how far we can get regarding all of those to be settled today, but it would be our hope to go as far as we possible can; as far as things are clear we would like to bring those as far toward final resolution as possible.

To proceed with the meeting, I call upon Carl Stahle of the NRC staff to report the status of their review.

STATUS OF THE NRC REVIEW (NRC/TVA)

MR. STAHLE: My name is Carl Stahle. I am Project Manager here for the Nuclear Regulatory Commission on the Sequoyah Project.

I shall start this review with the status of where we stand in particular on the schedule, and then we will follow the agenda as you have in front of you. Selected members of the staff are here to assist me in this review. Others are available on call if necessary.

In addition, Mr. "essman from INE is here to answer questions in the area of his responsibility.

The schedule for the plant: Delays have been incurred at the plant primarily due to conformance with a bulletin, 9714, which has to do with the manuals in support of the plant.

I am pleased to announce that this has been satisfactorily completed and the criticality was achieved on July 5th.

The Resident Manager informs me that the zero power tests are proceeding satisfactorily and we expect that on or about July 12th the zero power tests should be completed and it would be possible at that time to initiate tgd kov povdq sdrs oqnggal-

On the basis of that starting date, we estimate that the completion date for the low power test program is the end of July; and again assuming smooth performance with the low power test program, power could begin, if a license were provided, the first week of August, and subsequent full power operations.

Now, the plant schedule in large measure sets the stage for our review process because we always hope these instances to be consistent with plant schedule and the readiness of the plant.

With that in mind, the test program, that is, the low power test program, we plan to have completed formally our evaluation and license that must -- a license amendment that must be completed on or about Thursday of this week.

At this stage there are small revisions to the safety evaluation report, but I am confident that this letter amendment will be processed this week, consistent with the schedule I mentioned, of July 12th for the initiation of the low power test program.

Now, to license for full power the following things will have to be completed: Staff reviews must satisfactorily resolve the issues that have been identified in a number of documents; in particular, new Regulation 0694, which describes in particular the full power requirements.

Secondly, a letter, of course, from the ACRS is needed, and their guidance.

Last but not least, of couse, the Commission's approval or endorsement received beyond the low power test program.

To be consistent, therefore, with the schedule of the plant, it is obvious we need a very expeditious review on the part of the staff; very timely responses, of course, from the applicant in these areas in the event the review may reveal, may indicate, certain deficiencies; and, of course, to be as responsive as possible to the ACRS's guidance which we hope will be forthcoming through the series of meetings that we are now conducting.

The staff completion of a writen SCR, of course, now would be a formal supplement. To be consistent with this schedule, it would have to be, of course, completed within the next two weeks.

Now, this is the challenge we face, all parties -- staff, the applicant and, of course, the ACRS.

With that in mind, I think we have set the stage, based on the schedule, so I will proceed now to where we stand with respect to the status of the non-TMI items.

STATUS OF NON-TMI OPEN ITEMS

MR. STAHEL: I will use this chart and I will update it accordingly as to where we stand.

Number 1 is complete. The staff provided me the understanding that we have an agreement in principle with TVA. The program probably will take approximately 18 months to complete. Further details will be forthcoming in the next few months, and this will be part of confirming the margins of safety components of piping and structure for the system.

We think we have a program here that satisfies the intent of the ACRS.

Item 2 I consider complete. We are waiting on data on settlement. I have not seen any data that identifies any unusual settlement problems. The only remaining issues would be the continuation of monitoring of settlement. For my purposes, I consider this resolved.

Item 3, I consider this complete. Staff has completed its re-review. The draft SER is prepared. I am awaiting from the applicant a confirmatory letter to provide the data officially, but that has already been in the hands of the staff.

Item 4 is open at this moment. A few more details of discussions are needed. I expect this week to show this item completed; however, I still regard this as open.

Item 5 -- excuse me?

MR. MARK: Are you waiting for additional communications on Item 4, or you have not had time to check through the ones you have?

MR. STAHLE: We need some more information here, a letter, a commitment, from the applicant. There have been discussions with the staff, so I am quite sure we will have that in hand this week.

MR. MARK: Fine.

MR. STAHLE: Item 5, I regard that as complete. Revision 4 is in our hands. The staff informs me that we feel this is satisfactory at this point in time. I should have an SER on this shortly.

There would be involved later on a demonstration of this system, but for all intents and purposes the evaluation, I think, is complete.

Item 6, we identified this item at the last meeting. It is an important item, a critical path item, an item which has been under intensive review by the staff and the applicant.

We were provided a substantial submittal on June 16th. The staff has been reviewing this. Fortunately, based on the Commission memorandum order of May 23rd, 1980, there is some relief or flexibility, if you will, in the scheduling.

The schedule now with respect to this plant and others is, the SER reports, which are due by February of 1981, and compliance by June 30, 1982 -- this does not diminish the important of this item; this simply provides a little more flexibility on the time in which to review it.

There are issues that are under review. We will be discussing this with the staff. That most likely will have to be completed before we issue a full power license. Nevertheless, I wanted to indicate to you some flexibility in the scheduling we are dealing with.

I do now want to diminish the importance of this item. To emphasize this point even more, I think, as others know, there will be, beginning next week, regional meetings on this entire subject matter. So the industry, the public, will understand more fully our requirements in this entire area. But I would repeat, this item is still open, but I am confident that we will be able to mvoe this item to completion prior to the issuance of the full power license.

Item 7, I regard this complete on the basis of the fact the staff said they have a memo in preparation. It should be complete.

Item 8, it is ongoing; it is not complete. I regard this as open. The staff informs me we feel that additional input coming into this area, it will take them approximately two weeks to complete their schedule. Again, I am confident we will continue to meet our schedule in this item.

Item 9, and actually Item 13, are bulletins. Let me just provide you a brief status in this area for you.

With regard to Bulletin 9727 -- excuse me 7927, this bulletin dealt with the loss of non-Class IE instrumentation control power system during operations to all power reactor facilities with an operating license to those nearing licensing.

This bulletin outlined the actions to be taken to address control system malfunctions and significant loss of information for the control room oeprator as a potential consequence of loss of 120 volt alternate 80 current control room to these plant systems; and, further, the INE information notice of 80-10 issued March 7th provided information relating to Crystal River, February 26th, in which a significant loss of information to the operator resulted in loss of power.

I indicate here that this item is still ongoing. The response to this bulletin officially, as I understand it, was July 11th. I am confident also this will be resolved, but I will continue to regard this as open and its implications for licensing will have to await further review of this matter.

Items that I passed over here, which was 8006, the staff has been reviewing this material. Further review is necessary, according to the staff, but we will continue to follow this effort jointly with INE. Its impact on the licensing and so forth will have to await further review.

Again, I don't foresee any major problems in this area other than possibly the timeframe at which we consider that this has to be completed.

Moving back to Item 10, this item is complete. There is in the staff review that I just received a short while ago a number of design and procedural changes that need to be made by the next -- well, the next review. These have not been conveyed to the applicant at this point in time. We plan to discuss these, but subject to these changes and agreement to perform such, this item will be closed.

I assume that this will be completed to our satisfaction as well as the applicant's and, most importantly, this item, of course, becomes an item to be completed at a later date. This item is only subject to a commitment to perform.

Item 11, there is need for a commitment letter in this area to provide certain confirmatory analysis, and a commitment to support an audit analysis. I have yet to make this available to the applicant. I will do so shortly. I regard this as a relatively minor item. I think at this point I judge it as a complete item. Nevertheless, it will take a few days to resolve this, but it will be done and consistent with our schedule.

Item 12, this is now complete. The staff has reviewed this. Telecon discussions show that there are no problems in this area other than now simply to get a confirmatory letter to confirm those discussions that have been held in this area.

In sum, the total on the items that are indicated in this chart, in my judgment eight are completed and five are open, all of which I think will be satisfactorily completed within a week or two-week period.

Since our last discussion there have been two items I wanted to bring to your attention.

One deals with the steam generator ports and tube inspection. As the result of cracking in some of the Westi-ghouse steam generators, we issued a letter to the applicant on June 23rd indicating that we wanted a commitment to put in steam generator inspection ports; and, secondly, we informed them of the possibility of a need to plug at a later date maybe Row 1 of the steam generator tubes, and TVA's response to this is as follows:

With respect to the steam generator inspection ports, they are proposing to intially test a new camera device for remote inspection of two supports in lieu of additional ports. The camera inspection device will be inserted through the existing handholds located between the tube sheath and lower support plate in line with the tube length.

They further informed me that if the device proves unsuccessful or inadequate, TVA remains prepared to install the required inspecton ports before startup after the first reviewing.

The results of the inspection of the device should be available about March 1st, 1981. The staff informs me that this is acceptable. They eagerly await the results of the camera approach.

Now, with regard to the tubes, the TVA indicated that it does recognie the potential for cracking on the first row and, of course, eagerly awaits the results of the tube testing, nondestructive and destructive testing of these tubes, recognizing that at some point in time it may be necessary to plug the first row.

The second item that has not been discussed with you, but we feel it is desirable to bring to your attention, is as follows:

Last year, during the hot functional test, one of the pipe supports, Item 33 on this schematic drawing, actually was inoperable. This pipe failed to slide vertically in the direction as the pressurizer expanded during the heatup and the result being that the pipe bent.

There were two options at that time to fix the pipe: One would be to replace it completely. The second alternative, which TVA opted to followed, as I understand it, is the weld-draw technique. This technique involves cutting two grooves 270 degrees around the circumference of the pipe and rewelding this material, and because of the welt metal shrinkage, providing the stressing to plastically straighten the ipe.

The technique works as far as straightening the pipe

MR. CATTON: Couldn't they have left the pipe bent?

MR. STAHLE: No, sir. As I understand geometry, the confi ion was such that it was necessary for having a slope on this pipe for appropriate drainage of the accumulated water on this line and for other reasons.

Maybe TVA can explain them, but basically it could not remain in its condition.

DR. ZUDANS: Was the weld status examined before this weld-drawing technique was applied? Were there cracks or any defects?

MR. STAHLE: I don't think so, but I would like to defer my comments on this.

DR. ZUDANS: I would like to hear a positive or negative answer to Ivan's question. Why wasn't the pipe left the way it was?

MR. STAHLE: Maybe TVA can respond as to why the pipe was not left as it is.

MILLS: Larry Mills, Manager for Nuclear Regualtion Safety, TVA.

We do have a gentleman highly involved in this from our design organization, Ed Merrick. I will ask him to reply to this question.

MR. MERRICK: My name is Ed Merrick. The pipe was left in the configuration, where it was needed to be bent in the configuration, that it was in part bending because of the fact that you have a vapor lock at the pressurized relief pipe.

MR. CATTON: That horizontal section was tipped down?

MR. MERRICK: Right.

MR. CATTON: How much?

MR. MERRICK: I believe it wan't much, maybe three-quarters of an inch.

MR. CATTON: What is the required slope on the horizontal line?

MR. MERRICK: I am not qualified to answer that.

DR. ZUDANS: On this picture, could somebody sketch how the pipe looked after the pressurized valve had moved up nd the support didn't slide? The support is which one? That is where the pipe support text is written, that is the support that didn't move?

MR. STAHLE: Number 33 was the support that failed, and the consequence of that wass the pipe actually was bent. It did not have the ability to move when it went through the 500 degree temperature.

DR. ZUDANS: The report said the support was supposed to slide.

MR. STAHLE: It was supposed to.

DR. ZUDANS: It didn't slide; therefore, the pipe was bent upwards. So what went upward was between 33 and 35?

MR. MERRICK: Thirty-five and 36 were bent slightly upwards.

DR. ZUDANS: And they were enough upwards so that it would create a vapor lock? In other words, they were more upward than the pipe diameter?

MR. MERRICK: No, sir.

DR. ZUDANS: Why would you create a vapor lock?

MR. CATTON: Right at 33, the inside of the pipe, if it were lower than 35.

DR. ZUDANS: If 33 was lower than 35 by less than a real thickness, there is still lots of diameter to flow horizontally. It could not create a vapor lock.

MR. CATTON: That is true.

MR. STAHLE: Let me suggest, let me introduce why, the subject is brought up, and I think some of these facts and more detail will be presented.

We have a minority opinion in this area by one of our staff members with respect to the repair of the pressurized relief line. Mr. Halapatz looked at this matter, actually it was done at the site. He has continued to raise questions about this repair that was made to the pipe, in the case here the weld repair; and so we thought we would give him the opportunity to discuss this matter with you.

However, before Mr. Halapatz puts on his presentation, staff member Mr. Campbell wants to make a statement in this area.

As you will recognize, at this point in time, TVA has completed its evaluation, it has concluded what was done was satisfactory. The INE inspection has reviewed this matter and also concluded it is satisfactory.

MR. MILLS: I would like Jerry Ballantine, Plant Superintendent, to expand on the reason why we felt it was necessary to put it back in its original configuration.

MR. BALLANTINE: I am Jerry Ballantine, Superintendent of Sequoyah.

When this occurred, we had just a small kink in the pipe, as shown with the weld repair. That up at the top of the pressurizer where all of those lines are located is a very confined space. The reason we put it back was simply that it was much easier to restore it to its original configuration.

So, where all the hangers fit, where all of our analysis fit, it was to redesign the hangers and reanalyze the pipe. The repair was much the simpler approach.

DR. ZUDANS: Sir, why would you have to redesign the hangers? If I understand the picture in this, and the discussion, correctly, you had a pipe support at the location 33; that pipe support failed to slide when the pressurizer expanded, and as a consequence the kink was formed, I guess, through the other side where "weld repair" is writiten, right?

MR. BALANTINE: Yes, sir. The reason we would have to resdesign and relocate the pipe supports was that the pipe was not in the geometry that it should be to be supported by the hangers.

DR. ZUDANS: How much was it out of geometry?

MR.DILWORTH: In the analysis we ran, all the stress analysis that was run on the system was different.

DR. ZUDANS: How much was it out of position, half an inch, two inches?

MR. BALLANTINE: It was only out a very slight amount, but it was out --

MR. ZUDANS: How much?

MR. MERRICK: Three-quarters or something like that.

MR. BALLANTINE: Between a half an inch or three-quarters of an inch.

VOICE: We can get that answer exactly.

DR. ZUDANS: If it is in the order of magnitude of a quarter of an inch, are you going to tell me every piece and every anchor and every dimension you have in a plant that you analyzed, that if fits exactly to a 1/100th of an inch the dimensions you used in the compilations?

What is your tolerance?

MR. BALLANTINE: We have some design tolerances that our analyses are based on, and this was bent sufficiently that it was outside those tolerances. Our choice was to reanalyze and possibly redesign th anchor system or get the pipe back within those tolerances that we had put upon ourselves. George?

DR. ZUDANS: I have no qualms about the statement you made. I just want to express my opinion, if you would go back in the plant and use some nonexisting instrumentation that would allow you to measure with a high precision, you would find out that you are more than the amount you are discussing now away from the drawings as installed.

There is just no way to build such a system with the precision you are talking about.

MR. MARK: But I guess at this stage the question -- which we don't want to take up at this moment -- has to do with the efficacy of the atte pted repair.

MR. STAHLE: That is correct. The matter of the minority opinion deals with the repair and not the technique or the rationale.

MR. MARK: You suggest we come to that a little later?

MR. STAHLE: We are going to discuss it now. I have a statement the staff member would like to make, a continuing analysis.

MR. MARK: This is the place for that?

MR. STAHLE: We would like to do that if we may. We will have staff make a statement on this and then Mr. Halapatz will, I think, provide his presentation.

MR. CAMPBELL: My name is Ronald Campbell. I work for the Materials Engineering Branch, Division of Engineering, NRC.

I would like to read a memorandum addressed to Howard D. Denton, Director of NRR, from Richard H. Vollmer, Director of the Division of Engineering.

The subject of the memorandum is "Differing Professional Opinion Concerning Repair of the Pressurized Relief Pipe in Sequoyah Unit One." And that memorandum reads as follows:

"This is to advise you of the status of the resolution of the differing professional opinions submitted by Mr. Halapatz of the Materials Engineering Branch concerning the subject line repair.

"As background, the pressurizer line was bent during the hot functional testing of Sequoyah Unit One last year when a pipe support failed to yield and the system heated up. As the result of this, the pipe was deformed.

"TVA proceeded to perform a repair on the pipe, which consisted of grooving the pipe in two places and inducing a shrinkage by weld repair. As a result, the pipe was straightened.

"This repair had been given extensive review by NRR and INE and as a result of two principal concerns raised by Mr. Halapatz, number one, he feels that it is not possible to conclusively demonstrate by the evidence in hand that the pipe was not penetrated during the repair operation. If the pipe had been penetrated, the code would require that a hydro testing system be performed.

"Item Two: He is concerned that the welding repair has since sensitized that region of the pipe to make it suspectible to intergraining or stress corrosion cracking. He feels that this could threaten the integrity of the primary system during the plant operation.

"Although NRR and INE have concluded that this repair is acceptable from a code standpoint and does not pose a safety problem, Mr. Halapatz expressed a driving professional opinion by the enclosed memorandum dated June 16, 1980.

"In an effort to resolve this differing opinion, I have met with my staff and Mr. Halapatz separately on two occasions. Based on these meetings and on a review of the information available, I have tentatively concluded that any possible safety concerns would be satisfactorily addressed if an augmented in-service inspection program, as represented by Mr. Newman, is implemented.

"I believe this would make a failure of this pipe as or more unlikely than any other location in the primary system pressure boundary.

"In order to supplement my judgment on this problem, however, I have recently asked Larry Shaw of Research to perform a peer review of the issues leading to this differing professional opinion and give me a peer recommendation on appropriate reactions. I have asked Mr. Shaw to complete this review by July 11, so that I can give you my final recommendations promptly and in accordance with the Commission procedures on resolution of differing professional opinions.

"The end date for your resolution is July 16th, 30 days from Mr. Halaptaz' original memorandum on the subject. If additional time is required either by peer review or by your review, an additional 15 days is allowed in these proceedings.

"I would be happy to arrange for a briefing on the issues at our convenience."

The memorandum is signed "Richard A. Vollmer, Director, Division of Engineering."

MR. STAHLE: If there are no comments, I would suggest we proceed and have Mr. Halapatz present his opinions and provide you the benefit of both sides.

MR. MARK: Shaw has undertaken to perform this review of his. His conclusions are not yet available?

MR. STAHLE: Yes, that is correct.

MR. HALAPATZ: My name is Joel Halapatz and I am assigned to the Materials Engineering Branchj of the Division of Engineering, Office of Nuclear Reactor Regulation.

You have been made aware of my definition, my expression of a differing professional opinion, in the matter of the adequacy of Sequoyah Unit One, weld draw bead repair of the pressurizer relief pipe.

You have also heard the statement that the adequacy of the repair draw bead repair has been acknowledged as acceptable.

The majority opinion acceptance of the weld draw bead repair, in my opinion, is not based on the guidelines which exist today in current licensing requirements. The repair in questions is subject to the jurisdiction of section 11 of the ASME Code, which states that "After repairs by welding on the pressure boundary, a system pressure test shall be performed in accordance with IWA-5000."

The code allows for the exemption, the following exemption: That piping, pump, and valve repairs that do not penetrate through the pressure boundary are exempted from hydrostatic testing. Therein lies a minor deficiency in the majority's acceptance of the weld draw bead repair.

Further, the NRC majority by virtue of a memorandum dated September 21st, 1979, Rubinstein to Paris, directed that TVA institute third party inspection of the Sequoyah Nuclear Power Plant.

In the acceptance of the weld draw bead repair it has not been estabished that, one, the repair did not penetrate the pressure boundary as determined by a third party, which was required by the Nuclear Reactor Commission.

It must be demonstrated at this point in time, and demonstrated and determined by a third party, that, one, the repair weld did not penetrate the pressure boundary. In the event that pressure boundary was penetrated, then it is incumbent that the pressure test, the hydrostatic pressure test, be performed.

If it is determined by the third party inspector that the pressure boundary was not penetrated, then follows the second question of whether the determination has to be made by the third party again, whether the extent to which the material has been sensitized. These are the two issues which were identified in the memorandum, in the Vollmer to Denton memorandum.

At this point in time the acceptability of the Sequoyah Unit One pressurizer relief pipe repair has not been in accordance with the current licensing requirements.

Now, during the investigation of this matter attention was paid to a mockup which was made by TVA which at one time was presumed to represent the production repair. The majority's consultant viewed the mockup, examined metalographic samples which were prepared from this mockup, and on the basis of his examination determined that the weld repa was, in fact, acceptable.

As minority, I challenged that opinion and I was given the opportunity to examine the mockup myself. I visited Knoxville and had some sections cut from the mockup and examined them metalographically. These may not be visible back here to you people, but what I did was simply take sections, took photomicrographs from the ID to the OD, the OD being here; here is the weld here (indicating). This is the ID.

I proceeded from the weld fusion line out to where I fettered out the sensitization. Now, he technique that is used is polarized light. The light areas here at the grain boundaries are sensitized, precipitated carbide. This is the OD here; here is the weld fusion line; here is the weld here (indicating).

What I see here is evidence of through-wall sensitization. This through-wall sensitization represents a potential crack path. We are concerned, as far as the reactor coolant boundary, we are concerned with crack paths through the wall. This is what I observed. As it later developed in the course or additional discussion and controversy over the matter, the majority opinion, the majority opinion stated as of last week, was that the mockup did not, in fact, represent the production repair.

Now, if that be the case, section 11 of the Code commits that "prior to authorizing repairs by welding, the owner shall conduct an evaluation of the suitability of the welding procedures to be used to make the repair. The evaluation should consider cause of failure to assure that the selected repair procedures is suitable. Repair programs shall be subject to review by the enforcement and reguatory authorities having jurisdiction at the plantsite."

So, the repair was completed after the fact. The examination was completed after the fact. The repair was already made, and yet when we examined this mockup we found not a very desirable situation.

Okay, so this mockup now is decreed not to represent the production weld. It is now intended merely to demonstrate this weld draw bead repair will straighten the pipe.

What we have today, we have no evidence in hand of what is in that production pipe. There has been, as the result of some of the controversy, there was an in situ metallographic examinatioin performed. In situ metallography is extremely difficult to do. There was an attempt to -- I might add, before I pass on here, that this mockup was fully penetrated. The exemption from hydrostatic testing is based on the provision that the weld repair should not penetrate the pressure boundary.

The mockup did penetrate the pressure boundary but in itself was enough to disqualify the exemption to hydrostatic testing.

In the in situ controversy it was compromised that in situ metallography would be performed as metallography on the actual pipe and that radiography would be used to determine whether or not the pipe was fully penetrated.

That is what has to be done. Number one, it has to be demonstrated that the pipe is not fully penetrated. If it is, then you have to hydrostatically test it. As far as the sensitization bit goes, the in situ metallography was an attempt to establish the degree of sensitization so as to get a handle on whether or not a potential crack path existed in the pipe.

Inasmuch as third party inspection, which was invoked by the NRC, was not conducted, there is a question about the validity of the acceptance of the weld repair. It is as simple as that.

My position is very simple, that I think what we have to do is, we have to comply with current NRC licensing requirements. We should hae third party inspection to determine, number one, whether the pressure boundary has been penetrated and, number two, to determine the degree of sensitization so that we can get a feeling.

I made the recommendation in correspondence relating to the minority opinion that intergranular corrosion testing be performed on through-wall specimens, specimens which would expose potential crack paths to the aggressive environment, whatever it may be.

The environment that this material will see is 2. ppm oxygen bearing steam, not .RR5 oxygen bearing water, a big difference.

We have little, if any, data available on corrosion of sensitized stainless steel, oxanific stainless steel, in oxygen-bearing steam. One could -- and this is being done -- one could seek extrapolation from other data. Somehow, as a minority I don't feel too comfortable with it, in view of the consequences of failure. There is some disagreement about what actually are the consequences of failure. The nonconformance report that was submitted, and which caused the necessity, which provided the necessity, for the repair, cited that failure could result in uncontrolled blowdown of the reactor cooling system.

This is in very serious contrast with the majority opinion, that should failure occur, one would observe only a small leak, which could be detected by in-service inspection.

I think that this is a significant issue which should be resolved.

I emphasizeed that the matter could be very readily resolved by third party inspection. The NRC commenced TVA to third party inspection. This inspection should be performed. The determination of the third party, I think, should be the binding determination. Right now I don't think we have complied, or there has been compliance, in this matter, with the current licensing requirements.

Are there any questions?

DR. ZUDANS: Yes. Assuming that this third party inspection will confirm what you suspect, that the weld has penetrated the wall, and there is a certain degree of sensitization, what is next?

MR. HALAPATZ: I think the only alternative would be to cut the pipe out. I think it would be too big a gamble to take, given the pronouncements of pipe crack study groups, I question --

DR. ZUDANS: Okay. Supposing you do that, what would make the following inspection of a new piece of pipe in the weld different from this one?

MR. HALAPATZ: Okay, a good point. The technique that was used to effect the repair --

MR. HALAPATZ: Herein lies a very serious criticism of the technique. Two grooves were grouped in the wall opposite the kink. These grooves extended 270 degrees. The depth of the grooves, from what I could determine from the data, was 2T/3 which was two-thirds T.

This is a six-inch schedule 160 pipe with a phenomenal .2718 wall. So this groove was two-thirds of that. We are talking a half inch penetration of the groove. The way the draw was made, these grooves were filled with weld metal and this underlying base metal experienced a sensitization range.

Then the weld metal was ground out and the grooves were filled again with weld metal. Again exposing, again adding an additional time exposure to the sensitization range. Now when one equates the metallurgical experience that these particular weld deposits and underlying base metal have seen with an installation weld, they may not be the same. They probably are not the same.

The argument is made, well, how are these any different than repair welds?

If we knew the history of those repair welds, then we could make that comparison. Today we cannot. We know the history of these repairs or these weld grooves. But given any weld in that system, we cannot tell you what happened to it. We do not know metallographically what that base material saw. We do not know the extent to which it is sensitized. We do not even know th extent to which this is sensitized really.

Because all we have tried to look at is outside. And we have tried to get an idea of how much sensitization was there.

DR. ZUDANS: Well, if you cut the pipe out in the same or adjacent locations, you would have to make a full penetration weld all around.

MR. HALAPATZ: That is right.

DR. ZUDANS: And what I fail to see is if you think that there is no way of telling what history this particular material saw in this weld, which as I assume under controlled conditions, how could you expect more information from the weld through the pipe? I do not see that.

MR. HALAPATZ: Well, I make a recommendation that -- there are 18 pieces of pipe of the same heat which were bought on the purchase order that was installed here, 18 pieces. What I would recommend -- I recommend originally take a vote sample. Cut a section out of this, just cut a section out, take it to the lab where you can examine it real carefully and get a good handle on exactly what you have there.

There was disagreement there and it is a valid disagreement. You could possibly add more sensitization. So the point I make is given that we have this uncertainty about this repair weld, we cannot get a good enough fix on it to leave us with a comfortable feeling and have this plant go into operation.

The point I make is let us get some pipe, the same heat number, mock it up exactly the way you welded this to this production repair, then cut it up, take it in the lab, cut it up, examine it, and perform some trans-wall test, intergranular corrosion tests.

Take sections here, expose them to the copper sulfide solution. Better, expose them to the environment this material is going to see in service: .2PPM oxygen bearing steam. That will give you an answer. You know, you cannot isolate this leak. Should you get a leak here, you are not going to be able to isolate it.

This is why the nonconformance report made the statement: A nonisolatable leak, okay, uncontrolled blow-down of the reactor cooling system. This I think is what has to be done to demonstrate that this repair is safe. Somewhere along the line we need the data.

DR. ZUDANS: Well, is it the actual repair history is recorded precisely enough that you are fairly comfortable with the feeling that that experience can be repeated on that piece of pipe?

MR. HALAPATZ: I do not know that it is acceptable but I think you will get a reasonable simulation. That is the purpose of mock-ups, you know. I think you would get a reasonable simulation. It will tell us, it will give us a pretty fair idea of what we have in the ---

DR. ZUDANS: Why was the mock-up test declared invalid?

MR. HALAPATZ: Well, Dr. Zudans, there has been much controversy, and I might add perhaps a little bit of acrimony associated with discussions. And in my latest memo in which I respond to the majority opinions stated I find it very conclusive that the mock-up that was made originally was intended to represent the production piece.

The fact that it is now declared otherwise ---

DR. ZUDANS: Okay, now your findings in the mock-up are such that the weld penetrated the boundary?

MR. HALAPATZ: That is right.

DR. ZUDANS: As far as sensitization is concerned, it is not in a dangerous range?

MR. HALAPATZ: No, I would be very reluctant to put a piece of material like that in service given the trans-wall hydro structure.

DR. ZUDANS: Okay, so on those counts you feel the mock-up indicates there is a problem in this area?

MR. HALAPATZ: That is why I raised the difference.

DR. ZUDANS: Has anybody else commented why the mock-up was declared invalid?

MR. HALAPATZ: Well I can cite you the references memoranda. It shows -- okay, I quote now from the memorandum Pelikey to Newman, dated June 27, 1980: "The mock-up used by TVA was not intended to duplicate either the pipe material or the welding procedure used in straightening of the pressurizer relief pipe. TVA informed us on March 13, 1980" -- quite a while after the thing was in production -- "that the mock-up material was 304 stainless steel while the pipe material is 316, which is normally more resistant to intergranular stress corrosion tracking caused by weld sensitization than 304.

"Furthermore, the heat input used to make the single weld in the mock-up was six times higher than that used on the Sequoyah relief pipe, leading of course to more severe sensitization of mock-up material. The primary purpose of the mock-up was to experiment with the effectiveness of the straightening process itself, and very little can be deduced about the condition of the pressurizer relief pipe from the examination of the mock-up."

DR. ZUDANS: Well that sounds ver. reasonable.

MR. HALAPATZ: It does sound reasonable, and that may well be, but what this says is right now we have no evidence in hand that would justify our conclusively accepting the repair.

DR. ZUDANS: If I may remark, I feel that it would be not a bad idea to mock up another piece of the very same material, and just run through that experience. And that has nothing to do with when -- and the time element is not that critical.

So if you have a piece of pipe made of the same material, you might as well do it. I, however, believe that the difference between this weld, the weld -- what do you call it? A complicated name technique, the difference between this and the real world in the plant is not great.

MR. HALAPATZ: Well you know, Doctor, the possibility exists that some of the welds that are in the real plant are not too good either.

DR. ZUDANS: Now we are opening up a totally different question. If you say that, then you really are in trouble. I guess we are in trouble.

MR. MARK: Your point is if the pipe were cut and the section welded in, you cannot see why one would be any more ---

DR. ZUDANS: Why would it be any better than this one, because the weld was done under the same controlled conditions as it would be done. That would be full penetration weld. And what this would do in addition to already what this did, affects the other pieces now.

Have you examined nozzles in the ---

MR. HALAPATZ: What is being overlooked here is the fact there were two welds leaving those grooves. What happened was the groove was first filled with weld metal. Then it straightened out a little bit. Then the weld metal that was deposited was ground out and then more weld metal was put in there, you see?

So the base material here experienced the welding environment twice: a full penetration weld.

DR. ZUDANS: I would assume they would have ground out all the heat affected zone, not just the weld.

MR. HALAPATZ: I do not have that assurance.

DR. ZUDANS: Don't they have records of that?

MR. HALAPATZ: The way it was done, I do not think it possible to confirm they did in fact grind out ---

DR. ZUDANS: In other words, you do not believe they can confirm convincingly?

MR. HALAPATZ: That is right.

DR. ZUDANS: I think it would be a good practice to do that.

MR. MILLS: May I make a comment for TVA? I want to add a little to this. You know, this is not a question we heard for the first time today. And probably that weld here has been studied and locked at both by the NRC staff and TVA experts more than any weld in this country.

Mr. Halapatz did come down to TVA. We met with him at our Signalton Laboratories where these mock-ups were done. I think that we supplied all information that we had at that time. As the result of that, and he uses the word "we" several times, I think it would be more appropriate to use the word "I." We did have agreement up here for Mr. Halapatz's concerns.

We agreed to go back and do some etching to show we did not have overlap between those welds of sensitization of the steel. We agreed to do additional radiographs. All of this was done with NRC personnel observing.

At that time when we outlined that small program Mr. Halapatz and the rest of the staff stated that they believed that that would conclusively prove one way or the other whether or not that was a satisfactory weld. It is my understanding that all the NRC staff, with the exception of Mr. Halapatz, was completely satisfied after we went back, took the insulation out, and did those radiographs from different angles.

I wanted to make sure that was in the record.

MR. HALAPATZ: Okay, also for the record I would like to include that I did request to see the glossies of the metallography that was performed. I had available to me xeroxed photographs of the shots that were taken.

I, as a metallurgist, would be extremely reluctant to pass judgment on what anybody can determine from those. I see smeared ·tal. And to be honest about it, as I mentioned, it is an extremely difficult job to do in situ metallogr ·y. It is tough. I have done it so I do not talk too lightly about it. It is a touch job.

But to make any determinations and to draw the conclusions that were drawn, I certainly do not -- I cannot concur. I see smeared metal here. As far as the radiography goes that was to confirm that the wall was not penetrated, it is my judgment that the technique that was used does not have the capability to determine whether or not that wall has been penetrated.

So that is why the solution is simple. NRC has directed that third party inspection be performed.

Let's do third party inspection. I do not understand why it was not done before. On September 29th, 1979 this was the direction from the NRC. If you are interested, Dr. Zudans, just as a matter ---

DR. ZUDANS: I read your memo.

MR. HALAPATZ: You did not look at the pictures there, Doctor. These are the photographs ---

DR. ZUDANS: While you are walking, why wouldn't the TVA provide you with actual shots?

MR. HALAPATZ: I have no idea.

DR. ZUDANS: Can you answer that question?

MR. MILLS: Those shots have been provided to INE Inspection Region 2 in Atlanta and also that gentleman observed, the INE gentleman in Atlanta observed the entire sequence of past radiographs we did in Sequoyah.

DR. ZUDANS: That was not my question. I said why Dr. Halapatz was not provided?

MR. MILLS: I cannot answer that question.

DR. ZUDANS: In other words, they are accessible to him if he wants to see them?

MR. MILLS: We were directed to deal through Inspection and Enforcement in Region 2 on this problem. We supplied Region 2 with all information they asked for. I cannot say why Mr. Halapatz does not have these glossies. I do not know.

MR. MARK: When were those pictures available in Atlanta? Is this very recent or months ago?

MR. MILLS: No, back in January, February, the January/February timeframe, before we loaded fuel -- no, wait a minute, I am sorry, fuel was loaded.

MR. MERRICK: March 13th.

MR. MILLS: It was in March. The core was already loaded but it was -- that was practically all done. We had loaded the core.

DR. ZUDANS: So you could see them if you would want to.

MR. HALAPATZ: Doctor, during metallographic work, a glossie is not the best thing to do. Xerox is worse.

DR. ZUDANS: No, actual shots. Film.

MR. HALAPATZ: Well, you can tell a little more from the glossies than from the xerox. But you can best tell when you actually eyeball the film.

DR. ZUDANS: That is right. I am talking about the actual film. If you do not see the actual film, you cannot be assured you do not see a movie production. I wouldn't believe other things than the actual negatives. But TVA, is it so difficult to mock up that six-inch of pipe and go through the procedures and satisfy yourself? Because you see there is a question raised about sensitization.

I am not so concerned about weld complete penetration of the boundary because the hydro test won't tell anything. We will take the hydro test without any trouble. But my concern would be about sensitization.

But if you provide a potential for a crack path, I think you should feel at ease yourselves to mock it up and test it, that little piece of paper. It should not be such an economic penalty.

MR. MILLS: I am not a metallurgist. We do not have the metallurgist with us today to discuss any details, although it has been discussed with the staff in more than one meeting at great length. And I think that when you say

TVA ought to be satisfied, TVA is completely satisfied with the adequacy of this weld.

We think we have proven it to ourselves. We think that the staff members, with the exception of Mr. Halapatz, are completely satisfied. This is one of those things that we can continue on for years and years. However, we believe the problem is solved.

DR. ZUDANS: Isn't the solution much simpler by going and mocking up that actual steel piece of pipe, welding it, cutting it up, and showing that if you follow the procedure by welding, grinding it up and rewelding, that you do not at all sensitize.

Because you cannot go back to the pipe and cut a piece off there without further damage.

MR. MILLS: I think you might have a different idea if you saw the mock-up we actually did.

DR. ZUDANS: I understand it is different material.

MR. MERRICK: It is 304.

DR. ZUDANS: I understand also it is overly sensitized.

MR. MERRICK: The A-262 test, which is the reject test for the pipe.

MR. HALAPATZ: I want to take issue there. The 262 does not tell you very much. You can weasel around 262.

MR. MERRICK: That is what you used up there.

That is practice.

MR. HALAPATZ: Practice A, but this shows me to the eyeball that you have a potential crack path. I can see the grain boundary carbide here. Now had you tested a trans-wall specimen, 262 doesn't tell you what the specimen orientation should be.

But what good does it do you to test just exposing the IL surface? 262 doesn't tell you that. We are interested in cracks propagating through that wall. So you test a through-wall specimen.

MR. MERRICK: You test where the crack is going to initiate, which is at the ID sample.

MR. HALAPATZ: If a crack initiates at the ID and stays there 40 years, good, you have no problems. The problems start whenever you get a crack growing and propagating through that wall. That is what you want to check. This is where, you know, sometimes the thinking gets a little fuzzy.

So this is why I deliberately -- when I went down to Knoxville, Tennessee, when I looked at that mock-up, that is what I looked at.

MR. DILWORTH: George Dilworth, TVA. I would like to say I do not think it would be appropriate for our staff here to debate Mr. Halapatz on this issue on a point by point basis. However, we stand ready to reply answers to any specific questions of ACRS. But we have debated this, as Mr. Mills said, time after time and after time and I do not believe a debate between us and him would be appropriate at this time.

MR. MARK: I think ---

MR. STAHLE: Mr. Chairman, it may be advisable at this point to have Mr. Wessman from INE just make a statement with regards to the inspection that was done and give you the benefit from all parties' points of view. So Mr. Wessman will read a statement.

MR. WESSMAN: I am Dick Wessman from the Office of Inspection and Enforcement. I have in my hand here a copy of the inspection report that was filed. I have in my hand a copy of the inspection report that was filed by the metallurgists from the Region 2 office after they did their on-site inspection and in the period of March 17 through 20 of this spring.

The inspector did look at TVA's radiographic film. He did examine the activities idn the field. He then asked TVA to do some additional radiographs and looked at those radiographs as well.

His final conclusion, and as stated in the report, was that no weld melt-through areas or defects were noted.

And I have a copy of this entire inspection report, which I will be glad to give you if you desire to take a look at it.

MR. HALAPATZ: In rebuttal I wish to make a statement that the fact that you do not see any nuggets does not mean that you haven't penetrated that wall. You can have a fully penetrated weld, which will not show up on a film.

In fact, a technique that was used -- I would be very very apprehensive about drawing any conclusions about it from it. Permit me to go to the board and let me ---

MR. MARK: Could I ask, was there any comment in the report that was just referenced about sensitization?

MR. WESSMAN: Sorry, sir, I do not recall. Let me look at it.

MR. HALAPATZ: That is a special test.

There was something in the report that they did not see any sensitization, which is very anomalous. I have never seen a piece of stainless steel that was welded that did not show some sensitization. A comment.

The way the radiography was performed was it is a 90-degree shot, it is a double wall, the film is here. They used a number 12 penetrameter. And if you take a look at what your film will show you, you will see this crown of the weld first, okay? You will see that.

Now what you are looking for -- let's assume that weld is not fully penetrated -- what you are looking for is the discrimination, a density different which will show you that here I have a different film density here than I have out here and, therefore, I can conclude that that weld was not fully penetrated.

Now I made the suggestion before the radiographer, re-radiography was performed and I said "Hey look, you have a fully penetrated weld at TVA. It is there. Why don't you shoot that and use that as a base line to see if the technique that you are going to use is capable of getting you this film density difference so that you can conclude whether or not that weld is fully penetrated?"

It did not go over too well, you see.

Okay, so given this technique I challenge, and this is what I included in my minority opinion, I challenged that you can draw that conclusion from the film that was made. That is why I say let a third party inspect, let a third party look at that radiograph. Let him make an unbiased objective opinion. That is it.

DR. ZUDANS: Well, sir, you see what makes me feel uneasy is that the third party is certainly not going to perform any miracles. If you couldn't see anything and an inspector couldn't see anything, the third party would automatically reach the same conclusion.

MR. HALAPATZ: Well, I asked to see the films ---

DR. ZUDANS: I think that what you stated before, if you had a through-the-wall specimen and did all the testing that was necessary, then you would be positive. And therefore one of your recommendations, namely the one to mock it up again on the 316 and do that, may be called research -- I don't know -- but the research of some future value.

I don't know the piping is unsafe or anything like it. I don't think you would learn anything in the hydro test.

MR. HALAPATZ: Well the hydro test, it is a technicality the Code commits you to. It is there.

DR. ZUDANS: Well, you already ran it at the full pressure, so what is the drawing in 15 percent? If you have a size crack that would leak, you would have detected it on your radiograph.

MR. HALAPATZ: But one of the conditions of the acceptance of the repair ---

DR. ZUDANS: I understand. That is a formality. The real question is is the material sensitized enough so that you have a crack path?

MR. HALAPATZ: I make the strong point that intergranular corrosion testing should be performed on through-wall specimens because that is the way your crack is going to propagate.

MR. MARK: Could I ask, not for discussion, but just check whether this statement is approximately correct, you have felt that procedures in one or another way have not been fully complied with. I believe I got the notion had there been a hydrostatic test and nothing had happend, you would then have said procedures had been followed and one could sign off, although my own feeling is that doesn't really address the point that is potentially worrisome.

MR. HALAPATZ: Permit me, sir, to give you some of the chronology of ---

MR. MARK: No, please, I think not.

Would a hydrostatic test in your view have allowed the question to be dropped?

MR. HALAPATZ: No.

MR. MARK: I certainly myself do not feel it would have addressed the the crucial point.

MR. HALAPATZ: In fact I raised this question with my immediate management. I first became aware the repair had been made through a memorandum written by a colleague in which TVA came in and asked for approval of the exemption from hydrostatic testing.

I observed it was a weld draw-beat repair that was made. I happened to be involved earlier in my career with repairs of this type that proved to be very, very troublesome. So I alerted my immediate management, Mr. Gambel, to this fact. He became involved in it.

The approval, the memorandum which approved this repair was withdrawn, was rescinded. And thereon developed the final, well, what led up to this discussion here today. I would not have bought it off even if it had been hydro ---

MR. MARK: When did this exchange take place approximately?

MR. HALAPATZ: Okay, this memo was dated December 4, 1979.

MR. MARK: Okay. Now the other thing, you say that if there were a third party inspection you think that would at least meet the formalities of the situation, although as Dr. Zudans said, he isn't quite sure what one could expect from that.

MR. HALAPATZ: Well, given the controversy which has developed, I stand as the lone minority. But I think I stand on solid ground. I think that an objective third party could give an objective determination. I think that there are people that have the capabili ies to examine problems of this type and to come up with a judgment that I think is a respected judgment, respected by both parties.

MR. MARK: Okay, if there were such an examination and it concluded that the majority opinion was the one to follow, that would not necessarily persuade you but at least compel your concurrence?

MR. HALAPATZ: I stand here as a professional individual. I have been involved in work of this type. I did not gain my background simply from reading. I have been exposed. I have crawled around pipes and did in situ metallography before I came t the NRC. I recognize what a problem it is.

I recognize the uncertainties that are related. That is why I simply make the point that given the consequences of failure, let's take that one extra look. That is all I say.

MR. MARK: Now the other thing, Dr. Zudans has raised the question, which seems obvious enough, why one should not examine a weld under as nearly similar conditions as one could devise and for heavens sake in the same material and in fact in the same heat in the same material I believe it ought to be.

That would give as good evidence as one could imagine to be obtained and it might either reassure one that the sensitization zone was closely restricted or not. And then I guess there is a curious question, which I do not think we want to do more than observe the existence of it, but it is a little hard to understand why, when the question came up, it wasn't automatic that you were shown the most revealing film to look at. If TVA is asked to deal through an office, then I can understand that is what they feel they should do. They should not go around a site.

But then that office might have said the film is public property and should be looked at by anybody knowledgeable, and I cannot understand why that wouldn't have been automatic.

As it ought to be clear and realized by everybody, the ACRS subcommittee at the very least regards metallurgy as distinctively mysterious and in situ metallurgy up in some high point of the ceiling on a bunch of pipes as even more mysterious. But the fact that it is complicated is certainly well realized.

I do not know that we can carry this any further.

MR. WESSMAN: Mr. Chairman, let me given an answer to your earlier question on sensitization if I may. In reading back into the report, the conclusion was made after the field examination by the inspector that there was no sensitization evidence.

DR. ZUDANS: How did he come to that conclusion? Is it stated in the report?

MR. WESSMAN: Yes, it is. I am not a metallurgist and I can only read the words to you. It was an examination with a field metallurgical microscope and they reached this conclusion after this field examination.

MR. HALAPATZ: Can I rebut that? Are there any furter questions? Okay.

MR. STAHLE: And that concludes the review of the non-TMI issue that was presented.

MR. MARK: Well, I am not quite sure what our next step will be, speaking of the question of the weld repair.

DR. ZUDANS: Mr. Chairman, may I make an observation?

MR. MARK: Please.

DR. ZUDANS: I think that between the two choices of having third party inspection by using the documentation as it exists and not extracting through-the-wall specimens versus a new mock-up, I would certainly prefer the new mock-up where you can take the specimen through the wall. I do not think I would be very pleased to drill holes and take specimen out of this wall and do further damage to it.

So I do not think that the third party inspection means anything. And I do not think that peer review means anything. It is just another opinion. The only thing that we are interested in, is this material sensitized or not?

And the only way you can do it is either to cut out the piece from the pipe or reproduce the piece of pipe outside of that installation. These are the only two choices in my opinion.

MR. MARK: Well, I think we probably have to leave this question.

DR. ZUDANS: Yes, just for the record.

MR. MARK: --- just at that point for the record. Whether this will become a matter that requires discussion before the full committee I guess I am not sure. It is going to depend upon how we progress through the other items on the agenda.

I think we will not probably be able to complete the consideration of the case without again having discussion of this point. But as I say, that doesn't mean that we go to the full committee with it if for other reasons we cannot make progress there. Please go on.

Mr. Stahle. You had mentioned two new items of the non-TMI issues.

MR. STAHLE: So that concludes the items ---

MR. MARK: I guess we would just have to say -- I lost my record of those things temporarily.

The steam generator cork cracking is either resolved or resolvable because the camera either satisfies people or ---

MR. STAHLE: Otherwise they would have put in inspection reports. So I would consider that item is closed or completed.

MR. MARK: This item we cannot say as closed.

MR. STAHLE: We would have to identify it as open pending deliberations.

MR. MARK: That sort of gives us the picture on the non-TMI list. Then would you go on?

MR. STAHLE: We will move onto the agenda item which 22(c), which is the status of TMI open items. What I plan to do over the next few moments is to provide an overview with respect to these items, a brief overview. Please note on item three, this gets into the implementation of the TMI items.

So we will look at it in two parts, though I will briefly show the viewgraph and then come back to them so we can go with each item with more detail, including TVA's participation on these items if you so desire.

Let me start then with a brief introduction to emphasize again, as I did at the last meeting, first with respect to the structure of the SER supplement number one. At that point, we had dealt, the first section had dealt with a scope of review, but was normal to NRR review based on requirements that are described in our standard review plan.

The second portion of our SER supplement number one dealt with the lessons learned from the TMI-2 accident but only dealing with the fuel requirements. For those requirements that had to be met prior to issuing the current license to allow them to load fuel and to conduct the load power test program.

Supplement number two, as envisioned, of course is one that would deal with the remaining non-TMI issues that I have just discussed; and, secondly, it will now deal with the so-called full-power requirements. In essence these will be based on the new REG 0-694 entitled "TMI Related Requirements for New Operating Licenses."

One aspect I should stress in this document in how we proceed is in the studies that went on and proceeded into the Action Plan on requiremens, it devised a categorization of requirements in order that we may implement the near-term OLs.

And first and foremost was the definition of what constituted fuel load requirements; and secondly, full power requirements; and then a third category described as NRC type actions; and fourth, data requirements. Data requirements are included in this new REG 0-694.

From the point of view of a project manager I have looked at these issues as being ones that have to be addressed in a number of ways. Most certainly those are issues that need to be reviewed prior to full power, they must be fully addressed in the upcoming SCR, whereas dated items have to be reviewed from not only a technical point of view but from an admdinistrative point of view as to how we will deal with this in the license, the legal instrument that will permit us to go to full power.

This matter then from my point of view, I have treated those items that I have to address in one manner or another, and I have included them in my viewgraphs. It adds a degree of complexity here, but I think nevertheless these are the types of requirements that have to be addressed either immediately or in the very near future.

Let me provide then the overview and then we will discuss this a little later. There are about 37 items that I filed as not only full power requirements but data requirements. From the first of 37 items I did identify about eight that I felt are near completion.

I must revise this chart a little bit. By the way, I should identify it. For simplicity I elimi ated or deleted from this chart the identifications that are used in new REG 0-694 as well as our other documents, namely, the Arabic identification to these items for simplicity purposes in this review.

Items actually two and seven are not complete -- sorry, items two and four are not complete at this moment. They are satisfactory with regards to their review in the conduct of a load power test program. They will be complete shortly in the next two weeks through a visit to the site and further reviews of the information that we have in hand.

And then item eight has one or two open items that will have to be closed out.

But nevertheless, excepting this, I regard these items as about complete as far as the status report.

Now let me quickly, and recall now we will come back to each and every one of these items to discuss implementation, from the listing of the remaining 29 items we have here that were identified in the new REG document, the list looks rather formidable but let me quickly note several things about this list.

First and foremost there are 15 dated items in this listing: items, most of which are requirements that occur in January of 1981. One of the items involves rulemaking. And item 29, the last of which deals with power ascension. This constitutes at least 15 items out of the 29 remaining not demanding immediate review or in depth review to these items.

The other thing I would like to note from this chart here is items 21 and 24 and 25 actually that deal with the whole subject matter of emergency preparedness. These three items are under a special group under intensive review at this moment. I am confident that this month we will see the completion of these reviews and acceptance of the emergency perparedness plan on the part of TVA as well as state and local plans.

The targeted date again is to be in a position to say that emergency preparedness plans and issues are completed by the first week in August.

The third note I would want to identify in this overall listing here in the overview is the hydrogen control question. You will hear today as a separate discussion on a, if you will, on a resolution of this matter. And I will view this in considerable detail. So from this list we could say the item on hydrogen control is probably resolved.

DR. ZUDANS: Which number is that?

MR. STAHLE: Hydrogen control, number 11. I have been informed also in the area of auxiliary feed-water evaluation, item 14, that is resolved pending a small item. I am pleased to report as of today I think the safety evaluation report is written subject to a few minor comments. That can also be placed in the completion column.

And last but not least, the one item on control room habitability is near completion. Simply a letter of confirmation is due. So this item itself now approximately this week can be put into the completion column.

So as an overview to what we have I believe this is doable within the remaining two to three weeks in July with an expeditious review by the staff and a quick turn-around upon the applicant's part: all of which I understand material will be in this week and will therefore provide the staff hopefully the time to complete its safety evaluation report.

I would suggest at this point we defer discussions of each specific item, since all of this will be reviewed under item three of your agenda and that point we are talking about.

MR. CATTON: I do not see pressure vessel (?) Levasent are anywhere. TVA did make a commitment t 'esign one I believe.

MR. STAHLE: It is really a dated item, sir.

MR. CATTON: Thank you.

MR. STAHLE: I suggest then we go on to the next agenda item and then come back to item three, if this is acceptable.

This next one I believe TVA will discuss the design features for protection against floods.

MR. BURZYNSKI: My name is Mark Burzynski, staff engineer, TVA. And I am going to discuss the flood protection plan for Sequoyah. Starting out I would like to give you a brief discussion on the different flood modes for the Sequoyah plant.

There are two, a rainfall flood and a seismic flood. TVA views both of these as rather incredible events. But in following the guidance of Regulatory Guide 1.95 and looking at the total TVA watershed, we come up with some incredible floods.

The rainfall flood, as noted here, involves a storm over the entire watershed, over 20,000 square miles with an average rainfall in three days of 16 inches. There is a particular pattern to get the flood. You have to have a three-day storm followed by three days of non-rain, and then the major storm following for another three days.

The seismic flood, the worst of which is the failure of four dams coincident with one half the probable maximum flood. There are other combinations of these two modes that exceed plant grade, but these are the largest and the fastest rising.

The flood protection plan ---

MR. MARK: Is that comparable maximum flood something for which there is a nationwide accepted definition given by the Corps of Engineers or somebody or is it something you invented?

MR. BURZYNSKI: No, this is developed from the regulatory guide. It involves rainfall patterns and a certain amount of precipitation and a given watershed.

The flood protection plan is a two-staged plan designed to, number one, limit economic loss in the event that the flood does not fully develop. That is the stage one portion; and number two, a more severe set of actions to protect the plant if such an incredible event should occur.

The operation of the plant in the flood mode has been designed for 100 days post flood, although the flood waters should recede in six days or less in probable maximum flood.

We did no consider the combination of events with this probable maximum flood; that is, LOCA or steam-line breaks or a site specific earthquake.

At the state one plan there are easily reversible actions: calling for additional personnel, shutting down the plant, vorating to a rather severe shutdown margin, moving supplies, distributing equipment, and making some load adjustments to the diesel generator system.

These load adjustments are to really provide for sufficient lighting for the flood mode operation.

DR. ZUDANS: Can I ask a question? When is this stage one to be implemented? On what signal?

MR. BURZYNSKI: That is coming up.

MR. MARK: Are you also going to say how long it would take at an easy pace to go through that?

MR. BURZYNSKI: Yes.

The second stage are the more severe actions. TVA has designed this flood scheme in light of the fact that we believe that it is a rather incredible event. And we allow for some rather unusual means to mitigate the effects of the flood.

As noted here, a high pressure fire protection system would replace the auxiliary feed-water system. We have prefabricated spool pieces that would connect the submersible fire pumps to the steam generators. And that would provide for cooling.

We would also make some changes to the component cooling water system and the ERCW system to absorb the heat loads from the one system, once again using prefabricated spool pieces. The liquid rad waste tanks would be filled to prevent floatation.

DR. ZUDANS: How long would it take to fill that liquid rad waste tank?

MR. BURZYNSKI: Not long, hours.

DR. ZUDANS: How fast would the flood rise?

MR. BURZYNSKI: In my next slide I will discuss those aspects.

DR. ZUDANS: I am just wondering whether they will float away before you could fill them.

MR. BURZYNSKI: No.

What I have termed here nonessential loads would be de-energized. These are the full set of ECCS pumps, the high-head, intermediate head, and the low-head containment spray pumps. Most of the plant auxiliary equipment would be de-energized in this stage.

The batteries that are below the design basis flood elevation would be disconnected. These are not the vital batteries. The vital batteries are above flood elevation and would remain in operation with the charges operable.

We would also seal the drains in the diesel generator building and the emergency raw cooling water building. These buildings are designed for minimum leakage. There is a sump pump provided to keep them fully operational.

The important point in all of this is to be able to predict a flood with sufficient time to implement the flood protection plan.

TVA in evaluating the different flood combinations, both rainfall and seismic, had performed sensitivity studies on the arrival times. We have done that in the rainfall floods by looking at the rainfall pattern and putting the heaviest rainfall on the first day of the storm, the second day, the third day; and looking at which combination gives you the fastest arrival time.

We have also moved the storm itself around the Tennessee Valley area to find the worst location both from elevation and from arrival time. We have evaluated all combinations of dam failures and coincident with flooding by moving an earthquake around to get multiple dam failures and have evaluated that both for elevation and for arrival time.

And we came up with 27 hours after we have made calculations and predictions. Our trigger levels give us 31 hours: four hours to make confirmatory calculations and provide for communication times to give the plant at least 27 hours to implement the flood plan.

And as I have noted here, stage one takes 10 hours, stage two takes 14 hours, and we provided a three-hour contingency in there. There is only one combination that gets us to this time. In all other cases we have upwards of 50 to 60 hours to implement those flood protections from the time that a trigger point is exceeded.

Our Division of Water management is the noticing agency. They have been in this business now since TVA has been in existence as to where we all started. And they have an extensive watershed model that has been verified by recent flooding in the past years.

We feel confident that we can predict the flood and give adequate warning time.

DR. ZUDANS: One question. When you started out your flood scenarios, I understood in the beginning there were two distinct or discreet modes: one is rainfall and the other seismic.

MR. BURZYNSKI: There are two distinct modes. The slide listed the worst of both modes.

DR. ZUDANS: In other words, you get by rainfall the probable maximum flood level and then the dams break, is that what you are telling me; that gives you the worst one?

MR. BURZYNSKI: No, let me go back to that slide.

The rainfall flood just involves a lot of rain. As noted here, 17 inches of rain averaged over 20,000 square miles, but it is patterned based on rainfall data, with some areas getting as high as 29 and 30 inches, okay?

That much rain on the ground runs off and will cause dam failures as the result of flooding.

The second mode involves half that much water falling to give you pool elevations that are high and then a well-placed and welld-time earthquake that knocks out four dams.

DR. ZUDANS: At this point now, when would you begin implementing your flood stages in the second case?

MR. BURZYNSKI: In the second case? In all cases in seismic floods we have more than 31 hours available after a dam has failed.

DR. ZUDANS: In other words, they are far away enough so that there is nothing?

MR. BURZYNSKI: Yes. Any other questions?

MR. MARK: Just from curiosity, you have sort of batted down the hatches except for diesel generators running. The water is now such that the plant supervisor has to row to work. How long can the diesel generators run under those conditions?

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MR. BURZYNSKI: We have stored on site approximately 30 days worth of fuel.

MR. MARK: But does that take with tank trucks running around to get the fuel to where you need it?

MR. BURZYNSKI: No, those are storage tanks, from a storage tank.

MR. MARK: Are connectable?

MR. ZUDANS: Your plant exit doors are not flooded now.

MR. BURZYNSKI: Oh, for some part of the time the plant would be inaccessible.

MR. MARK: The water will be above grade at the plant?

MR. BURZYNSKI: Yes. That is the whole reason why we have got a flood plan. We got 30 days worth of diesel fuel and the floods recede in 6 days or less.

MR. ZUDANS: So you do have to row to work, right?

MR. BURZYNSKI: Well, if you notice on your Stage 1 --

MR. MARK: If Sequoyah Dam gives out I think they are really safe again, except that their homes are all flooded.

MR. BURZYNSKI: Yes. We are nice people, and as you notice here the first thing we do is call everybody to the plant.

(Laughter.)

I should add that there is a Stage 3 warning, and the flood plan there calls for unleashing the ark and we just go --

MR. MATHIS: What is the water elevation at the plant under these maximum conditions?

MR. BURZYNSKI: Okay, maximum conditions is 720 feet mean sea level. Plant grade is 705. And as a conservative measure we have also included wind-induced waves to a tune of 50 mile an hour winds blowing on the plant, and it gives us another four and a half feet of elevation, and that is our design flood -- 726 feet.

MR. MATHIS: How much of Chattanooga is under water on that basis?

MR. BURZYNSKI: That would probably give you 40 to 45 feet of water in downtown Chattanooga. My office is on the fifth floor.

(Laughter.)

SPEAKER: Since when?

MR. MARK: Walter?

MR. LIPINSKI: You mentioned this 720 feet mean sea level. Are there any openings in the plant that are above the 720 feet that are critical, such that if you rise above 720 you do get flooding within the plant?

MR. BURZYNSKI: In the flooding of the plant, as I mentioned, the diesel generator building and the ERCW building has penetrations and openings below 726, but they are designed to be to minimize leakage, and some pumps are provided to keep thos buildings operational.

The reactor building remains dry. The service building, auxiliary building, control building and turbine building all flood.

MR. ZUDANS: I have another question. Let's forget about this external flood. What kind of a flood level is postulated inside the containment not due to natural sources, if any?

In other words, it might happen that you leak some water and start filling the containment, like for example in Three Mile Island, and what kind of a water level does the Sequoyah plant consider in design?

MR. BURZYNSKI: Inside the containment building we have considered reactor coolant system values, the ice and the fueling water storage tank and all the accumulated volumes for the water level inside containment.

MR. ZUDANS: How high does it go?

MR. MATHIS: 20 feet.

MR. BURZYNSKI: Yes, somewhere between 13 and 20 feet, depending on --

MR. MILLS: Approximately 19 feet is what we --

MR. BURZYNSKI: Depending on whether you use the minimum or the maximum calculation.

Any other questions?

MR. MARK: Mr. Stahl, do you have a comment on this topic? Is the staff over the situation and decided that it is well intended to?

MR. JOHNSON: My name is Ted Johnson, staff hydrologist. The staff has looked at the plans, and we find that basically those times are correct. We made independent checks of the times that are available and also the levels. We go into that.

MR. MARK: And does that include your judgment that the plant is well set up to cope with even such a level?

MR. JOHNSON: Someone else would have to answer to the ability of the plant, some people from the auxiliary systems I believe would be appropriate.

VOICE: Mr. Parks will answer that.

MR. PYLE: My name is Owen Parks from Auxiliary Systems Branch. We have looked at the results of the water reaching a height as they discussed here today. Of course as engineers we shudder somewhat to think of losing equipment, but the equipments that are provided by the technical backup we consider acceptable.

MR. MARK: I think that covers that question then. If that is all on the matter of floods, I think before going to take up the new item we might --

MR. SIESS: One question.

MR. MARK: Yes, sir.

MR. SIESS: Can I assume that TVA's emergency procedures include warnings to the people of Chattanooga?

5 MR. MARK: I thought Walter Cronkite would do that.

(Laughter.)

I don't know, do you want an answer to that, Chet?

I think we will declare a 10-minute break before taking up the item on the Single/Two Unit Set. A quarter to eleven.

(Recess.)

MR. MARK: We will resume. I will call on the staff to continue with the next agenda item, (e) I guess, or 2(e).

SPEAKER: Yes, TVA will provide a five-minute briefing on this item.

MR. MILLS: I will call upon Mr. Don Williams of our engineering design organization to address this.

MR. WILLIAMS: I am Don Williams with the TVA's Engineering Branch. In the last ACRS subcommittee meeting Mr. Ebersole asked us to provide a list of major design features or design changes that would be implemented prior to U2 operation, things that may require consideration, additional consideration.

We reviewed the plant design, and there are three major design changes which I want to give you a brief discussion on. There is a new ERCW pumping station and new natural draft cooling towers, and number 3, interim auxiliary building secondary containment enclosure.

Now items 1 and 2 are related, and they are permanent design features of the plant. Item 3 is a temporary feature that will be removed when Unit 2 goes into operation.

Let me give you some of the history of the first two items. In our original criteria for the Sequoyah plant design we expected that the states I believe, or the State of Tennessee would accept our proposed river water temperature rise of 10 degrees Fahrenheit and maximum river water temperature of 93 degrees Fahrenheit.

However, with the passage in 1972 of federal Water Pollution Control Act, we found ourselves constricted to a maximum river temperature rise of about five and a half degrees Fahrenheit and a maximum river temperature of 87 degrees, approximately.

Therefore we were required to revise our service water heat dissipation facility to provide more capacity.

After considerable review we decided on adding new natural draft cooling towers. The reasons these cooling towers primarily were added is that our design study showed that for a two unit operation under worst type conditions, the summertime, there were some couple of months where the river water temperature would be exceeded by our discharge.

Therefore, we have now provided natural draft cooling towers for open helper enclosed cycle operation.

Now the design of these cooling towers is such that they discharge into the condenser water intake pumping station.

Now this pumping station is the location of our present ERCW pumping station.

Now our design studies also showed that under certain conditions, maximum temperature conditions, the discharge from the natural draft cooling towers would exceed the design temperature of the ERCW pump.

Therefore, another design review was undertaken, and the decision was made to install or to build a brand new ERCW pumping station to cover both units.

The pumping station is located at the end of the intake skimmer wall and is located there to relieve ourselves of being dependent on the effluent from the cooling tower.

As for the third item, the interim auxiliary building secondary containment enclosure, the enclosure itself is that part of the auxiliary building which is defined in our plant SAR, which provides an effective barrier to hold in airborne contaminants in the event that such contaminants are released to the auxiliary building during an abnormal occurrence.

Now this enclosure encompasses the ventilation systems which are designed for, which are intertied from both units, and a number of rooms in the auxiliary building. With the difference in construction schedules between Unit 1 and Unit 2 we could not assure ourselves that all of the Unit 2 features would be properly sealed and isolated. Therefore, an interim boundary was drawn up and with this interim boundary the doors and penetrations received the additional sealing required to prevent any leakage or to minimize leakage of airborne contaminants. And the ventilation systems were provided with isolation dampers, which closed, received the same signal to close that start, I believe, the auxiliary building gas treatment system.

When Unit 2 is ready for operation, the entire enclosure will be preop tested to verify that the enclosure is functional, that it will maintain a negative pressure in the auxiliary building. At that time the sealing, additional sealing that has been required for the interim boundary will be left with the isolation dampers and the ventilation systems will be locked open. They will remain there -- some may be removed, but those that are not removeable will be locked open and de-energized.

And these are the key design features that Mr. Ebersole was interested in.

MR. ZUDANS: Do you have any kind of a sketch that shows how this secondary enclosure is? Is it a separate building or is it just the same --

MR. WILLIAMS: No.

MR. ZUDANS: -- building but separate compartments in the --

MR. WILLIAMS: It is in the auxiliary building itself.

MR. ZUDANS: Separate compartments.

Why do you call it containment?

MR. WILLIAMS: Secondary containment.

MR. ZUDANS: It does not contain the auxiliary building. It is part of the auxiliary building, right?

MR. WILLIAMS: Part of the auxiliary building, yes.

MR. ZUDANS: I see, except it might provide the boundary that you can close and seal off?

MR. WILLIAMS: Yes, in the event that airborne contaminants are released into the auxiliary building.

MR. ZUDANS: And would you normally keep the auxiliary building -- -- this is only in the event of the accident?

MR. WILLIAMS: In the event of the accident.

MR. ZUDANS: Normally it is kept on the atmospheric pressure, right?

MR. WILLIAMS: I believe so, yes.

MR. MARK: Are there any other questions of Mr. Williams?

Are there any comments on these plans as viewed by the staff?

MR. STAHLE: No comments. Item 3 of course has been thoroughly reviewed by the staff as far as the interim measures necessary to isolate Unit 2 from Unit 1, and these are satisfactory and have been reported in our SER supplement.

MR. MARK: So there is no continuing need for discussion on this point?

MR. STAHLE: No, sir, not to my knowledge.

MR. MARK: Thank you, Mr. Williams.

In that case we will at least have a comment on the, I guess it is the plans for the low power test.

MR. STAHLE: With regards to the Low Power Test Program, particularly the Safety Evaluation Report, Mr. Bear, the team leader, in the past has reported to the committee on our status and has reported the results, at least substance of our results on the reviews that have been ongoing for several months.

We are now at the stage which I will categorize as fine-tuning the safety evaluation report with respect to the low power test program, and I anticipate it being complete, final if you will, in the next day or two.

The safety evaluation report covers several aspects of the program, the safety aspects as far as the tests themselves. Procedures were reviewed carefully to assure that the program can be carried out. The exemptions, a waiver if you will, of Tech Spec requirements that were needed to carry out the program were carefully reviewed and determined that indeed these waivers would not introduce any additional hazards to the public.

In addition to that, we have gone through these procedures on the simulator and assure that the procedures are

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practical, workable.

Last but not least we will tie it into this program the extent necessary the emergency procedures that may be needed in the unlikely event that something should occur during the program.

The conclusion, I think that we stated before, at least tentative, with Bob Bear, is the same; namely, that this program can be carried out without risk to the public.

This document is an important document. It becomes part of the license, amendment to the license that we need in view of the changes in the Tech Specs that we required.

So consistent with the schedule I mentioned before, in anticipation of the initiation of the low power test program on Saturday, we plan to issue the amendment, the report and so forth, consistent with that date. Thursday or Friday is our targeted date, which I am quite optimistic in meeting.

This report will be available to you, if you wish, as a separate item. It certainly will be included as part of our Safety Evaluation Report, Supplement No. 2.

I have no further comments other than to mention the report is rather detailed, and I think it should when it is available to you answer your questions.

MR. ZUDANS: This low power test program includes all those natural circulation tests in different modes?

MR. STAHLE: Yes, sir. The low power test program

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is basically the natural circulation tests that have been described on numerous occasions.

MR. MARK: It seems unnecessary to go through the list again. It has not changed as far as I am aware.

MR. STAHLE: No, no, I just wanted to make sure -- no, sir.

MR. MARK: There are ten tests, I believe, listed and given a lot of discussion.

MR. STAHLE: Yes, sir.

MR. ZUDANS: I just happen to see that you start it next Saturday, right? That is what you said?

MR. STAHLE: Saturday is our goal, or TVA and ours.

MR. ZUDANS: You might see the results someday.

MR. STAHLE: They will be published.

MR. ZUDANS: Nothing matches full-scale tests.

MR. STAHLE: Excuse me?

MR. ZUDANS: Nothing matches a full-scale test.

MR. TEDESCO: Mr. Chairman, I might add -- Tedesco from the staff -- that we had approved North Anna last week, and the North Anna has been running these tests.

MR. MARK: Oh, have they?

MR. TEDESCO: The past several days, yes. And there were some minor adjustments but the tests have been going along very well.

MR. MARK: Mr. Mills, do you have comments on this

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general topic?

MR. MILLS: No, sir, that is pretty well in line with our schedule. As I think I stated before, we would expect to commence those tests this weekend, on Saturday. We would expect that they would be completed prior to the end of the month.

MR. MARK: Walter.

MR. LIPINSKI: At our last meeting it became apparent that your station blackout test was a simulated test and you had agreed at this meeting to itemize all those features that would not be included as part of your simulation that would normally be encountered during a true station blackout.

Have you done so?

MR. MILLS: Yes. We are ready to respond to that. Would you like for us to do that at this time?

MR. LIPINSKI: I think we should.

SPEAKER: It seems like an opportune time to do so.

MR. MARK: Please.

MR. MILLS: All right, I will ask Joe Bynum from our Sequoyah plant to discuss this for us.

MR. BYNUM: My name is Joe Bynum, TVA's Sequoyah plant staff. We did respond to Mr. Ebersole's question. I think the response has not been received by the ACRS yet.

What I would like to do is just briefly read over the five items that we identified in our response. The following systems which are operable during Special Test 7 would not be operable in a loss of all AC power condition. This list is not all-inclusive, but it does consider those major systems which could affect plant conditions.

The first item I have is component cooling systems supplying cooling water to the reactor coolant pump, thermal barriers, and the normal auxiliary building loads.

The second item i reactor coolant pump seal injection flow from the chemical volume control system with normal charging being isolated.

The third item is all containment ventilation systems, including upper and lower compartment coolers, control rod drive mechanism coolers, along with their emergency raw cooling water supplies, the control air system supply to all plant equipment with the exception of the turbine-driven auxiliary feedwater pump, control valves -- these are the valves which regulate the flow to the steam generators -- the steam generator power-operated relief valves. All other valves which are supplied by the control air system will be positioned to simulate the loss of all AC power. All auxiliary building ventilation which would supply areas where heat loads would not exist on the loss of all AC power conditions. For example, the CVCS pump rooms, et cetera. Those areas which would carry the heat load during an actual loss of all AC power will have their ventilation shut off during the test. For example, the turbine-driven auxiliary feedwater pumps.

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So that essentially what we are doing is we are taking that equipment which is affected and which is required, reliance is required, for the mitigation continuous of natural circulation.

Then we do have that equipment selectively de-energized. Auxiliary feedwater pumps, power-operated relief valves -- those are the primary pieces of equipment. So we have taken the heat loads and taken the ventilation systems off in those areas, lighting off in those areas.

The containment upper and lower compartment coolers will remain normal.

I can provide this list. I will give this to Dick and he can make copies of it. It will be sent, you know, as a formal reply.

MR. MARK: Do you have comments on that, Walter?

MR. LIPINSKI: No, I was just going to ask whether we will receive this written material, and that was just answered.

(Pause.)

MR. MARK: Mr. Mills, do you have anything you would be anxious to put in at this point?

MR. MILLS: Dr. Mark, I believe we have responded as the items occurred. I don't think we have any additional comments unless there is questions of us.

MR. MARK: If there were, you could add them in after

Mr. Stahle has gone through the list that he is about to proceed with.

MR. STAHLE: The first review is an overview of where we stood on these items trying to categorize them between the complete and not complete and identify it for your benefit, particularly in the chart that showed 29 issues. At least 13 of them were data-type items. One was a -- and a number of other aspects reducing the, at first review, an extremely heavy workload of items to be reviewed that are prerequisites to issuing a full power license.

I would like to go back now then to the listing. Both ourselves, staff and TVA can comment as we proceed through these items. They are an attempt to respond to each area, one of the items identified in NUREG 0694. Some are quite small, uncompleted; for instance, item 1 here. Indeed, there is a reactor inspector at Sequoyah, and therefore, we comply with this item. A matter of incidental.

Items 2, 3 and 4, and particularly 2 and 4 that I identified on emergency procedures, I indicated they were not quite complete. They were satisfactory for the low power test program but not complete for full power in spite of identification in the charts. Brent Clayton is here tc say maybe a few words on the status -- if he is there -- Brett, would you cover then the items in the action plan, are really identified as, I guess, the C1, 1(c)(7), 1(c)(1) and 1(c)(8)?

MR. CLAYTON: I am Brent Clayton of the NRC staff. We have, well to go back, I guess, item 1(c)(1) requires that we look at some emergency procedures for the plant: procedure for LOCA, inadequate core cooling, and some other accidents to transients which are undefined.

The staff has determined that for the near-term OL's that there is other accidents to transients that we are going to look at, concentrate on: steam generator tube rupture and the loss of main feedwater.

We have completed a preliminary review of these procedures, and we think that they are adequate for the low power operation.

We are in the midst of our final review, and we are working with TVA. We are going to be meeting with them next week to discuss the procedures. We are going to be going down to the simulator and observing their operators walk through those procedures, and then we are going into their control room the following week if things work out, and want to have their operators walk through at least one of those procedures in the control room.

I think that we will be completed our review prior to the time that will be required for a full power license.

MR. STAHLE: Fine. By my shot that covers items 2 and 4. Item 3 is, we already have in hand the NSSS response to the procedures, so that requirement is also closed out.

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Let me move on to the next item. On 5 you have already heard. That was the low power test program. So this has been satisfactorily completed as a result of -- now we will be publishing our safety evaluation report.

Plant shielding is the next item indicated here as complete. Indeed, I do have a safety evaluation report. The staff has reviewed the material provided. The identification here, the Sequoyah radiation and shielding design reviews, use source terms and criteria as contained in the Regulatory Guides 1417.

The second, information, Documents 14844 and General Design Criteria 19.

Sequoyah designed so that access is not required outside of the main control room for 30 days after accident, except for limited access for the shutdown board room and structures away from the main components.

Our review goes on to indicate that the radiation and plant shielding design described by the applicant meets our position in the NUREG Document 4578 and therefore are acceptable. This will be in our SER. It is a completed item.

Item 7 is completed. An SER is available on the pressurizer, heated supply. We find what TVA has designed is quite adequate and therefore this item is closed.

Item 8, a draft safety evaluation report that identifies one or two items that are open. I think I would like

Mr. Stater to address this briefly, provide you a status report.

MR. STATER: Bill Stater, NRC staff. We have reviewed the submittal of Sequoyah -- -- radiation sources outside containment, which should include the waste gas system. The procedures so far provided by Sequoyah do not include the waste gas system as one of its areas of review.

In addition, we have requested the results of the tests of the liquid systems. We have not as yet received the results of those tests.

MR. STAHLE: TVA may wish to respond at this point to this item.

MR. MILLS: Could you repeat that again? We are a little fuzzy about what you are talking about here.

MR. STATER: The procedures that were submitted for this item for the radiation sources outside containment cover leak testing of several liquid systems containing primary coolant.

NUREG 0578 identified in this group of systems that the waste gas system, including the waste gas headers, be leak tested. We have not seen the procedures for that, and that item has been transmitted to TVA as an open item.

Similarly, we have requested that the preliminary tests made of the liquid systems, that the results of those tests be provided to us. We have not seen those results.

MR. BALLENTINE: Yes, I can respond to that. I am

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Jerry Ballentine. Those tests have been completed, and if you haven't seen the results I think it is just a matter of the paper mill. But we have completed the tests.

The procedure for leak testing of waste gas system, we have prepared it and we have performed the tests. The results are favorable and we should have those to you shortly.

MR. MARK: Do you anticipate problems here apart from the fact that you do need to see these submissions?

MR. STATER: No, we don't, but we do need to have them to complete our review.

MR. MARK: Walter.

MR. LIPINSKI: I would like to go back to the pressurizer heater power on that previous viewgraph. The power to the heaters was one question with respect to TMI, but there are also level controls that inhibit power from being applied to those heaters if the level is too low in the pressurizer.

The power supply and the level control is not safety grade. Could you address this with respect to item 7 as to whether there is manual override with this emergency power or is it still subject to the failure of the level controls?

MR. TEDESCO: On that particular one there is a requirement that you have emergency power available at the pressurizer level too. That is a requirement.

MR. LIPINSKI: Okay, so the level and the heaters themselves have emergency power.

MR. TEDESCO: And we want emergency power to the level and to the heaters as part of the -- -- that is required.

MR. MARK: Anything else on this list of eight items?

MR. STAHLE: Does TVA have any comments on that? Okay. The first item on the list is a dated item, and no comments at this point are necessary other than a recognizing here that it does exist. The staff has no comments. TVA may wish to make a comment on the shift technical adviser and the requirements for January of 1981.

MR. MILLS: We do have a problem with item number 9, the post-accident sampling, in meeting the date of January 1981, and our continuing discussions with NRC staff with regard to that.

And item number 22, the near-site EOF, we have a different proposal than the requirements call for on that, which we are discussing with the staff or have already discussed and are continuing discussions with those people with regard to the nearness to the site.

MR. CATTON: What is the problem with the post-accident sample?

MR. DILWORTH: It is a matter of construction time on being able to implement the system that we already have procured. At this point we have bought the Century system, and it will be delivered -- the equipment, most of it, will be delivered in late fall. And we just don't believe that we are going to have enough construction time to complete it by 1-1-81.

And also we are looking at this time at the possibility of another system that might be able to go in quicker, a real-time on-line post-accident sampling system.

So we really feel like that the 1-1-81 is probably not achievable. But within six months of that time we feel pretty firmly that we could get it done.

MR. CATTON: Could you remind me what post-accident sampling includes?

MR. DILWORTH: Well, that is the sampling of the reactor coolant water, the containment atmosphere, several other samples, of course that are highly radioactive. There is some discussion going on in the industry now whether or not you required a grab sample that could be removed and taken offsite or whether or not you can use a system that would analyze it immediately inline. And we are trying to resolve that.

So this is the reason that I feel a little bit like we might not be able to make 1-1-81. We are attempting to make that date, but it is shaky.

MR. CATTON: What about item 19? That is inadequate core cooling instruments.

MR. DILWORTH: We intend to have that done by 1-1-81.

MR. CATTON: Do you have any documentation on your level sensor that I could have?

MR. DILWORTH: Excuse me, I didn't hear that question.

MR. CATTON: Level sensor, how are you going to do it?

MR. DILWORTH: Yes, we are. We do have documentation, and we can provide it.

MR. CATTON: Very good, thank you.

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MR. LIPINSKI: On item eight, reactor coolant system vents, where will you lead the vents? Will they just dump into containment?

MR. STAHLE: Let me -- one aspect of this area I shouldn't note here. It was identified as proprietary in nature. It does restrict what we can say. I advised TVA of that fact here. We would be pleased if the staff and TVA maybe to discuss that under a separate session. But I am just advising you at this point that Westinghous[illegible] [illegible]VA has stamped the schematic and some of the information as proprietary. I haven't looked into this any further than to respect the labeling and therefore will have to caution a certain amount of discussion here.

If Westinghouse would li[illegible] to comment on its own material, I guess that would be acceptable.

MR. DILWORTH: Westinghouse says they will provide it.

MR. STAHLE: Very good.

MR. LIPINSKI: Okay, another question applies to item 16, containment dedicated penetration. This is specifically for the hydrogen case, the dedicated penetration -- although it doesn't say it up there, but from the action plan that is what the dedicated penetration was related to. Where inside containment does the penetration go?

MR. DILWORTH: That does not apply to our ice condenser plant.

MR. STAHLE: Yeah, I should add that they have internal combiners, so that is not applicable.

MR. LIPINSKI: Well, then, please amplify what item 16 is, then?

MR. STAHLE: Jim, do you want to comment on the item 16, on dedicated penetration and --

MR. PULCIFER: Jim Pulcifer, NRC staff. That's a requirement for dedicated penetrations for operation of recombiners or purging of the containment for hydrogen control. In this case, Sequoyah has internal recombiners and there's no reason or there's no need for penetrations through the containment to perform the recombination. So this does not apply to them.

MR. STAHLE: I will eliminate that item from the list. This list identifies all the items, and I am in error in including it on this. I had forgotten at the moment that they had internal combiners. But I have been informed of this, so it will be deleted as being not applicable to Sequoyah.

MR. DILWORTH: Someone asked the question a minute ago about where do the reactor vessel vents vent to. One goes to the reactor coolant drain tank, and one goes into the pressurizer relief tank, and the other goes into the containment atmosphere. There are two paths for that vent.

MR. MARK: Does that give you what you wanted?

MR. LIPINSKI: Yes.

MR. ZUDANS: What is anticipated to be done on item 13?

MR. STAHLE: This is a dated retirement. I think this is a program that TVA can best address at this point in time.

MR. LAMBERT: Item 13 is the IPRI relief valve test program, which TVA is an active participant in with respect to this docket, to Sequoyah docket. And our latest information with respect to that program is that it's on schedule both with respect to review submittals and test activities.

MR. ZUDANS: But it's not going to be done at Sequoyah now?

MR. LAMBERT: Oh, no. No, this is a multi-facility, multi-scale generic test program. All of our related valve and design information has been submitted to the IPRI-NSAC team that is defining the program and conducting the program, and we have been assured that we are covered under that program umbrella, our test.

MR. ZUDANS: You are just a partner to it, correct?

MR. LAMBERT: That's correct.

MR. STAHLE: I'd like the staff member to comment on items two, three, four, as I understand it, five, and 16 as it relates to operator training, licensing; this is a matter of course of interest. And Mr. Buzzi from the staff, I think, will, at least, provide you a brief status report of our review at this time.

Mr. Buzzi?

MR. BUZZI: I'm Joseph Buzzi, NRC, operator licensing branch.

Items two -- which involved immediate upgrade of (ACRONYM UNINTELLIGIBLE) qualifications involved the facility to increase the requirements as far as experience, as far as (WORD UNINTELLIGIBLE) experience, and supervisor experience, as far as the licensing (WORDS UNINTELLIGIBLE). Licenses that were issued for TVA early this year. In addition to that, the Commission required additional experienced personnel to be present for core loading and low-power operation.

CHAIRMAN MARK: Could you speak into the microphone. We can't hear you.

MR. BUZZI: Item two further required compliance training programs with --

MR. ZUDANS: There's something wrong with your machine.

(Pause for adjustments)

MR. BUZZI: Okay, I guess I'm on now. The immediate upgrade of SRO and RO qualifications required additional experience for senior operators, particular, for both conventional and time involved in nuclear power plants prior to licensing. TVA has already been licensed; the licenses were issued earlier this year. The qualifications were -- also required additional training in thermodynamics, hydraulics, and plant transients.

We have already examined 18 and licensed 18 operator seniors and five operators for TVA.

The second -- the third requirement, administration of

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training programs for licensing exams, involved that the instructors who teach systems, integrated responses, transients, et cetera, shall have completed an SRO examination. TVA has four instructors, all of whom are SRO qualified.

Item four involved the revised scope criteria for normal licensing examinations. This involved upgrading the score requirements 80 percent overall and 70 percent minimum in the category. All these items have been completed, and -- as far as the previous license is concerned. We intend to provide, give examinations to nine additional seniors next week. These are Watts Bar people who will be assigned to the TVA -- to the Sequoyah operating plant -- operating staff.

Item five involves a revised scope for criteria for simulator exams. This is an overall -- this was a requirement that will involve the plants in the long term; however, the short-term items that TVA does have a simulator and we have been using TVA's Sequoyah simulator for all our examinations.

The item -- item 12, the degraded core, I'm not sure -- degraded core training, TVA has started a program on degraded core; however, the NRC issued a letter to all licensees March 28th requiring additional training and outlining the training program for TVA -- for all licensees. TVA has not yet responded to the -- to the requirements. They are supposed to be respond by August 1st.

MR. CATTON: I have a question on six, procedure for verification of correct performance of operator activities. In the letter from TVA dated October 30, 1979, they indicate effects of the following: operator's failure to act when required; operator's inappropriate accident -- actions during an accident; additional failures in selected system operations.

How are they going to establish the realistic assumptions that they claim are needed in those areas, to fulfill item six?

MR. BUZZI: I can't address that particular item.

MR. STAHLE: Can maybe TVA be -- respond to that? It is unresolved at this moment from the staff's point of view. TVA may --

MR. CATTON: If I can read a little bit, it says, "The purpose of this action is to improve the performance of reactor operators during transient and accident conditions."

The reason I ask the question is, TVA has a simulator and with the use of the simulator some of these questions can be addressed. I've raised the question every opportunity, and yet I find that there is no relationship between TVA's safety people and the TVA operations people who run the simulator. There does not seem to be -- at least, there's no apparent feedback between the two.

MR. BALLENTINE: I'm not sure that -- I'm Jerry Ballentine -- I'm not sure that I understand fully your question. We do have a coupling between our -- the plant performance and our simulator. And we have taken the opportunity on any operation of the plant to provide our simulator people with actual plant performance during pre-operational testing and hot functional, initial critical, each of the evolutions that we have gone through with the plant, wherein we could provide actual plant performance to our simulator to confirm the simulation, we have done so.

Is this --

MR. CATTON: That's -- let me read the question first. The question that was asked -- that TVA was addressing is: "Discuss all possible plant transients in which operator action may increase the consequences of the transients." Part of the response indicated that the things you needed to do were to address things such as operator's failure to act when required, operators's inappropriate actions during a transient, additional failures, selected system operations.

And what you're telling me is that you train them well. Well, I believe that. But you have a simulator and you have an opportunity to answer these questions. How does the operator foul up when he's running the simulator will in part answer these questions.

And as I understand it, the backup computer codes have built into them a record-keeping process where you can begin to record the types of errors they make, where they make them, and so forth.

I have asked this question before. I don't -- I hear the same answer every time, that we are programming the simulator to represent the plant. I understand that. But what you're not doing is keeping a record of what the operators do wrong, in order to build it back into your approach, and to complete item six.

MR. BALLENTINE: I think that -- I think we have the --

MR. CATTON: Am I being clear?

MR. BALLENTINE: -- the answer to your question, but I am unable to give it.

We need our simulator staff, someone from there, I think, to fully answer that.

MR. ZUDANS: It seems like they would be doing that naturally.

MR. CATTON: Well, they're not.

MR. ZUDANS: They're not.

MR. CATTON: Not that I know of, anyway.

I think the problem is, you've got two separate groups -- you've got operations and you've got engineering; and they're somehow --

You've got a third group, too? Training. Well, training comes under operations.

(Pause)

MR. MILLS: We, we would be -- this is Larry Mills -- we would be happy to bring the, you know, the people in charge of our simulator up at the full committee to respond to that question. We -- we don't have any of our simulator people here. But we do feel like we have a very close connection between the operations and the simulator. They report to the same people. It is not a matter of isolating those people or any such thing as that. There's a very close connection between them. And we're --

MR. ZUDANS: It's not a question of isolation. The question is very simple. Are you keeping records --

MR. MILLS: That's the reason --

MR. ZUDANS: -- of errors that people make on the simulator?

MR. MILLS: Well, that's the reason that I have to tell you that I'll be glad to respond to it at the full committee, when we have our simulator people in charge of our simulator up here.

MR. ZUDANS: Okay.

MR. MILLS: I can't answer that question today.

CHAIRMAN MARK: It's possible that that answer would clear off your questions about item six.

MR. CATTON: Yes.

CHAIRMAN MARK: It's possible it might not.

MR. CATTON: That's correct also.

MR. LIPINSKI: On the subject of additional training in the area of thermal hydraulics, I assume that the Sequoyah training manuals were written before the thermal hydraulic question arose, as requiring additional training.

MR. BUZZI: That's correct.

MR. LIPINSKI: They then wrote an additional section to the training manual to cover this material --

MR. BUZZI: I believe so.

MR. LIPINSKI: -- from a theoretical standpoint.

MR. BUZZI: And also applied. Also applied.

MR. LIPINSKI: Okay. Because that was the next question: did this theoretical material then get applied to the various plant systems, with examples to carry the theory over to the particular system, such that the operators would have a better understanding of the theory when they're involved with a specific system?

MR. BUZZI: Well, we administered examinations earlier this year, and I would say, although I was one of the examiners, yes, they have. That would be including both the operating oral exams and written examinations.

MR. LIPINSKI: Because we had reviewed the TMI manual, and they had subcontracted to the university and it gave them a very nice section on thermal hydraulics and that was put into the training manual -- and that's where it stopped. It did not get incorporated into applying that theory to the various systems, which indicates a deficiency in trying to educate an operator as to what he is doing.

MR. BUZZI: I don't think that's the case for Sequoyah.

MR. LIPINSKI: Thank you.

MR. ZUDANS: Could you give us some comments on item 17, what is it all about?

MR. STAHLE: Yes. Mr. Pulcifer I think can comment on that.

Jim, on item 17, isolation dependability?

MR. PULCIFER: Jim Pulcifer, NRC staff. That concerns provisions such as diverse isolation signals to the isolation valve, such as containment high pressure and also safety injection initiation, a diverse number of signals that will initiate containment isolation during an accident. It also concerns dividing systems up into essential and non-essential systems, that is, essential to mitigating the accident, and determining that the non-essential systems are isolated by diverse signals, which was not always the case, for example, at Three Mile Island. It also is concerned somewhat with the containment purge valves and their isolation, to make sure that there are diverse signals on the purge and ventilation valves, including a high radiation in the containment signal to close those valves.

Sequoyah has met our requirements on this item.

MR. ZUDANS: So this entire item is at the level of initiators and signals, and not at the level of the hardware itself, that, in fact, the valves and dampers and whatever else is not part of this consideration? It's just what kind of signals do you provide for isolation?

MR. PULCIFER: I'd say that's correct. This item doesn't specifically address --

MR. ZUDANS: Just the instrumentation.

MR. PULCIFER: -- the hardware, no.

CHAIRMAN MARK: Is it true that 17 would not need to be on a list of incomplete items?

MR. PULCIFER: I'd say that's complete now -- or will be shortly.

CHAIRMAN MARK: A matter of --

MR. STAHLE: We'd comment on that. There is -- I do not have the typical safety evaluation report in hand, but I have been assured in the next few days I will, on the basis there are no problems unresolved from the point of view of technical review. So I'm pleased to say that this item can be identified as completed.

CHAIRMAN MARK: I wonder if we could start from your list --

MR. STAHLE: Go back to the items, I think --

CHAIRMAN MARK: -- and items one through five, I believe, are all --

MR. STAHLE: I think we covered that. We have discussed six.

CHAIRMAN MARK: -- settled, really, except for --

MR. STAHLE: Right.

CHAIRMAN MARK: -- formally stamping them.

MR. STAHLE: Let me go through the items and skip as necessary.

Item seven we have not discussed. And this actually refers to safety evaluation report supplement number one, and identification in that report, there are four items that need resolution at a later date. These are being discussed with the applicant, one of which involves procedures, two of them involve some modifications in the control room, the fourth item is a long-term nature, the staff is reviewing this. At this point in time, from the staff's review, this remains unresolved, although I have been assured that we're doing two things -- identifying the need to close these items out as full-power requirements and identifying others that may be at a later date. Item four is a long-term one.

The SER, again, describes the nature of the four items. I think it wouldn't at this time be advantageous to go through it. But it is open, unresolved, but I'm confident will be in the next day or two.

CHAIRMAN MARK: I think that brings us up -- eight and nine I believe have been --

MR. STAHLE: Yes. Just to comment on eight, again this area involved intensive review resulting in some questions, two pages of additional questions; the applicant is responding to it;

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my understanding, we will have those in hand this week, and I am hopeful that in that two-week period these design aspects of this Bendix system will be resolved.

MR. ZUDANS: Which specific places in the reactor coolant system are vented, specific locations where these vents are provided?

MR. STAHLE: Again I have to cite this: this is proprietary.

MR. ZUDANS: Oh, that could not be proprietary, which. I can't see how it would be.

MR. STAHLE: Maybe TVA can respond to this.

MR. ZUDANS: Is this a reactor head in the highest point, someplace else? What else? How could that be proprietary?

MR. STAHLE: The staff is here. Would you like to comment on that item?

MR. MENDOZA: This is Marvin Mendoza from the NRC staff.

The -- there is a vent location in the upper head, yes.

MR. ZUDANS: Where else? Where else?

MR. MENDOZ: The pressurizer, I believe.

MR. ZUDANS: In the pressurizer?

MR. MENDOZA: Yes.

MR. ZUDANS: In addition to the relief line there is another vent in the pressurizer?

MR. MENDOZA: I believe they plan to use the particulate --

MR. ZUDANS: Any place in the piping, primary coolant system, else? Or these are the only two that you know?

MR. MENDOZA: Those are the only two that there are. Their -- our requirements or criteria require procedures to assure that, for example, in the U-tube there is adequate flow, adequate heat removal.

MR. ZUDANS: You couldn't vent the 3,000 U-tubes, that's right. No.

Isn't there any -- any kind of possibility the pump casing itself could be a high point? I don't --

MR. MENDOZA: The pump casing?

MR. ZUDANS: -- know how -- yeah, the primary coolant pump, how is it arranged, could it have -- no? Low point. I don't have a picture of it.

MR. LIPINSKI: Could we go back to those two vents? The one from the reactor vessel, then, is the one that gets vented into containment? And the one from the pressurizer is the one that goes to the drain tank?

MR. MENDOZA: No, there is provision to go to both places, I believe.

MR. LIPINSKI: From the reactor head?

MR. MENDOZA: From the reactor head, yes.

MR. LIPINSKI: Okay, so there are two in the reactor head, plus --

(Several speak at once)

Well, there's two ways to vent that reactor head --

MR. MENDOZA: Correct.

MR. LIPINSKI: -- plus the pressurizer, then, has its relief valve path for venting.

MR. ZUDANS: Well, I don't think it's two ways to vent the head. It can go in two directions, but it's still one nozzle in a head.

MR. LIPINSKI: Well, there's two paths to dispose of the effluent. That's what I am concerned about, as to where the effluent goes.

MR. ZUDANS: Oh, only where the effluent goes.

MR. MENDOZA: I have been informed that the reactor coolant pumps do have vents on them.

MR. ZUDANS: Aha. Aha.

MR. MENDOZA: I think they're low.

MR. ZUDANS: Yeah, but they could form a trapped --

MR. MENDOZA: Oh, yeah.

MR. ZUDANS: Or they would need it for other reasons.

MR. MENDOZA: Again I have been informed they're not, the vents are not, operable from outside. So it's not a remotely operable vent.

MR. STAHLE: It looks like TVA will put on a schematic

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here.

MR. MILLS: We might be able to clarify this somewhat with a little sketch on the board.

(Pause)

CHAIRMAN MARK: Does that check off number eight as far as this discussion goes?

MR. STAHLE: Yes, sir. It does.

Moving along, I think we have again --

CHAIRMAN MARK: Dealt with --

MR. STAHLE: -- discussed --

CHAIRMAN MARK: Dealt with nine.

MR. STAHLE: -- nine, yes, sir. Moving along, item 11 -- we dealt with ten -- 11 will be discussed separately this afternoon. A major subject. Twelve we've done. Fifteen.

Fourteen I've been -- it's my understanding that the safety evaluation report is completed on this item. I think I'd like Mr. Parr to briefly state the results of this safety evaluation for it. Since I think I can, I'll categorize number 14 as a completed item from what he has told me.

MR. PARR: Olin Parr, auxiliary systems branch. We have, for all practical purposes, completed our safety evaluation for the auxiliary feed water system reliability evaluations. We are waiting for one input from the reactor systems branch, which I am told will be on my desk when I get back today. So if that's true, why, the SER can be issued probably tomorrow.

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MR. STAHLE: Any questions on the system or the results?

Again, this is complcted, so we'll skip item 15, a dated item. Moving to others, 16 and 17 we have discussed; 18 is a dated item. I'm not prepared to provide any comments other than recognizing, again, as I stated before, the data requirements are identified in the NUREG document; we will be addressing them in one manner or another. It's of course necessary as part of licensing Sequoyah for full power.

Nineteen we have touched on.

Twenty -- this is still under review. There are some problems at this point as to the number of items included in this area. I'll have to defer discussion at this point until I have further information from the staff. I think we'll just move off this item, if I may.

Twenty-one and of course the several there, twenty-two, twenty-five, it's my understanding that the group doing this work, the emergency preparedness plan from TVA is ongoing review, and anticipate, hopefully, that it will be completed this month. The review by FEMA on the state and local level plan seems to be in order and satisfactory, and the drill that was conducted two weeks ago, something on that order, was highly successful, and I think all parties agreed, too, that it was a satisfactory drill and in compliance with the requirement identified in this area.

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Tape 4

Any questions?

MR. ZUDANS: I'd like a question back to the schematic, if I may, Mr. Chairman.

All these valves are remotely operable?

MR. DILWORTH: These are remote manual valves.

MR. ZUDANS: Remote manual. It means remote slack (?) manual? You can do it either way?

MR. DILWORTH: You operate them from the control room manually. They don't open automatically.

MR. ZUDANS: No, I know. But you operate remotely, so you don't go up to the valve and unscrew it.

Question: how are they designed? Are they safety grade or not?

MR. DILWORTH: Yes, they are. Yes.

MR. ZUDANS: They are. All the way to the pressurizer drain tank?

MR. DILWORTH: I think I am correct to say it here, and I'll double-check before this afternoon, but it's essentially a Class 1 system all the way to the tank.

MR. ZUDANS: All the way.

MR. DILWORTH: I'm told it's Class 1 to the first isolation valve, Class 2 to the tank from that point.

CHAIRMAN MARK: That gets us to 26, does it?

MR. STAHLE: I'm not sure if the staff has any input at this point on 26. It's ongoing review. I expect this input next week. I have not -- I am not aware of any problems with this item at this point.

Twenty-seven is a, again a dated item.

Twenty-eight, if I may skip to that one, is incomplete. A confirmatory letter is needed to close out the issue. The staff has prepared its draft SER in this area. At this point, based on today's conversations, I regard that item as also complete.

Last, an item that is really an I&E function, that's an item that will be done, of course, it must be completed prior to the power ascension test program. I have no comments on this, unless Mr. Weston (?) has.

MR. WESTON: The review is in process. They will witness the test when it starts acting.

MR. STAHLE: Okay. The review is in process, if I may repeat, and they will certainly witness the power ascension program.

Well, that briefly covers the situation as I see it. The task remains, of course, to translate this into an SER input.

MR. ZUDANS: You may have mentioned but maybe I missed it, on item 24, what's intended under communications? What's it all about?

MR. STAHLE: I've forgotten that myself. I'll have to look. Communications, is that with regards to the support center?

Emergency planning. Communications and emergency planning.

Would you describe that, Larry?

MR. ZUDANS: No, just what's under it. I don't need -- communications is such a broad subject.

CHAIRMAN MARK: Yes, but it's just with respect to the emergency planning.

MR. ZUDANS: Okay.

CHAIRMAN MARK: I think that leaves it, then, with the exception of 11, which we'll go into separately --

MR. STAHLE: Yes, sir.

CHAIRMAN MARK: -- that this whole list seems to be at the verge of being possible to write down that the things have been attended to --

MR. STAHLE: Yes, sir.

CHAIRMAN MARK: -- or are committed to as of 1/81, or things of that kind, with the possibility that the sampling gear (?) might have to ask for a short delay for installation reasons, for instance. And --

MR. STAHLE: My comment with respect to the data items, this is a matter of review at this moment and the ability of the applicant and ourselves to achieve this is --

CHAIRMAN MARK: Yes. It's not a matter of difference as to what effect needs to be realized or agreement that one will.

MR. STAHLE: That's right. To my knowledge.

CHAIRMAN MARK: It's timing. The only other question I have relates to 19, where I presume it means instruments for inadequate core cooling --

MR. STAHLE: Yes.

CHAIRMAN MARK: -- instead of what's written.

MR. STAHLE: Yes.

CHAIRMAN MARK: Walter?

MR. LIPINSKI: This list of 29, plus the eight on the complete list, are a total of 37.

MR. STAHLE: Yes, sir.

MR. LIPINSKI: And these are all drawn from the action plan as items that are required for the individual reactors?

MR. STAHLE: Again, these are items drawn from the NUREG document 0694. One or two actually were drawn from the previous SECY paper, or Commission paper. And as I stated before, the items that need be completed prior to full power are -- have been identified. Then there's a category of items that are called "dated items," which I have identified in the chart, that have to be completed on the dates designated. And we hope to be able to address these items, and the applicant, in turn, is looking at its ability to comply with all of these dated items. So in my approach as project management I need a resolution, a satisfactory resolution, of these items, from one or two points of view -- a technical review of the items, or items that will be needed from an administrative point of view, identified as conditions to the license or things of this sort.

Mr. Tedesco I think would like to comment. Yes?

MR. TEDESCO: You're bringing up a very important subject, and I wonder, Mr. Chairman, if we can't take a few minutes now to talk about the overall schedule for proceeding for a licensing on Sequoyah for full power. I think there are aspects that we should appreciate, specifically how tight the schedule is. And if it would be all right with you, I'd like to take a few minutes to go over it.

CHAIRMAN MARK: I think that that's perhaps the main purpose of today's discussion. Or very close to it.

MR. TEDESCO: Well, let me see if I can put some light onto it for you. Okay?

And as I indicated, we have a very tight schedule and require full cooperation from TVA and the staff and the committee involvement, that we're pursuing the -- also a similar schedule on North Anna and Salem, and those two plants do not have to come before the ACRS for full-power licensing.

Now, all the plants are bunched up together. They -- North Anna is nearly finished with its low-power testing, which means that it will be ready very soon for full power. We expect on Sequoyah to approve low-power testing the end of this week; Thursday or Friday I think is what Carl mentioned.

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MR. STAHLE: That's correct.

MR. TEDESCO: Now, once we -- we do that, the plant will be ready for full power around the end of the month, give or take a few days but somewhere in that time frame. Now, right now we're projecting ahead that we will have an SER available somewhere around the 28th of July to the 4th of August; and I would say the 28th of July might be a 50-percent confidence factor and the 4th of August might be something like 90 percent.

Now, if you look at those dates, we're shooting for a full committee meeting the first week in August, somewheres around the 7th or 8th. And this then places the question before you about the timeliness in having this report to the full committee. We're realizing the schedule is very short, but we are trying to give you as much information as possible at these committee meetings and at Thursday -- or at Friday's meeting also. So the committee is being brought on early to understand where we're going.

So with the full committee the 7th or 8th of August and then we would have an expectation of a letter in a supplemental SER around the 20th of August, and then we have to appear before the Commission to brief them on full power operations and where we are, and then we would have an expectation of issuing full power license somewhere around the 28th of August.

Now, we're talking now about -- talking about four weeks between the time the plant is ready for full power operation and when we can issue it, assuming this optimistic schedule. So that underscores the importance of our doing our job as soon as possible.

CHAIRMAN MARK: As you see it, where are the pressure points most threatening?

MR. TEDESCO: For Sequoyah it would be moving through the ACRS and getting our report down to you as soon as possible and then having a complete and open dialogue on all the issues to tell you where we are very early in August.

CHAIRMAN MARK: Well, I don't know that we can do anything but take note of that. And I think we have been aware for some time that this schedule is tighter than comfortable. I guess we had hoped until a month ago that it would be possible to be at the end of August by now. But I don't see what we can do about the thing.

It looks as if the things on the screen need not produce roadblocks. Again, we have this item 11 not yet discussed. The other eight items, I think, are not -- that were there earlier are not candidates.

MR. TEDESCO: Well, we are confident of those that we now have.

CHAIRMAN MARK: And if your SER is, indeed, July 31st plus or minus three days, or minus nothing plus three --

MR. TEDESCO: It's minus nothing yet. I think we should look forward to another subcommittee meeting in early

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August, though.

CHAIRMAN MARK: I can't see how one can look forward to it.

(Laughter)

Try to do it if you can. Or what is necessary.

MR. MILLS: Dr. Mark, Larry Mills, may I -- may I make a statement here?

CHAIRMAN MARK: I was going to ask you to comment on this whole session up to this point, if you had things you wished to put in, Mr. Mills.

MR. MILLS: Well, I'd kind of like to comment on the last comments I heard, I guess, more than the rest of it. We're a little bit surprised at -- you know, in discussions of an August ACRS meeting. We were hoping that you would consider a favorable letter in July. I recognize that, you know, perhaps you would like to see some results of our special tests, which will not be completed -- you know, we voluntarily will come back in and provide any data and so forth that you might like to have on that in a, you know, in any one of your sessions. But we were very hopeful that the -- we had progressed to the point in this review that you could consider a favorable letter, you know, this Friday.

CHAIRMAN MARK: I can't speak with any confidence on that. We -- I think we have to complete the other items that are on today's agenda, and that would still leave it unclear as

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to whether the committee would feel at ease with that suggestion or regard it as possible or not.

MR. MILLS: Yes, sir.

CHAIRMAN MARK: To that I cannot speak.

MR. MILLS: Yes, sir, I understand that. However, I did not want you -- hoping you would not preempt the possibility of considering a favorable letter this month.

CHAIRMAN MARK: I think I am not willing to preempt that as of this hour.

Does that get through the things you had?

MR. STAHLE: That covers what I had in mind here for item three of your agenda.

MR. LIPINSKI: Mr. Chairman?

CHAIRMAN MARK: Yes?

MR. LIPINSKI: I have one other question that didn't come up in the course of the presentations. That's fire protection. It's not on your unresolved list, so I assume it's a resolved issue for Sequoyah?

MR. STAHLE: I think I'll have Mr. Tedesco speak to that subject.

MR. TEDESCO: I think the question of what modification in equipment and design aspect has been resolved. We are faced with the question about implementation of the Commission order as to the timeliness that everything has to be implemented, and we are working on that now.

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The Commission order would lead one to believe that everything has to be done by November 1st. And that's being assessed right now: does it really mean that or is there another option available?

But as far as the agreement on what has to be done, we have agreed to that. So in that regard it is resolved.

MR. STAHLE: If I may add, the safety evaluation report on this matter is complete. It's reported all in the SER supplement 1. So from a technical point of view, there's a full and complete understanding of the fire protection aspects for Sequoyah.

It's a matter of the schedule for implementing things that have to be in place by a certain targeted date. But there is no technical input needed at this point in time. Our review is complete.

MR. LIPINSKI: Thank you.

CHAIRMAN MARK: Well, I guess that brings us through item three, then.

MR. STAHLE: Yes, sir.

CHAIRMAN MARK: All right. Our next item which we can take up at this point is four. It has to do with a discussion of the nozzle crack situation that TVA will give. And adjoined to that, if you are inclined, Mr. Mills, comments that you might like to make over the things discussed up to this point.

MR. MILLS: Dr. Mark -- this is Larry Mills -- I don't

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have any comments with regard to the status of the items that -- mentioned by Mr. Stahle. I'd like to reemphasize the fact that, considering the status of this review and our plant schedule, as we hope it will be, I would again emphasize I hope you will consider that in your considerations on Friday regarding a favorable letter on Sequoyah.

Following into that, with the nozzle cracking, we have about a 30-minute preparation here for that, or a prepared talk about 30 minutes in duration. I know that Mr. Shewmon was the one that asked this question. Would you like for us to give the entire 30-minute presentation, or would you like for us to give about a five-minute overview and answer questions? We can do either.

(Pause)

CHAIRMAN MARK: You have provided Dr. Shewmon with some comments on this subject already?

MR. MILLS: Yes, we have.

CHAIRMAN MARK: And have you had any communication with him about the feelings he had from that?

MR. LAMBERT: Yes, sir, we have. Westinghouse provided the material directly and had a conversation with him. Westinghouse reports that he is satisfied with the material as provided.

CHAIRMAN MARK: Is there anyone else in touch with Paul's opinions on this? I am not. But I think the five-minute summary statements would be welcome.

MR. MILLS: All right. We'll ask Mr. Tom Timmons from Westinghouse to just briefly summarize the status, then.

MR. TIMMONS: My name is Tom Timmons. I'm the manager of mechanical systems evaluation in the nuclear safety department at Westinghouse.

This is a brief summary of what is involved in the underclad cracking issue.

Sometime in September of 1979 Westinghouse was made aware of the fact that our French licensee, Framateux (?), had detected some cracking in reactor vessel nozzles in nozzles which they had produced at their Creusot-Loive facility.

The cracking was characterized as being in the base material of the reactor vessel nozzles, in a broad area of the nozzle bore. It was more prevalent in the thicker section. It was confined to the heat affected zone of the second layer of cladding, that is, the heat affected zone produced by the deposition of the second layer.

The longest crack detected was about one inch in length and about a quarter of an inch in depth. And these cracks were just -- were detected by destructive and nondestructive examinations. They were using a 70-degree-angle beam ultrasonic test technique. Subsequent to finding it with that technique, they took a number of boat samples out of nozzles and did destructive examinations to characterize the metallurgical considerations and the exact nature of the cracks. This is how they determined the depth of the cracks and the actual length.

Framateaux and the Westinghouse metallurgists believe that the cracking is hydrogen-induced due to the welding process and the result of the welding process and the heat treatment used on the cladding.

The cases where this type of cracking was detected had a process where there was no preheat put on the nozzle prior to the deposition of the second layer of cladding. It's believed that this is the -- the factor which induced the cracking.

These cracks are very small. This became an issue for Sequoyah following inspection of one of the Sequoyah nozzles and a meeting with the NRC where it was determined that the other nozzles in the Sequoyah vessel should be inspected for this phenomena. And it was subsequently inspected.

Some indications were detected. Metallurgical evaluations were made of the indications that were detected by the ultrasonic techniques. Fraction mechanics analyses were made. And subsequent to that, the NRC staff reviewed these findings and accepted them.

Any questions?

MR. ZUDANS: What size indications were found in Sequoyah?

MR. TIMMONS: At Sequoyah the maximum length of the indication was approximately five-eighths of an inch, the maximum indication.

MR. ZUDANS: And that indication was left in there?

MR. TIMMONS: Yes.

MR. ZUDANS: Based on --

MR. TIMMONS: All of the indications that were found at Sequoyah were evaluated relative to the acceptance criteria of the ASME code and found to be acceptable.

The ultrasonic technique that's used is roughly five to six times more sensitive than the methods that are required by the ASME code. So it detects cracks that are much smaller than those which would be determined to be acceptable.

MR. ZUDANS: If you would apply the same sensitive test to a location away from weld, what size indications would you discover, if any?

MR. TIMMONS: These indications are due to the cladding process.

MR. ZUDANS: So you would find them --

MR. TIMMONS: They're in the base material underneath the clad.

MR. ZUDANS: And that's all over the place.

MR. TIMMONS: Yes. They're not related to welding or the thick sections of the metal. They're related to the clad deposition process for the second layer of clad where there is no preheat applied.

MR. ZUDANS: That means that if you would go to an

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operating reactor that has been in service for a number of years, you might find the same -- will find the same things, is that right?

MR. TIMMONS: There is a possibility. One Westinghouse operating reactor was determined by our evaluations to be susceptible to that phenomenon, because their nozzles were produced by the Creusot-Loive facility and were clad using the same processes which resulted in underclad cracks in the French vessel.

MR. ZUDANS: See, what's sticking in my mind is the following thing. If you know what they are now, after the clad has been deposited, and then have another facility where the system has operated for a number of years, it would be very instructive to find out whether there is a size change that's been -- has made and after being in functioning for several years. Do you have any kind of a --

MR. TIMMONS: We don't have any correlation in that regard. Programs are under way to conduct in-service inspections of the one which we determined to have been clad by the same process, which would be susceptible to it. Methods have been developed, and that inspection will be conducted sometime this year.

MR. ZUDANS: Then you will be able to compare whether or not cracks have grown in the process of time?

MR. TIMMONS: Yes. The fatigue crack growth calculations that were made say that the crack is not going to grow more than about four mils over the 40-year life of the plant, due to the thermal cycles that are specified in the design of the plant.

TVA also has a commitment to do an inspection of the reactor vessel nozzles for this phenomenon at the end of ten years. So information will be available in 1990 as to whether the cracks that were found -- indications that were found in TVA have grown to any significant level.

MR. ZUDANS: If I have not misunderstood you, you will have some such information is six months already.

MR. TIMMONS: Yes. That's from another plant. But we don't have any base-line information for that plant.

MR. ZUDANS: Okay, except that it's been made by the same process and by the same company.

MR. TIMMONS: Right. And we'll know generally, based on observations at a number of nozzles, what size the cracks are.

MR. ZUDANS: Already. Well, that information is very important, I am sure.

MR. TIMMONS: That information will be available to the staff.

CHAIRMAN MARK: You probably said it, but I guess I missed it -- the pieces, specimens you are speaking of at TVA were not made by the same --

MR. TIMMONS: No.

CHAIRMAN MARK: -- manufacturer?

MR. TIMMONS: The TVA nozzles were manufactured by the Rotterdam Dockyard, using a process which was similar but not exactly the same as the one that the French manufa turer used.

CHAIRMAN MARK: In American plants what's the population of nozzles made by different people and different processes?

MR. TIMMONS: Different processes, there's probably 30 or 40. But different manufacturers, you have Chicago Bridge and Iron, Babcock and Wilcox, Combustion Engineering, Purcel DeLauw (?), and Rotterdam Dockyard, are the different manufacturers who have made reactor vessels.

CHAIRMAN MARK: The schematics of the process would be similar in all cases --

MR. TIMMONS: Yes.

CHAIRMAN MARK: -- but the details not?

MR. TIMMONS: The details of the exact cladding process that we use may differ slightly from plant to plant. In general, the procedures that were used by all manufacturers except Creusot-Loive and Rotterdam Dockyard apply preheat before the deposition of the second layer of clad.

CHAIRMAN MARK: Is that good or bad?

MR. TIMMONS: That should do away with the problem in those plants.

MR. ZUDANS: They apparently only cracked where there was no preheat applied, that's what you are telling me?

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MR. TIMMONS: Yes.

MR. ZUDANS: So you found five-eighths of an inch indications in Sequoyah?

MR. TIMMONS: That's correct.

MR. ZUDANS: And these were made by Rotterdam Dockyard?

MR. TIMMONS: That's correct.

MR. ZUDANS: Were made without preheat for the second layer of --

MR. TIMMONS: Yes.

MR. ZUDANS: Okay. Then you have another Westinghouse plant which was made by the same methodology, not by the same people, and that will be tested in six months, right?

MR. TIMMONS: Yes, sometime later this year.

MR. ZUDANS: So these two pictures when brought together might give you more information than the analysis?

MR. TIMMONS: That's correct.

CHAIRMAN MARK: How long has the piece you'll look at in six months been in service?

MR. ZUDANS: I don't know.

MR. TIMMONS: It's the Prairie Island plant. They've been in service for three or four years, I think. At least three or four years.

CHAIRMAN MARK: So it'll give --

MR. TIMMONS: It's more like six.

CHAIRMAN MARK: -- a useful time base.

MR. ZUDANS: How long was the French plant in service before --

MR. TIMMONS: It was a pre-service inspection.

MR. ZUDANS: Oh, pre-service inspection.

MR. STAHLE: Tom, do you know which Prairie Island unit that is?

It's about nine years that the units were in operation -- or, at least, the first unit, Unit One.

CHAIRMAN MARK: This will be a very interesting observation.

MR. TIMMONS: The problem you run into is that the detection method for the operating reactor is going to be slightly different. The ultrasonic test at Sequoyah was done with a hand-held crystal, with the guy kneeling inside that, the nozzle bore. The one for Prairie Island, of necessity, is going to be remotely done with a little bit different method.

CHAIRMAN MARK: But you can look at Sequoyah with that kind of equipment, I suppose?

MR. TIMMONS: Yes.

MR. CATTON: How much radiation or what's the dose that the guy will get when he makes this measurement?

MR. TIMMONS: The measurement that was done at Sequoyah there's no dose because the plant was not --

MR. CATTON: No, at Prairie Island.

MR. TIMMONS: Prairie Island will be a normal in-service inspection. The equipment will be attached to the normal in-service inspection tool and the guy won't be in there doing it.

MR. CATTON: Oh, it's done remotely?

MR. TIMMONS: Remotely.

MR. CATTON: Thank you.

CHAIRMAN MARK: I think it would be useful to have a plan to get strictly comparable photographs of the Prairie Island and the new piece, even though that --

MR. TIMMONS: Attempts will be made to correlate the data.

CHAIRMAN MARK: Yes. That was all I meant.

MR. TIMMONS: Any other questions?

CHAIRMAN MARK: Well, we'll have to confirm with Professor Shewmon if he had further questions. He'd be the most likely --

MR. TIMMONS: If he has further questions --

CHAIRMAN MARK: -- source.

MR. TIMMONS: -- he can contact us directly and we will be happy to --

CHAIRMAN MARK: And you've already been in touch with him, or --

MR. TIMMONS: People I work with have been in touch with him.

CHAIRMAN MARK: Yes.

MR. TIMMONS: We have sent him specific information that he requested.

MR. ZUDANS: You really don't know what that indication consists of, whether it's a crack or intrusion or what.

MR. TIMMONS: It's a crack.

MR. ZUDANS: Crack? In the base metal? In the --

MR. TIMMONS: In the base metal. In the heat affected zone. It's limited to the heat affected zone.

MR. ZUDANS: On the inside diameter?

MR. TIMMONS: That's correct.

MR. ZUDANS: In a nozzle transition or in a cylindrical portion?

MR. TIMMONS: It's in the cylindrical portion. For Sequoyah --

(Pause)

This is a representation of the nozzle and an indication there. This is the one that has the most indications.

This is the reactor vessel side, and this is the -- the -- the safe end, side. So you'll notice there's more in the transition region and fewer in the safe end side. The other nozzles had somewheres between zero and five indications, and this one just happened to have a break. That's why I picked this one to show.

MR. ZUDANS: And those are all of comparable sizes, so they --

MR. TIMMONS: They vary from three-eighths inch to five-eighths of an inch, the maximum being five-eighths of an inch.

MR. ZUDANS: Well, it looks like -- look Texas cracked up in this heat wave.

MR. TIMMONS: That was an outlet nozzle -- or inlet. This is another inlet. This is another one that only has two in it. They occur in random fashion.

Any other questions?

CHAIRMAN MARK: I think not. Thank you very much.

We come to --

MR. ZUDANS: I just want to make a remark with respect to the analysis. If they had that many intrusions or cracks that many and that close, closely spaced, then the conventional type of analysis for a single crack is not necessarily conservative. However, your result is such that it only grows four mils in 40 years, so you would have to be awfully wrong to be in trouble.

CHAIRMAN MARK: Anything else on this?

We come to item five, which has to do with the ice condenser risk assessment studies. I have been asked to relabel those so that the TVA discussion might be the first item of that list.

Now, we're not going to be likely able to finish this

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three half-hour pieces in 30 minutes. Mr. Mills, do you have an idea how long you would like for the discussion of the TVA assessment work?

MR. MILLS: We can make it, sir, any length you want. But --

CHAIRMAN MARK: Well, does it fit neatly into about a 30-minute --

MR. MILLS: No more than.

CHAIRMAN MARK: -- docket?

Let us proceed with that before lunch and then perhaps interrupt this series and come back to it later.

MR. MILLS: All right, sir. We'll call upon Mr. Bob Christie from our engineering design organization to address this, then.

END
APE 4

MR. CHRISTIE: My name is Bob Christie, with the Availability and Reliability Section, Nuclear Engineering Branch, Tennessee Valley Authority.

I would like to discuss with you a few of the reliability studies that have been performed in the last couple of years, and also one that will be performed in the future, that have some application to the Sequoyah Nuclear Plant.

Basically, the studies as shown on the slide have to do with a systems interaction methodology application program which was performed for the NRC by the Sandia Laboratories, a reactor safety study methodology application program, also performed by Sandia Laboratories for the NRC, an auxiliary feedwater system realibility evaluation, which was performed by the Command Sciences Corporation, Colorado Springs, Colorado, for Tennessee Valley Authority, and a plant full-scale safety and availability analysis, which will be performed by the Kaman Sciences Corporation for EPRI and TVA.

Those are the four that we will be discussing today.

The first study was a study performed for the NRC by the Sandia Laboratories which was to help resolve a generic issue, I believe it is A-17, about the interactions that occur between different systems in the plants. The concern was that under the present standard review plan and under the present licensing procedures some things might be slipping through the cracks, and they wanted to have a systematic way to evaluate some of these systems, see how they were interconnected, and whether or not the standard review plan was the proper way to evaluate these systems and interreactions.

The Watts Bar Plant was chosen as an exemplary facility for demonstrating the methodology. The Watts Bar Plant is a sister plant to the Sequoyah plant, and basically most of the conclusions, most of the studies have application to the Sequoyah plant.

It was not a primary objective of the study to examine the Watts Bar Plant or the Sequoyah Plant to determine whether or not it was acceptable because basically there were no exceptions criteria. It was an uncharted area. They wanted to have it a facility. TVA and the NRC and the Sandia Laboratory cooperated in this effort, and the conclusion that was reached during the study and also as part of the final report that the facility is, as is stated, generally well protected against interactions considered within the scope of this study.

MR. LIPINSKI: Is this work documented?

MR. CHRISTIE: Excuse me?

MR. LIPINSKI: Is the work documented?

MR. CHRISTIE: Yes, the work is documented.

MR. LIPINSKI: Do you know the report number?

MR. CHRISTIE: The final draft is a NUREG which I have in my briefcase over there, and I can give you the number if you want me to, or Wang, you can pull it out. It is a blue NUREG in there.

MR. MARK: And you say Watts Bar is a sister but not an identical twin to Sequoyah.

MR. CHRISTIE: That is correct. It is a -- plant. Basically, the direction that TVA had in producing the Watts Bar Plant was that as much as possible it would be used from the Sequoyah Plant. Most drawings at the Watts Bar Plant are Sequoyah drawings, for instance.

MR. MARK: Are there aspects which would be important from the point of view of this study which are importantly different?

MR. CHRISTIE: It is my belief in reviewing the study that the conclusions and the limited scope that the study had would pretty much indicate that the conclusions were also applicable to Sequoyah, and I am not sure, again, how the NRC feels about that, but in our examination of the study, I would say that yes, it is true. The Watts Bar study is probably applicable to the Sequoyah facility.

MR. MARK: I expect NRC will comment on that later.

MR. CHRISTIE: All right.

The report, the final report of the Phase One systems interaction methodology application program is NUREG/CR-1321, or SAND 80-0384.

Basically, the objective of the study was to demonstrate the methodology. It had as a couple of sidelights to look at the standard review plan and determine if the standard review plan was applicable, was covering the things that it should be covering, but basically I believe the NRC wanted to have a systematic way to examine some of the interactions between systems.

The way that the Sandia Laboratories went about this is, they took fault trees. Basically, the fault trees were done for what is known as the top event of unacceptable core damage, and there were three possible ways of getting to the unacceptable core damage, failure of the reactor subcriticality, failure of the decay heat removal systems, or failure of the reactor coolant protection boundary, in addition, which, when you fail the reactor coolant protection boundary, you must fail the mitigating systems.

The fault trees were done, basically three separate, independent fault trees, for a number of power operation modes, full power, startup, hot standby, cold shutdown, and they came out with, I believe, 20 separate fault trees for the different combinations of power loads plus looking at the three types, decay heat removal, reactor

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subcriticality, and the pressure boundary.

After the fault trees were determined or were formed, they went through and looked for the cut sets, and in the cut sets they did not look for minimum cut sets on the first time through. For the fault tree to work, if, for instance, a system would fail, if two components were to fail, that would cause a top event. It might not be a minimum cut set if the system could also fail if just one of those events happened. The minimal cut set would be the single failure.

However, what they were looking for here was connections between the individual components that would combine them such that perhaps you might need ten so-called independent events, but they would be connected in some way, and this method was to look at the ways of connecting them.

After they got done with the connections, they said, if we had had ten events and we combined six of them in some way that they were all connected, let's say, by cooling, we would be down to four events, three independent plus the six -- well, four independent plus the six, four or five independent events. They would stop at that point and not consider that evaluation any further.

However, if they look to ten events and eight of them were connected in some way, such as cooling, such that you had two independent events plus eight events tied together by some commonality, that would become a three-event cut set and it would be examined.

It does not say that there is a potential interaction that actually exists in the plant for those eight components to be combined. However, we will look at it.

This is the procedure that they were using in it. Now, they had some limitations. They did not evaluate any of the reactor coolant pressure boundary mitigating systems, accumulators, core injection, and so on. Condition 1 and 2 of the ANSI 18.2 were the only ones evaluated, normal operation and, I believe, moderate faults.

Functions relating to the consequences were not modeled. Fires, earthquake, hurricanes, tornadoes, flood, sabotage were all excluded.

Again, as you can go back, you can look at the three fault trees as they were basically put together. You had failure of decay heat removal system, failure of reactor subcriticality, and failure of reactor coolant pressure boundary with the mitigating systems. Those were the three that were looked at.

MR. ZUDANS: You probably have it some place hidden in here. How do the human errors factor into this type of analysis?

MR. CHRISTIE: Human errors were a factor which got evaluated after they looked at the hardware components and evaluated only to a limited degree. There was not a massive effort in order to tie together things by human error. It was mostly components and how components are tied together through hardware.

MR. ZUDANS: Supposing human errors would create a situation where you generate a systems interaction?

MR. CHRISTIE: It would not be covered in this study.

MR. ZUDANS: It would not be covered in the study?.

MR. CHRISTIE: It would not be covered in the study. I understand that they have a Phase 2 study which may or may not cover the human interactions part.

MR. ZUDANS: May or may not.

MR. CHRISTIE: Again, as I say, the funding for the program, and that, as I understand it, is still unclear within the NRC as to the continuation, and I am not sure what Phase 2 will cover.

MR. ZUDANS: And which systems were covered under this fault tree?

MR. CHRISTIE: Well, there were quite a few. Basically -- Well, for instance, you can take on the decay heat side, which I brought an example of. They looked at failure of the primary decay heat systems, which is basically the RHR, and that would cover you in modes such as shutdown for refueling and so on.

The secondary decay heat removal systems were basically the auxiliary feedwater system. So, you started at, you know, a top point which is say, decay heat, then you start working on down. You've got secondary or primary side. You looked at some things that might cause reactor coolant loss of circulation or inventory.

You looked at failure of the auxiliary feedwater system, and so on. You looked at failure of the backups to the auxiliary feedwater system. There were quite a few systems covered. I don't have an exact number, but basically they went all the way down. The trees, we've got about -- I am sure -- We can't lay them out in the room in any sense or fashion. I doubt if you could lay them out on this table in any sense or fashion either. They are just an awful lot of trees that they evaluated.

MR. ZUDANS: 400 feet?

MR. CHRISTIE: I would say, at least. Somewhat along those lines.

Now, the things that they looked at for the linking characteristics, again, as I say, human involvement was basically a sidelight to it. Basically, the things that they were looking for is if they could tie together some of the components by what is known as actuation power, which is the AC power, trains A and B, DC power, trains A and B. How you would actuate some of the level control valves, for instance, in some of the auxiliary feedwater systems, whether or not you would have the same sensor.

For instance, if you were using the same sensor to sense pressure, and it was feeding two pumps, and you believed that the two pumps were independent, but they were both working off the same sensor, you would go back and say, that is really not an independent system. You have tied them together by some actuation signal type thing.

Lubrication, whether or not they needed water cooling, whether it was air cooling, et cetera, how the air conditioning was done, how the hydraulics, the compressed air, and also locations. If the components were all in the same room or in the same pipe chase, or so on, they would try them all together.

MR. ZUDANS: Under your hydraulic, you mean those cases where they are physically -- they are connected, right?

MR. CHRISTIE: Well, it would also be more along the lines of control valves.

MR. ZUDANS: I see. But where does this linking characteristic where the two different systems are physically interconnected show up?

MR. CHRISTIE: That would show up in the fault tree directly.

MR. ZUDANS: The fault tree.

MR. CHRISTIE: You wouldn't have to have a connection between the two along the linking characteristics. They would be definitely in the same fault tree.

MR. ZUDANS: All right.

MR. CHRISTIE: Again, as I say, the scope of the study was limited. It used the Watts Bar facility, which is basically similar to the Sequoyah plant, and they looked at these linking characteristics on a hardware level only, and basically found that in general, the divisions, for instance, of trains was good, and very few problems along the hardware line for the limited scope that they we looking at.

The second study which Sandia Laboratories did was what is known as the reactor safety study methodology application program. Sometimes people say it as RSS MAP. The objective of it, after WASH 1400 was published, the NRC wondered whether or not some of the other types of plants would have different dominance sequences.

As noted in the WASH 1400 report, they put the accidents in release categories and some of the probabilities for some of the accidents were higher than others and became what is known as the dominant sequence.

MR. CATTON: Has anybody taken a look at those sequences to try to determine which ones are most

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susceptible to operator error?

MR. CHRISTIE: To operator error?

MR. CATTON: All of the different sequences that are looked at in WASH 1400 require different kinds of intervention or different opportunties for mistakes to be made by operators. And in that operator error is a pretty big number compared to the kinds of numbers you get out of here.

MR. CHRISTIE: I believe the --

MR. CATTON: I think the ones where operators can foul up would be of more interest than where operators are not involved.

MR. CHRISTIE: Okay. I believe the WASH 1400 sequences did include operator errors, but I believe Mr. Taylor will have that.

MR. TAYLOR: Matt Taylor, Probabilistics Analysis staff, in the Office of Research.

The answer to your question is yes. In WASH 1400 the human error possibilities were in effect carried through in the fault trees, right through the sequence level. Subsequent to WASH 1400, we looked at the contribution of the human, the overall contribution to risk, and the nature of the errors that he committed. Basically, what we found, and this was reported to Al Lewis and the Risk Assessment Review Group, and it is the subject of a memo that can be provided, basically what we found was that the human error was roughly half of the risk, and that the nature -- the nature of the error did depend on the design of the plant, that generally the PWR and WASH 1400 was like three-quarters pre-accident error, through maintenance errors or valve errors and what have you.

In the boiling water reactor, it was roughly three-quarters post-accident, or post-accident error, so the nature did bear in with the design.

MR. CHRISTIE: In the reactor safety study methodology application program, four plants were chosen. Sequoyah was chosen as the representative of the Westinghouse ice condenser plant. I believe the other three plants chosen were Calvert Cliffs for the combustion engineering, Oconee for the B&W, and Grand Gulf, I believe, for the D.

The study for Sequoyah in draft form has been provided. We have internally to TVA reviewed it. We still have some questions about the methods and also about the application of some of the success criteria for the various events. These have not yet been resolved, and it is still an ongoing process with TVA.

Again, the results, in just a quick overview look, were that Sandia Laboratories concluded that the ice condenser plants have different dominant sequences from the

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PWR that was evaluated in WASH 1400, which is the surry. However, the risk as related to the release categories, and again, no consequence analysis was done in the methodology application program, as evidenced by the probabilities and the release categories, the risk was similar in a large dry containment plant versus an ice condenser plant.

Now, TVA, as a parallel effort along these same lines, we performed essentially the same type of study that was done by the Sandia Laboratories, because when we started to review the report, we found that it was eaiser to try and do it independently than to come up with detailed evaluation of the way they did things.

Our independent evaluation, which again is a very limited scope affair, has no consequences in it, uses qualitative evluations of the fault trees, does not do any detailed fault evaluation. We basically came up with the same conclusion. We say that the risk on the plant as far as we can tell from just assiging probabilities to the release categories is similar for large dry containment or an ice condenser plant.

One of the items that should be noted, I believe, in these studies, is that the ice condenser plant was assumed to fail both in the Sandia study and in the TVA study, via either hydrogen burn or overpressurization. We did not have in either the Sandia study or the TVA study melt through through the base mat as was permitted in the WASH 1400 report.

We assumed that the containment would fail before it reached the point of melt through. That is, there would either be hydrogen burn or enough non-condensibles to fail the containment, essentially with just about a probability of one. We assign basically small values to the steam explosion, small values to having an isolation valve failure fail, for instance, and a very large value, generally about .99, to the failure by either overpressurization or hydrogen burn.

MR. CATTON: In the Sandia study, you said that operator error was not a factor.

MR. CHRISTIE: No, no. Now, this again is different than the Watts Bar study. The Watts Bar was systems interaction, and they were looking for hardware related interactions. This study took WASH 1400 and tried to compare WASH 1400 to the Sequoyah plant, the PWR surry versus the PWR Sequoyah.

All right. So, where operator action was part of the WASH 1400 sequences, it was also included in the Sandia report and in the TVA report.

MR. CATTON: Okay, thank you.

MR. LIPINSKI: Is there a document covering this work?

MR. CHRISTIE: There is a document which is in draft form for the Sandia work. It has not yet been released, but I do believe that the ACRS does have a copy. One of the provisions which I guess happened before I started in my present job was that the agreement between the NRC and TVA as part of the study was that perhaps it would not be used in the licensing process at any point.

So, we are still in the middle of the review process, and I don't know when it will be reviewed and finished and finalized and issued.

MR. MARK: I believe you mentioned there were some differences in details of conclusions as between the TVA exercise and the Sandia, and that those are under discussion. You have written some results for the Sandia study that there are different dominant accident sequences but risk levels are not very different between them.

MR. CHRISTIE: Their conclusion is that, yes.

MR. MARK: Their conclusion is that?

MR. CHRISTIE: Right, and our independent study would say about the same.

MR. MARK: That is what I meant to ask.

MR. CHRISTIE: Yes, sir.

The next study that we have is the Sequoyah auxiliary feedwater system reliability evaluation. This was a study in approximately November of last year. The

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Tennessee Valley Authority was approached by the Kaman Sciences Corporation and EPRI to participate in a full-scale plant model, which we will discuss in a minute.

As part of a program to understand what we would be getting involved in, we signed a contract with the Kaman Sciences Corporation to perform a study of the Sequoyah auxiliary feedwater system. The system was modeled using what is known as the GO computer code. It is a code that uses a success criteria rather than a failure criteria as used in the fault tree analysis.

It starts with an event, and then traces the possible past for that event, assigning probability of successes to it rather than probability of failures. It is a -- well, it might be characterized as a forward-looking code rather than a backward looking code such as a fault tree. A fault tree takes the top event that you don't want to happen and works its way back as to how it could. The event tree or success tree, the GO method, takes an initating event and looks at the consequences of it. Some will be success, some will be failure.

The analysis was done using the GO code. We basically had approximately four or five criteria that we wanted to look at. Most of the criteria that we wanted to look at was involved in having the auxiliary feedwater work under certain conditions. Some of the conditions that we

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looked at were the event of both main feedwaters and pumps tripped. We also looked at it where we would lose the off-site power, and we would have an event which would be a loss of off-site power, and we would then switch over to the diesel generators.

You can see some of the numbers here. The GO code does not use what is known as a cut set. It uses what is known as a fault set. And we found in the evaluation of the Sequoyah plant, for instance, that there were no single fault sets for the auxiliary feedwater system for some of the events that were used here.

It was basically a study to demonstrate the methodology because we had not had any practical experience with th GO code. The GO code as such had been provided by EPRI to TVA. It is presently a working code within TVA, and we have per`ormed some evaluations of our own with it.

Basically, I have -- the tentative results would indicate that it provides you essentially the same data that a fault tree code would. It has its own problems, and I don't know at this time whether -- I would recommend one or the other. It seems that both give the same re s. Some are better for some problems a..d you just have   a decision as to what occurs in it.

The model that you get involved in, first off, the Sequoyah auxiliary feedwater system is basically a three-pump model. We have one turbine driven aux feedwater pump, and two motor-driven aux feedwater pumps with four steam generators.

The cr eria for success is two out of the four steam generators receive water. We have a condensate storage tank as the primary source of water, and this is backed up by the ARCW supply, and this is basically what the system looks like on a flow type of basis. I don't waat to make things more complicated than they are, but this is what it looks like when you get a GO model type of thing, and it just -- it depends on operators.

They have operators which, for instance, would be an angate. Their operator which they call a Number 10 operator is an angate. Number 2 operator is an orgate. Number 11 operator is an amata angate. And you tie together the various components in a schematic such as this, and you assign probabilities of successes to the various components, and then the end result is the probability of success, which is what they work out.

As I say, the model for the auxiliary feedwater, I believe we ended up with about 1,200 components modeled in the system, and it took approximately four months to complete the study. We have provided a study to the NRC, and as I understand it, it is still under review.

MR. ZUDANS: How do you start this action here?

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MR. CHRISTIE: Well, the way you start it is, they have what is known as a type 5 operator, and that is a signal, and for instance, here, the signal is, yes, there is water in the condensate storage tank, and that is the start, and then you go to the next part, which is a component 1, which is probably just a valve, and that valve says, if you give me water, I will pass that water with a certain probability. Okay? You just work yourself on down.

MR. ZUDANS: Okay, now this is another fixed scheme pre-set, and you start it at the same point. Could you in some way interfere and make some physical change to this model?

MR. CHRISTIE: Oh, yes, yes. One of the benefits that we have perceived out of the GO methodology is that component changes are fairly readily adaptable into the model, and for instance, if you make a component change in a fault tree, sometimes you change your top event quite a bit, and it gets complicated as to whether or not the top event is still the same top event when you add another component.

Adding a component in a GO model appears to us to be a fairly relatively easy thing to do without too much trouble, and it is one of the reasons we like it.

MR. ZUDANS: Supposing in this case where you started at the top by stating, yes, I do have water, do you want it, you could change some probability in the system in

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such a way at some location that it would represent human error, and you have a small probability, a great probability of success, but supposing by human error it becomes a small probability.

MR. CHRISTIE: That is correct.

MR. ZUDANS: Could you do sensitivity studies?

MR. CHRISTIE: Yes, and we have.

MR. ZUDANS: On every -- oh.

MR. CHRISTIE: And we have. They have a program which provides sensitivity studies to you. However, they are having at the present time a little bit of difficulty with -- The Go program has been around for 15 years, and some of the more refined methods of changing and providing sensitivity studies are still being worked out. but they do have sensitivity. We have run sensitivity studies ourselves to see what -- how you would change the components and how it affects what happens type of thing.

MR. ZUDANS: In other words, if you have a certain probability of success at some point, you could vary that and see what the end result is --

MR. CHRISTIE: That is correct.

MR. ZUDANS: -- and you could in fact extract the most important components that way.

MR. CHRISTIE: Yes, and we have performed these studies.

MR. ZUDANS: You have?

MR. CHRISTIE: Yes.

MR. ZUDANS: It sounds very interesting.

MR. CHRISTIE: Yes, it is, and again, as I say, we have had problems by just changing one single point and seeing what effect it has versus their program, which is supposed to integrate all this together and give you the most important component. So, there are perhaps bugs still to be worked out in the program.

MR. ZUDANS: Thank you.

MR. CHRISTIE: The last study that we have is the full-scale safety and availability analysis. This is the one that EPRI and Kaman Sciences has approached us with last year, and which we have now agreed to participate in, and have a kickoff date of July the 1st, 1980. We basically started the modeling to provide two plant models, one to assess plant safety and one to evaluate plant availability.

The initial scope of the project, and I believe still will be quite a bit of the scope of the project, is plant availability. We did not only want to look at plant safety. We want to look at plant availability. So, non-safety systems will be included in the study.

The manpower requirements as we understand them today will be about 80 man months from Kaman and about 30 man months from TVA. It ' broken up into two phases, a

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Phase One, a six-month quick look at the system, I mean, at the plant, and then a Phase 2, which is a rather more detailed affair.

The method that we are using hopefully is to try and work from a top down model. We will use gross simplifications as best we can in the first phase, and then we will refine it and provide more detail as we go on.

MR. ZUDANS: I see here that this GO methodology was funded by EPRI and it is in the public domain, right?

MR. CHRISTIE: That is in the public domain. As I say, EPRI provided the GO code to TVA. We have it in our own house, and we do use it.

MR. ZUDANS: Very well. Thank you.

MR. MARK: Any other questions?

(No response.)

MR. MARK: Thank you. I think that is very good.

I would propose that we leave this discussion and resume it after lunch one hour from now, at which point we will go to the Item C on the agenda as printed..

(Whereupon, at 1:05 p. m., the meeting was recessed, to reconvene at 2:05 p. m.)

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AFTERNOON SESSION

(2:10 p.m.)

MR. MARK: I would like to resume the meeting. Before going on with the agenda I think Mr. Tedesco wanted to add something to his comments this morning on schedule.

MR. TEDESCO: Yes, thank you, Dr. Mark.

I did discuss before lunch what we thought might be a schedule, if we were to go ahead with an ACRS meeting. However, as an option we would like to make the following to you: that we would support and encourage the ACRS to give very serious consideration to writing the letter this month. And the basis for that would be as follows: the staff feels confidently that it resolved the outstanding items that we have to deal with in terms of the TMI action plan. We are estimating about 37 or so. There are about eight attendant -- remain still open.

So I think we can reflect confidence of progress toward the prompt resolution of them. We do not feel as though there are any unusual safety issues before us that would cause us to have great difficulty in resolving with TVA.

We have the assurance by TVA of their support, their willingness to cooperate, their very prompt and expeditious resolution of these items.

So on this basis we would request serious consideration for the committee to provide a letter this month that would address the full power operation of Sequoyah, both Units 1 and 2 with regard to the staff resolution of the outstanding items.

Now what we would plan to do on Friday to help the committee would be to give you a quite detailed status report in our oral presentations, where we are, and then be prepared with adequate staff to answer any questions that may come up.

So we are proposing that as a very serious option, but if more needs -- expended schedule that would carry us into August.

Now a footnote that I would add, that in any case how the committee would go, we would request a letter from the committee that would deal at least with the hydrogen issue.

MR. MARK: Well, I obviously can't commit the committee to action on these points, but both Mr. Mathis and I are perfectly willing to raise this question with the committee, and then I think it will depend upon the feeling the committee can derive from the status report, which I don't think needs to be anything like as detailed as they were this morning.

On the other hand, the list of things on which one is still not absolutely settled should no doubt be available with a covering comment as to how they stand. So with respect to commitments from TVA and expectations on the part of the staff, and we will let this go as far as we can.

Obviously, we have still to discuss some things which can be more sticky than some of the things which I know this morning didn't seem to raise big questions. And I guess both the general question concerning ice condensers and hydrogen in particular are kind of out on the table, not merely with respect to Sequoyah, at this particular time, and they are going to cause a lot of hard concern.

So thanks, Mr. Tedesco. I think we will take a note of that. We are certainly glad to do that.

MR. TEDESCO: Thank you very much.

MR. MARK: If we try to go on, I believe -- and I am not sure what the content of this next item is -- plans for future work is mentioned here, which I believe has to do with the risk assessment consideration of ice condenser, or Sequoyah in particular.

There may be some comments from TVA or some from the staff or some of each.

Does TVA have things mentioned under that heading?

(Pause.)

MR. CHRISTIE: This is Bob Christie, and I would say that basically as far as the TVA position on the future work, we are basically tied up with the Kaman Science Corporation EPRI contract, which started July 1st and hopefully will continue on through December 1981. And that is what our plans are as far as future work and risk assessment on Sequoyah are concerned.

MR. MARK: There might be added to that possibly the ongoing discussion you referred to also with Sandia vis-a-vis the WASH-1400 type review?

MR. CHRISTIE: Yes, there would be. I understand that they are having funding problems but they are trying to resolve that problem and that they had -- we are interested in trying to resolve it, yes.

MR. MARK: There being, I guess, a report which it might -- well, at least where there are differences it would be good to know that they had come to rest somewhere.

If that, I judge that that was probably your thoughts in the connection of followup to this discussion of yours earlier. Are there comments from the NRC on this general subject?

Mr. Stahle?

MR. STAHLE: I think it is time for Matt Taylor to discuss this issue.

MR. MARK: Is this for future work or is it the PAS draft study or is it both?

MR. STAHLE: Matt, do you want to comment on either, future or what?

MR. TAYLOR: With respect to the system interaction study this was a study sponsored by the licensing people, not Research. I don't presently know the status of that. I understand there are some funding difficulties. And so I am not sure that I can comment on what its future status is.

With respect to the methodology applications program, that has been sponsored by the Probabilistic Analysis staff. We are hopeful that the final reports will be complete on all four plants by the end of this fiscal year. But I always was admonished to acknowledge uncertainty of our schedules by qualitative estimates.

I believe that is it in a nutshell. We do have some work underway which is entitled Reliability Evaluation Program, which is looking at other designs. That is a topic that I am not qualified to talk about in great depth, but that is ongoing at the moment.

I believe that is all I have to say.

MR. MARK: Now is that relevant just to future work or does that cover what there is on the item (a) on this agenda?

MR. TAYLOR: No, I have a prepared presentation on item (a).

MR. MARK: Oh. I guess then we have come to that which we would like to have.

MR. TAYLOR: All right, I can make that brief or as lengthy as you want.

MR. MARK: How about something in the middle then?

(Laughter.)

MR. TAYLOR: I wasn't quite sure exactly what topics you wanted to discuss today, so I thought what I would do is

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recap some of the risk assessment work that we have done and also discuss briefly the insights we have with respect to the ice condenser risks, the WASH-1400 design.

By way of background, WASH-1400 was issued in late 1975. This covered two plants, basically the Surry PWR design and the Peach Bottom BWR design.

Subsequent to the issuance of the WASH-1400 we received a number of comments suggesting that perhaps we ought to look in a little more depth at more than just two reactors for an extrapolation on societal risk.

In response to those comments we undertook a methodology applications program, which basically had several thrusts, and we kicked this off in late 1976.

Basically, one thrust was a shorter term effort where we would try to assist the licensing people with generic issues and what have you. A longer term effort which was somewhat independent of any licensing user's request or need was to look at additional design. And this involved basically some analysis of systems as well as analysis of degraded core processes if you will, or core melt processes.

To this extent we undertook some code development work. Largely this was through Battelle Memorial Institute in Columbus, and the systems analysis was conducted largely in the Sandia Laboratories.

We undertook some code development work which was culminated in the MARCH/CORRAL code, the MARCH code being meltdown accident response characteristics which basically looks at various sequences, the timing involved, the containment failure mode characteristics, over-pressure if you will, various timings involved with respect to meltdown of the core, penetration of the reactor vessel, et cetera.

This code development was undertaken in recognition of the fact that the treatments in WASH-1400 for small LOCA's and transients which tended to dominate risk was quite limited. And so I think we are in a position now to say we do have upgraded capability to look at the meltdown accidents initiated from a small LOCA or a transient event.

I might mention that this MARCH code has been used in connection with some of the Rogovin investigations to look at the processes that went on in TMI.

Subsequent to WASH-1400 there was a risk assessment review group chartered that critiqued and provided peer review to the WASH-1400 techniques. They had a number of recommendations, some technical, some with regard to why the use of event tree, fault tree techniques.

Some of the recommendations were to eliminate the smoothing techniques which were used to carry over a certain fraction of the probability of one release category to another in WASH-1400.

They made some very explicit reference to uncertainties being somewhat larger than was stated in WASH-1400 and I certainly would, having looked at various systems and various designs, certainly subscribe to that statement.

So some of the work that we have undertaken has endeavored to recognize the recommendations of the Lewis report.

We have also within the past year had benefit of the risk assessment work that was conducted by the Federal Republic of Germany on their Biblis reactor. I will touch on, just give a perspective on what that was. Basically, they found that the transients and the small LOCA's also control the risks, as did WASH-1400.

MR. MARK: You didn't mention the CORRAL code.

MR. TAYLOR: Oh, the CORRAL code, I am sorry. That is really a containment radioactive release assessment, if you will. It is looking at the magnitudes of release of radioactivity that would accompany a particular sequence -- accompany release from the containment to the environment.

MR. MARK: So in the event of containment rupture it would discuss how much of what might get out.

MR. TAYLOR: Right. In other words, you might have so many curies or such and such fraction of the iodine --

MR. MARK: Does the attempt to assess the washing out of iodine or the deposition of aerosols and things of that kind?

MR. TAYLOR: Yes, sir. Largely, it is within containment processes. It would take account of such things as an ice condenser, the retention characteristics of an ice condenser. It would take account of retention characteristics by spray, retention characteristics by plate out.

MR. MARK: And compare these with Surry?

MR. TAYLOR: Compare these largely from one sequence to another, and then we have used this work to go back and rebaseline the results in WASH-1400 and I will show you a little bit of what the outcome was.

MR. MARK: Very good, and then beyond the fencepost on, goes back to the CRACK code as it was and uses that?

MR. TAYLOR: Right. One can use the release categories as some sort of an indicator. In other words, if you look at the release magnitudes you can then use the release categories in WASH-1400 as sort of an indicator of how the risk stacks up for various accident sequences.

MR. MARK: I like that approach, because I don't believe too much concerning CRACK code.

MR. TAYLOR: I will show you both here, a little bit, the results of both.

MR. MARK: You could guess the rest from a demographic comparison.

MR. TAYLOR: Well, I will show you the results of running CRACK on some individual sequences.

MR. MARK: Very good.

MR. TAYLOR: Again I did mention that subsequent to

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the -- well, TMI, there was some results -- or investigations, the Presidential Kameny investigations, the Rogovin investigation. I believe in both reports they supported wider use of risk assessment techniques.

The integrated reliability evaluation program was initiated back the latter part of this past year. It was undertaken with the Crystal River plant, an existing B&W plant, and I will mention some of the basic results here.

Recently, the probabilistic analysis staff has been involved in the question of the Indian Point interim operation. And a little scoping exercise was done there to explore how the dominant accident scenarios might (inaudible) risk for the Indian Point plant. And as part of that effort we did undertake to use the results from the reactor safety study methodology applications program, RSSMAP is the shorthand for it, to rebaseline to WASH-1400, PWR and BWR design.

And as part of this work we also took a look at how these designs might vary at a populated site; namely, the Indian Point 2 site. And I will show you or give you a quick overview on the results of that.

One of the plants we did use was the Sequoyah plant as an indicator of the risk of one design.

Now the results of all these studies that I have mentioned IREP, RSSMAP, WASH-1400, Federal Republic of Germany, the perspectives that they have provided are set forth on this

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slide.

One can see that there is, with respect to the estimated probability of severe core damage; namely, core melt, the estimates vary by roughly an order of magnitude. One can ascribe some significance to this. I would again acknowledge that there is substantial uncertainty around any of these estimates. But these are roughly best estimates or central estimates, if you will.

What I will be talking about subsequent to this largely will be the Surry, Peach Bottom and Sequoyah designs.

Now all the studies that we are aware of to date has suggested that one can go into a design as certainly a start point for focusing on the risk of a design, with a number of accident sequences. These are most likely the ones that you will find that dominate the spectrum of either core melt and/or the risks from core melt. There will be high magnitudes of release accompanying some of these. There will be lower magnitudes of release accompanying others.

MR. CATTON: Excuse me, how did your probability of severe core damage per reactor year for Indian Point compare with the mini-WASH-1400 study done by the Zion/Indian Point utilities?

MR. TAYLOR: I think we are higher, somewhat higher.

MR. CATTON: By a factor of 20?

MR. TAYLOR: No, like factors of 2ish. Oh, I should, let me go back to that slide. Let me take just a few minutes and point out --

MR. CATTON: I thought they were a factor of 100 below Surry.

MR. TAYLOR: No, not with respect to core melt. That may be with respect to a certain release category.

MR. CATTON: You are right, thank you.

MR. TAYLOR: I think we are comparable in the core melt estimates, within the uncertainties of these estimates.

MR. CATTON: That is right.

MR. TAYLOR: This reflects an estimate that existed after some of the interim steps were taken, which back in interim requirements were taken or put forth on Indian Point back in about December. These did look at some of the risk dominant scenarios and recommended some improvements in the interim.

MR. CATTON: But that number does compare favorably with theirs?

MR. TAYLOR: I can't recall their specific number, but I believe it is fair to say yes.

MR. MARK: Did you have a comment on just that point?

SPEAKER: (Inaudible.)

MR. MARK: Thank you.

MR. TAYLOR: Do you recall the number that --

SPEAKER: (Inaudible.)

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MR. TAYLOR: Yes, several times ten to the minus five. But I don't remember.

MR. CATTON: It doesn't bear on Sequoyah.

MR. TAYLOR: No.

MR. CATTON: It was just for my own information.

MR. MARK: Does the number that is shown here for Surry relate to Surry as she was in 1975 or after some of the action plans?

MR. TAYLOR: As it was in WASH-1400, so that would be 1973-4 vintage.

And the number I have here for Sequoyah relates to, as we understood it, several years ago when we undertook the --

MR. MARK: So it also is pre-TMI?

MR. TAYLOR: It is also pre-TMI.

Now as I was saying, all of these studies have served to point out a number of sequences that can be used as a point of departure if you will. These will likely control the core melt and probability and the risk spectrum.

For example, these are primarily geared to the PWR. In WASH-1400 --

MR. CATTON: How do I recognize hydrogen burn?

MR. TAYLOR: You don't on this one. The containment failure mode is not included here, but I will touch on that in a moment.

MR. CATTON: Okay.

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MR. TAYLOR: The Surry was dominated largely by those two sequences, the risk itself. The core melt was dominated by these sequences. In the IREP plant, for example, Crystal River plant, that particular sequence was a very important sequence. And they are like one-third of the core melt probability.

And the small LOCA's also came into play. In the ice condenser we found these sequences tended to be quite important to the risks, and I will give you some perspective on how important in a moment.

I think it is important just to recall when I talk some of this alphabet soup here that what I will be talking about in the case of the ice condensers is simultaneous failure of all engineered safety features, or mainly the containment spray and heat removal systems and the core cooling recirculation systems in the recirculation phase, following a loss of coolant accident.

Now in the reactor safety study methodology applications program, and I will call that RSSMAP for short hereafter. The backup of the sequences were compared with the WASH-1400 PWR design in this histogram form.

In other words, this is simply a histogram reflecting the release categories as set forth in WASH-1400. One could look at this and say, well, it appears to be some comparability of risk. Well, largely the risk was dominated in WASH-1400 by these three categories. So within the uncertainties of analysis one would say those were quite comparable.

MR. MARK: Which is ice condenser and which is RSS?

MR. TAYLOR: Oh, I am sorry. The ice condenser would be this one here, I believe. Yes.

Sorry, that is not where we marked on this.

(Pause.)

Well, this does not give you very much fine structure to take a look at just what sequences have the greatest impact, and we have tried to do that here, say in connection with our recent work.

The dominant, what dominated the risk of the PWR in WASH-1400 was this release category number 2.

MR. MARK: I am a little worried about your identifying the ice condenser. The frequency for the ice condenser is a little lower than that for Surry on the table.

MR. TAYLOR: In what respect? Which categories are you talking about?

MR. MARK: Well, probability of severe core damage. But these are the high release categories.

MR. TAYLOR: These are the high release categories; you do not see the lesser release categories which really contribute most to the probability of core melt.

MR. MARK: Okay.

MR. TAYLOR: Okay? They are more probable, but they

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are more biased toward the lesser releases if you will.

MR. MARK: Thank you.

MR. TAYLOR: I am not sure you have this slide. I just want to use it for a second to set the stage for the next slide.

When one talks about risks, there are a number of risk measures one might adopt. I believe the recent concern with respect to Indian Point was acute fatalities and possibility of latent cancer fatality.

So in other words, this is the concern that I think most people think about in terms of risk. So I will talk about the chance of public risk from the ice condenser versus the WASH-1400 design in terms of these two parameters.

One can also talk individual risk, which is rather independent of the population beyond the reactor. I will talk a little bit about that.

One can also look at the risk impact of particular sequences, and I will talk a little bit about that also. I would like to jump into these two first - individual risk versus distance.

(Pause.)

Well, on this viewgraph what we have is the reactor safety study, BWR/PWR designs, and the impact of the rebaselining effort, which as I said, involved rebaselining some of the principal sequences with the updated code, the MARCH and the

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CORRAL code development work. And it also involved eliminating smoothing and the calculation of the explicit sequences.

The net effect was not large. Now we also did the same thing with the ice condenser using the MARCH/CORRAL results and the results of the systems analysis done by Sandia and ran the CRACK model to get a perspective of how this particular, the ice condenser compared with the WASH-1400 design.

Within the uncertainties of these analyses one would say the risk of a few fatalities, the risk to an individual would be comparable, as a function of distance beyond the plant. Again this is site independent. This was only designed dependently.

I would point out also this assumes that a person, the individual located offsite can be quickly randomly located anywhere around the plant, and he would not move until one day after the cloud had passed. He would see one day's worth of exposure.

So in essence, no prompt evacuation.

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This next viewgraph illustrates the same information but with respect to an individual's risk for latent cancer fatality. Again you can see the ice condenser and the PWR/BWR designs in WASH-1400 are comparable, all similar.

We also took a look at, you recall this histogram I just showed, we took a look at some of the key sequences in that histogram and the MARCH/CORRAL predictions with respect to the release magnitudes.

MR. CATTON: How did the MARCH code handle the steam spike that occurs at vessel failure and couple it to the ice?

MR. TAYLOR: We did take a look at what they call debris fragmentation, which was a quenching spike, if you will, after you have penetrated the vessel.

Now you would not have any spike as you leave the vessel unless you had the vessel at some pressure. Not all scenarios do have that. The dominant ones here would, however. And that would be about, as I recall, don't hold me precisely to this number, but roughly 25 psi spikes, as I recall, as you penetrated the vessel.

MR. CATTON: So there must be some model --

MR. TAYLOR: There is a model.

MR. CATTON: -- that takes care of the condensation on the ice?

MR. TAYLOR: Yes. In fact, it depends on when particular sequence, the timing --

MR. CATTON: You have got a factor of 4 less on the pressure spike. I think it is about 120 without an ice condenser and a lot bigger volume. See, you got what, half the volume in ice?

How reliable? What is the uncertainty on that number?

MR. TAYLOR: Oh, there is uncertainty in all these analyses. Let me --

MR. CATTON: Well, I think I understand that.

MR. TAYLOR: I don't --

MR. CATTON: There is a lot of uncertainty in using MARCH alone.

MR. TAYLOR: That is correct.

MR. CATTON: And I think what you are showing here is that the risk associated with ice condenser is less.

MR. TAYLOR: No, I am saying it is similar, within the uncertainties of analysis it is similar. I can't make a conclusion that it is less or it is greater.

MR. CATTON: Backfill is not probability, and I see a factor of 10.

MR. TAYLOR: A factor of 10? Where?

MR. CATTON: Well, at the distance of 10 miles. I see a factor of 10, I believe.

MR. TAYLOR: Okay, well, one should not -- let me get back to that then. One should not conclude that the risk is dominated in that 10-mile spectrum, but most of your people tend to be out here. And then if you were going to measure the chances of acute -- or use as a measure the chance of latent cancer fatalities, this would be a small contribution to the total.

MR. CATTON: Okay, but the curve is clearly not above?

MR. TAYLOR: No.

MR. CATTON: So the MARCH calculation --

MR. TAYLOR: It certainly could be if you include these -- you know, look at the -- what I am saying is that we see no apparent great differences.

MR. CATTON: The conclusion I would draw from this then is that the ice condenser is about the same as any other part?

MR. TAYLOR: The overall risk is not significantly greater.

MR. CATTON: Okay. Now that is based on the use of the MARCH code, which by its originator is highly uncertain?

MR. TAYLOR: Yes.

MR. CATTON: Okay? That is the only point I wanted to make. I don't know how much faith to put in all these numbers.

MR. TAYLOR: Well, I think --

MR. CATTON: Very little?

MR. TAYLOR: Well, all I am saying is that whatever the uncertainty is, all right, and I have heard Dr. Vesely describe it on the core melt as perhaps order of magnitude, and when you get up to the large end of the consequence you can put your whole hand, there are several orders of magnitude. All right? But this is the overall uncertainty.

The point is that this does represent the best we have in terms of a consistent comparison.

MR. CATTON: But is it good enough to make decisions with respect to ice condenser containment and things like inerting or not inerting?

MR. TAYLOR: Okay, well I would go back to my first slide with these. These efforts were not undertaken for licensing purposes. They were undertaken for research purposes. I am not sure decisions do hang on this. All right?

MR. CATTON: Okay.

MR. TAYLOR: This is information that you may wish to consider with respect to licensing.

MR. CATTON: Thank you.

MR. MARK: Before you take that off I am wondering if there is a confusing point in the labeling. This is listed a probability versus distance given a core melt.

MR. TAYLOR: Right.

MR. MARK: But that really isn't what it means, is it? Isn't that the probability per year --

MR. TAYLOR: Right. In other words, if I were to multiply this in effect by ten to the minus four I would have the absolute.

MR. MARK: Okay.

MR. TAYLOR: I should go back --

MR. MARK: Then it is properly labeled?

MR. TAYLOR: Properly labeled, right.

MR. MARK: Then the note that the core melt

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probability is assumed to be this is totally irrelevant?

MR. TAYLOR: That is right. You are correct. Let me tell you why that is there. That represents the average of the population of the designs that we have seen so far. All right.

But if you look at the core melt probability that I showed on the earlier viewgraph, that roughly is the average.

MR. MARK: So it is an informative note, but it is --

MR. TAYLOR: That is all it is.

MR. MARK: -- it looks as if it were to be applied to the graph in some way, which it isn't. This is the probability per core melt.

MR. TAYLOR: Right, per core melt.

Now, Dr. Catton, I am not sure I answered your question fully on --

MR. CATTON: You answered my question. I am just more unsettled than ever about the use of these methods to come to any conclusions. These numbers are really awful small, and you are talking, and they are based on MARCH code, and it is not a random process.

I think when you got molten core on the floor it is a deterministic thing. It is not a random thing and probabilistic methods on the slide.

MR. TAYLOR: I agree, and now you will find that in WASH-1400, really the only uncertainties that were carried

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forward were those associated with component failure rates and the systems analysis.

It was not carried forth in the consequence modeling. It reflected best estimates. And then there were judgments made with respect to the uncertainties with regard to the CRACK modeling, the uncertainties with respect to the systems modeling, uncertainties with respect to the MARCH and CORRAL release modeling, if you will.

MR. CATTON: So I would put more, an uncertainty maybe 10 to the 3 on this rather than 10 or 100.

MR. TAYLOR: Do you have -- I don't -- well, I will be quiet. I was going to say I would love to see your analysis.

(Laughter.)

MR. CATTON: I didn't make any analysis, but nobody else did either.

MR. TAYLOR: No, there has been some analysis made on the -- this is an entirely different program, and perhaps one should talk about that at some later time.

Getting back to your question about the vessel penetration, this viewgraph is to illustrate some of the MARCH predictions with respect -- it is not in there, this is -- let me see if I have --

MR. CATTON: Is there any way I can tell from the offset soup whether the hydrogen burn policy --

MR. TAYLOR: Well, that is the point I want to make

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here.

MR. CATTON: Okay.

MR. TAYLOR: You don't have this, but the, if one looked at failure in the steam over-pressure mode, and let me say this is the risk dominant scenario that was identified. So if you look at the steam over-pressure you find that the MARCH predictions had that the core melt would start at about 152 minutes, end at about 180 minutes; the reactor vessel would fail at about 193 minutes; the ice would melt, would be complete in about 273 minutes; the containment failure, if now no other process occurred such as hydrogen burn or steam explosion, this is simply failure of the containment heat removal and the quenching systems and the core cooling simultaneously, that the containment would fail at roughly 300 minutes through steam over-pressure.

Now on the other hand, this again, looking at this next line, this is the failure if we have some fragmentation as we fall into the reactor cavity, combined again with the pulse of the reactor vessel itself, the pulse from the melt through the reactor vessel.

Again, start of core melt at 152 minutes, end at about 180 minutes; the reactor vessel would fail at about 193; ice melt would be complete at 195; containment fails roughly at 197. For all practical purposes the containment would fail within the time domain when the reactor vessel melted through.

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All right? So there is a significant difference here that can be attributed to this pulse during the meltthrough.

Now I don't think I have a great deal of credibility in saying there is a big difference between 193 minutes and 197 minutes. But I think the insight is that it would fail in that time domain.

MR. CATTON: Does it make any difference as far as the ultimate consequences, that early, 301 minutes versus 190?

MR. TAYLOR: No, I think the big element here is the relationship to the ice melt, which can be a retention factor for retaining the fission products. That is the big variable in this case.

Then I can go through the similar things, assuming a steam explosion, you would fail your vessel prior to, through the steam explosion mechanism prior to melting through the bottom of the reactor vessel. And you could look at the hydrogen burning. One can't say just when the hydrogen will burn as it accumulates in the ice condenser, but certainly as the core undergoes degradation you have a large window here where hydrogen could burn.

MR. CATTON: If the hydrogen burn occurs early, before the reactor vessel fails, you may not have the scavenging effect of the ice to rely on.

MR. TAYLOR: That is correct.

MR. CATTON: Does that change things dramatically?

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MR. TAYLOR: Yes, it does. Well, I won't say dramatically. I will give you a perspective on that when we get into the individual sequences. All right?

MR. CATTON: Okay, sure.

MR. TAYLOR: But it does have an effect in that you do fail at an earlier time and the retention processes are not operative, simply because you are failing in this earlier time. I say not operative; I am saying it has less time before those retention processes attack on the release. That is the biggest difference we see.

Now as you recall from the histogram, these are individual sequences that we have run. I am sorry, this is a very poor viewgraph here, but this is the risk of -- individual risk of acute fatality versus distance for various core melt scenarios that were found to proponently contribute to that histograph and/or the risk.

You will find that at the close-in distances what we see here is that the combined failure, the S_2HF sequence is dominant. All right?

The CRACK model does not predict fatalities beyond about one and a half miles in this particular scenario. On the other hand, you will find that the steam explosion can't have an impact at some greater distance, so that if you would take care of just one sequence, for example, you still would have some chance of affecting people at some greater distance with a

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different sequence.

The point is here there is a hierarchy of sequences. Now if we were to inert this containment for example, let's say we eliminate the hydrogen burn. All of these cases here entitled Gamma are the hydrogen burn cases. All right?

If we were to eliminate those three sequences, the rough magnitude of the surface might look something like this. That is, again within the uncertainties of the analysis, that is what one would see.

MR. CATTON: Factor of 10 maybe?

MR. TAYLOR: Maybe. But that is only on this particular measure of risk.

MR. ZUDANS: The reduction of distance is only, I guess related to the fact that you have a larger radius.

MR. TAYLOR: Well, there are some imperfections on the scenario. There is mixtures of isotopes that are predicted to accompany each one of these scenarios. This particular case here is failure of the emergency core cooling system in the recirc phase following an intermediate size loss of coolant accident. Then a steam explosion occurs.

This would be a rather heated energetic release, and this would tend to put a release out at somewhat greater distance of the plant.

Again this slide is the individual, or viewgraph is the individual risk of latent cancer fatality for various

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scenarios, quite similar to the other one. You can see again that the S_2HF scenario, the combined failure of the recirc phase of the core cooling and containment systems, followed by hydrogen burn.

Again, the gamma scenarios are dominant in terms of the expected risk of latent cancer fatality to an individual. If one were to again eliminate these particular cases, the gamma cases, then you could see here is again this Event V scenario. So depending on where you want to take your point of reference beyond the plant you could have maybe an order of magnitude considerably less, due to preventing the hydrogen burn.

Now I mentioned I would show some of the design comparisons in terms of societal risk. We did this work recently for reporting to the commissioners in connection with the Indian Point design. The Indian Point site is acknowledged to be one of, probably the most populated site.

What we did was assume, we placed, magically placed these designs at the Indian Point site, and took a look at what the impacts might be. And I don't want to dwell on this, but this is the CRACK model results for the Indian Point site. Again one can see here is the WASH-1400 BWR design, the WASH-1400 PWR design, the ice condenser design, and the estimated Indian Point design.

And item number 5 is the effect of some interim actions that were taken, as best we could gauge their effects.

And again at the Indian Point 2 site, this is the latent cancer risk. You can see here that the ice condenser design and the Peach Bottom design or the WASH-1400 BWR design, quite comparable, in terms of the predicted results from the CRACK model.

This intermediate case is the PWR design, and the two lower cases are the Indian Point design.

So on this basis I would conclude whether you wish to accept that for decisionmaking and licensing or you wish to accept that as simply a viewpoint of someone exercising these techniques. I would conclude that the Sequoyah design is quite similar to the WASH-1400 PWR and BWR design as far as the risk, public risk from reactor accidents.

That is about all I have to say.

MR. CATTON: In the WASH-1400 study was the BWR containment inerted?

MR. TAYLOR: Yes.

MR. CATTON: Thank you.

MR. MARK: So the gamma sequence didn't apply?

MR. TAYLOR: Well, there was a little different nomenclature, but --

MR. MARK: Yes.

MR. TAYLOR: -- perhaps I should say something. Inerting had no effect because the dominant scenarios broke the containment before the core melted.

Again, these are all based on MARCH and CORRAL.

MR. MARK: Can you easily say what accounts for the appreciable difference between, say in that last graph you had of the latent risk?

MR. TAYLOR: The last graph?

MR. MARK: You have Indian Point appreciably, but whether significantly or not is another matter, below Surry.

MR. TAYLOR: Right --

MR. MARK: Why is that?

MR. TAYLOR: My perspective is that they have taken a -- the perspective I have now is that those were high release categories that appeared in WASH-1400 as the other designs, have been somewhat reduced in probability, considerably reduced in probability.

MR. MARK: Well, you set the numbers down for steam explosion?

MR. TAYLOR: No, that is not the case. For example, that event (inaudible) in essence -- -- loss of coolant action followed by the core melt. The containment has no real (inaudible) This scenario is located inside of the containment -- -- that design is accommodated somewhat uniquely with respect to this scenario.

MR. MARK: So Indian Point has better mitigating features than Surry did?

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MR. TAYLOR: On the basis of what we can say or see on that scoping analysis, using development accidents (inaudible) see the problem in a number of other designs.

MR. MARK: Very good.

MR. TAYLOR: Yes.

MR. MARK: Now what accounts for the further setdown from 4 to 5, which is Indian Point before fix and after fix?

MR. TAYLOR: This is again the results of the interim actions that were taken -- -- we thus engaged them, and they had a certain impact on particular accident scenarios, all the types that I showed.

(Inaudible)

MR. MARK: Those fixes could just as easily then be applied to Sequoyah and Surry?

MR. TAYLOR: Well, I won't say just as easily. Perhaps. I can't say just as easily. I would guess, yes.

MR. MARK: But aren't they the TMI action items?

MR. TAYLOR: No. No, these are special. Well, they included TMI action items, but there were special interim actions on the Indian Point design.

I believe these actions were also taken up at Zion. Now perhaps -- -- would like to add -- am I correct?

(Pause.)

MR. GARY ZIMMERMAN (?): Dr. Mark, you might remember last winter, early spring we were down here to talk

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about design at Indian Point, special reviews in view of the high population density. Like Matt Taylor indicated, they were interim measures that we elected to pursue with those two licensees and those design, Indian Point facility, and there were differences of opinion as to how much it did contribute to a reduction in risk.

I think for the most part what Matt has up there was a general consensus -- -- contributing.

Some of them could be applied to Sequoyah and other plants, and some were quite unique.

MR. TAYLOR: There is one, for example, that might be difficult to apply. Sequoyah has automatic, as I understand it, automatic initiation in the recirculation phase for their core cooling system. Indian Point does not. So they are dependent totally on the -- I won't say totally but heavily on the human and the human procedures.

So there was some extraordinary effort gone to to strengthen the procedures, the training, and what have you.

Now that in itself amounts to roughly a factor of 2, if you will. The human was roughly one-half. Hardware was roughly the other half in that particular scenario.

So Sequoyah would not benefit from that extra training necessarily since the human does function as a backup. It would largely be hardware dominated.

So one would have to examine from a sequence to

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sequence basis.

MR. MARK: Well, one last question. I am sure it is not in the study or on this graph. We have seen the statement a number of times that, and the arguments concerning hydrogen control and small containments. But since TMI the probability of accidents releasing much hydrogen has been greatly reduced.

And so I am wondering what that would do to one of these graphs. That was in the SECY 107, for instance.

MR. TAYLOR: I think the probability of getting to that conclusion (inaudible) condition has been reduced, but once you get to that condition the probability of hydrogen being involved is still about the same.

MR. MARK: I would understand that.

MR. TAYLOR: And then the probability of --

MR. MARK: It must have been a prevention step.

MR. TAYLOR: Right, it is a prevention step, which could be taken on a number of sequences --

MR. MARK: Which presumably applies fairly generally to quite a number of plants since the TMI action plan is generally applied.

MR. TAYLOR: That is true. Certainly you would improve the operator's perception of the factors in input degraded core circulation.

MR. MARK: But that statement is not based however on the study of the sort that you have described? It is just an

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article of faith?

MR. TAYLOR: Would you repeat that?

MR. MARK: Well, the statement that we have reduced the hydrogen generation probability by quite a bit, which allows us to continue to operate certain plants, by the TMI action, is not based on the study that you are describing to us or a similar study?

MR. TAYLOR: No.

MR. MARK: But is, I said, an unquantified article of faith.

MR. TAYLOR: Well, I think it is generally agreed that they have improved the situation with respect to the TMI scenario, the transient scenario. But it is an unquantified article of faith.

(Pause.)

end T. 6&7

Burrell

MR. DILWORTH: I am George Dilworth, Chief, Nuclear Engineering and Engineering Design for TVA, and today I would like to discuss with you the efforts that TVA has made in evaluation of hydrogen above the design basis at Sequoyyah, and other TVA plants.

At TMI-II, the core was uncovered to the extent that severe core damage and resulting hydrogen generation from zircwater reaction occurred. This ultimately led to hydrogen release to the containment atmosphere, and the subsequent assumed hydrogen burn which produced a 28 pounds per square inch pressure spike. The hydrogen release, a portion of which burned, resulted from the zirconium water reaction in the estimated range of approximately 25 to 50 percent.

We recognized in our nuclear program review that we had in-house at TVA in 1979 the need to thoroughly investigate hydrogen generation as a result of core damage that could occur on a TVA plant.

Beginning without Sequoyah and Watts Bar plant, we started to study. Our initial efforts in the study of hydrogen were focused on the TMI vent, and the capability of Sequoyah containment to withstand hydrogen combustion.

Since these initial efforts that we made, we have made a limited study of risk similar to WASH-1400, and Mr. Christie spoke somewhat of this this morning. We identified represen ative transits which could lead to some core degradation, and investigated the more important concepts for the prevention or mitigation of consequences from hydrogen combustion.

Concurrent with these efforts, TVA has been aggresssively pursuing the implementation of the new NUREG-0578 requirements, and other NUREG requirements, and those that TVA also has self-imposed on itself to substantially reduce the chance of a situation similar to TMI where core damage could occur, and hydrogen result.

We believe that because of its low risk level, Sequoyah is safe to operate at full power based on the present capability of the ice condenser containment, its recombiner and hydrogen purge system, and the substantial improvements in equipment and training which are being implemented.

We believe that the additional reduction of overall risk may be achieved by protecting the containment from consequences involving metal water reactions and hydrogen releases beyond the design basis of Sequoyah nuclear plant.

We have initiated what we believe to be a reasonable and positive approach to the problem by committing substantial resources in an effort to install, afther a thorough safety review by TVA and NRC staff over the next couple of months, an interim distributed ignition system, and improve this interim system to a permanent system as development work proceeds.

MR. CATTON: Are you going to describe this?

MR. DILWORTH: Yes, whenever we get to that.

The activities that are going on TVA have been going on for the past nine months.

As I stated earlier, we have studied the hydrogen question for about nine months. We have determined that Sequoyah can withstand substantial amounts of hydrogen above the design basis. We have made significant modifications through NUREG-0578 requirements and other TVA imposed requirements to reduce the potential degrading events.

We have done a limited risk assessment showing that Sequoyah is comparable to the WASH-1400 PWR reference reference plant.

We have studied a number of proposed concepts for resolution of hydrogen issues. We have an interim distributed ignition system that we chosen for implementation at Sequoyah. We have development work on controlled ignition that is proceeding now, and we are also studying halon suppression as an alternative approach.

Just quickly here to review the capability of the Sequoyah containment. It has a yield pressure of 33 psig, an ultimate pressure of 42.5 psig, a volume of 1., million cubic feet. It has a containment capability to withstand, we think, 25 percent metal water reaction based on this ultimate strength.

I would like to say, in regard to the analyses that we have run to determine this capability of metal water reaction, we feel that the analyses have been very conservative, simplistic to a great degree, and should not be the final basis to determine the capability of Sequoyah containment.

As you will see later, we have attempted to develop more proper physical bases for modeling and determining what the pressure, and therefore the final capability of metal water reaction would be.

MR. MARK: That pressure of pressure of 33 psig that is design pressure?

MR. DILWORTH: Design pressure is 12 psig.

MR. MARK: So that is close to three times.

MR. DILWORTH: This is yield pressure of 33 psig, yield strength.

MR. MARK: Fine.

MR. ZUDANS: You say that your assumption is burn is instantaneous?

MR. DILWORTH: Let me clarify that instantaneous.

MR. ZUDANS: How instant is it?

MR. DILWORTH: That estimates that means to me that it is a rapid burn, a fash burn without shock waves.

MR. ZUDANS: All right.

Have you generated something like a pressure package through the containment, or you just compute the ultimate steady state?

MR. DILWORTH: This was not a time history.

MR. ZUDANS: Would you expect that this type of burn would create a very high spike with decay to some steady value state?

MR. DILWORTH: Assume that all the hydrogen evolves to the upper compartments, and the rapid burn occurred in the upper compartment instantaneously with no shock wave, and the resulting pressure --

MR. ZUDANS: How did you compute the resulting pressure?

MR. DILWORTH: We worked backward from the ultimate strength, and came up with a 25 percent metal water reaction. I don't know the total pounds of hydrogen. I don't have it with, but I think that we could probably get it.

Dave Goesser of Westinghouse, is working with us here.

MR. GOESSER: Dave Goesser of Westinghouse. It is a little over 500 pounds hydrogen burn in the upper compartment, as George described. It is a very rapid burn with no variation of the heat synchs that are there in terms of the containment sprays, and only in the upper compartment volumes.

MR. ZUDANS: But the computed pressure is some kind of a static quantity. It is not dynamically computed pressure.

MR. GOESSER: It is not dynamic.

MR. ZUDANS: Your pressure spike as a function of time could be 10 times as high?

MR. GOESSER: No.

MR. ZUDANS: How many times?

MR. CATTON: This is an upper limit, taking a fixed volume, putting some energy in it, and getting a pressure. If you went through the transient calculation you would get a lower -- So this is an upper limit.

If you had a shock wave, it could be higher, but he is not assuming it.

MR. ZUDANS: The burning process will take that into consideration?

MR. CATTON: He dumps in a certain amount of energy, and --

MR. ZUDANS: Right.

MR. CATTON: It is only if you have detonation that you get higher pressure.

MR. ZUDANS: But he says that that kind of a pressure will be reached. Eventually what happens in this transient is unknown. If your calculation said that that kind of pressure was reached instantaneously, or easily instantaneously, then your deformations in the strength of the structure are twice those of static values.

MR. CATTON: The rate at which you reach pressure is important?

MR. ZUDANS: Right. If you reach pressure instantaneously at that level, your deformations are doubled.

MR. CATTON: I guess they would have had to have done that, then.

MR. ZUDANS: I don't know, and that is why I am asking how dynamic this process was assumed?

MR. GOESSER: The nature of the analyses that we have been doing so far have been in terms of conservatively enveloping what would happen from a burn, a continuous burn as opposed to a detonation, is to burn fast enough such that we do not see an impact of the sprays or other heat removal sources which will not be in a regime, at least according to the best data that we have today, that would load structures very rapidly.

The propagation velocities, given the kinds of volumes that you have here, would lead you to somewhere between five and 30 second actual burn duration. At the five second regime, you are in area where the containment sprays will not take out enough of the heat energy to provide substantial mitigation of that.

Data would indicate that that burn velocity will be considerably lower, and therefore the burn time will be higher, therefore, the sprays will become a much more important factor.

MR. DILWORTH: Another point I did not make is that we assumed that it all burned, and it burned down a certain level and stopped. It was assumed that it all burned.

MR. ZUDANS: What you are seeing is that the burning process, although very fast, is slow enough so that you could ignore the dynamic effect?

MR. DILWORTH: That is true.

MR. ZUDANS: If the pressure built up in milliseconds, instead of seconds you would be able to carry half of the pressure in that containment. Five second, I don't know how that relates to natural frequencies, and things of that nature, which are still a dynamic process.

There is some dynamic amplification because load is applied dynamically, and does not grow very steadily and slowly. If it grows steadily and slowly the deformations follow the load. If it grows very fast, the load deformations overshoot the load and come back, and oscillate around that. Eventually they settle down at that level.

Eventually you will have that level.

The question that I have and that maybe bothers me a little bit is how fast this process is, and whether or not you can have faster process. If you have a faster process, you may shoot the head off before you even move the bottom.

MR. DILWORTH: Wang Lau of the Engineering Branch has done some of the work, and I think he could probably -- He has been working on condensers since TVA had Sequoyah, so I will ask him to respond to that

MR. LAU: Wang Lau, TVA. The natural frequency of the containment is around 15 Hz, or something of that nature, so when you talk about five seconds or longer as far as the structure is concerned, it is a steady load.

MR. ZURANS: That is why I was pretty happy with the five seconds, and not five milliseconds.

MR. LAU: Therefore, the instantaneous is very misleading. It is not instantaneous.

MR. CATTON: What is the hydrogen concentration at 25 percent metal water reaction?

MR. DILWORTH: There was about 500 pounds of hydrogen in the containment. I think that we probably have that information with us somewhere here.

MR. GOESSER: It is in the neighborhood of six to 10 percent.

MR. CATTON: Thank you.

MR. MARK: The 28 psi was interpreted, no doubt, questionably at TMI to have resulted from something, I think, close to 8 percent hydrogen concentration all burnt in situ.

MR. ZUDANS: That is what bothers me. Why did that spike not stay there.

MR. MARK: But that would give a scale for what concentration would lead towards working to.

MR. ZUDANS: If the process would proceed the way it is described here, you reach the peak steadily for a while until your mitigating estimates, whatever they are, take it away, but in TMI the spike showed up in a few minutes -- There was not that kind of burning.

MR. CATTON: TMI was over several minutes. The spike looked that way because of the scale.

MR. DILWORTH: The best information that we have been able to get is about four minutes.

MR. ZUDANS: Then I stand corrected.

MR. DILWORTH: The way that we have done this, we have been conservative in what percent metal water reaction we think we can handle. Granted, we did not consider an explosion, we considered a very rapid burn, but in terms of energy release, and taking no credit for reducing the energy release, and there are a lot of things in a ice condenser containment that you would look to do that in the real physical happening, we feel that the 25 percent is a conservative number.

MR. ZUDANS: What is the highest temperature in the atmosphere in the containment in this process?

MR. DILWORTH: I think that it is in one of the informational memorandum.

MR. ZUDANS: I saw a number here like 5500 degrees burn, and I am wondering whether that was --

MR. DILWORTH: What report are you referring to there?

MR. ZUDANS: I don't know what report that is.

MR. DILWORTH: Sandia?

MR. ZUDANS: No. It was a summary to a report.

MR. DILWORTH: We will see if we can find that information.

MR. GOESSER: Our analyses are showing numbers in the range of 2000 degrees Fahrenheit very briefly during the burn.

MR. ZUDANS: Very briefly, and then this very brief is caught by other devices coming in --

MR. GOESSER: Both effects going on, the burn itself has a very high temperature during the course of the burn, and then you see a pressure build up that without any mitigation would rise and hold, but the sprays in these plants, or in the large plants as we have seen them in the

Indian Point studies, are extremely effective in knocking down the peak that comes from this pressure development.

MR. ZUDANS: How does the temperature decay in this case? How quickly does it go down in terms of time?

MR. BRUCE: Bob Bruce of Offshore Power Systems.

The decay from a pressure peak like TMI with spray operating would have a time constant of about 30 minutes or so, so after a few minutes the temperature will have come down very, very sharply.

MR. ZUDANS: What about the temperature?

MR. BRUCE: The same thing. So there is not enough time to heat up the walls of a containment building, or anything like that.

MR. GOESSER: That is correct. That set of analyses has been done in the Zion and Indian Point studies and have shown that there is no substantial heat content to drive the walls up from that high temperature spike that comes with the burn.

MR. DILWORTH: The next step after looking at the capability of the Sequoyah containment as it is constructed was to look at the risk. We made a limited assessment of the risk related to the non-ice condenser plants, and this was discussed already in some detail by Mr. Christie, and then later with the staff, so I will not cover that.

I would like to say again, it should be noted that the ris: of what you called a while "first article of safe," the things that have occurred since TMI that we have implemented into Sequoyah are requirements of NRC plus our own. We do feel in a real sense have lowered the probability of ever getting to the degraded core condition that would prevent this hydrogen. So we are significantly below where we were before Three Mile Island.

We have studied a number of concepts, and these have been gone over before.

MR. ZUDANS: I would like to cover more questions, if you don't mind.

How fast by design do these containment sprays begin to act, in terms of time?

MR. DILWORTH: In terms of time, I guess you are saying from the time that the hydrogen burned?

MR. ZUDANS: That is correct. If it took five seconds for hydrogen to burn completely, at that point we are reaching a high pressure, and a high temperature. Now the only mechanism to reduce these two peaks is by conducting by some basis, putting the spray into action.

MR. LAU: Let me try to answer this question.

I think we were misleading Dr. Zudans a little bit earlier when we said that the temperature was covered out in degrees. It is true that you have a couple thousand degrees Fahrenheit, but it is only when the front of the flame travels. As soon as the flame is gone, the temperature overall in the containment is on the order of 400 degrees, something of that nature, Fahrenheit.

MR. ZUDANS: What does it mean?

MR. LAU: The way that the flame would travel at a speed of around six feet per, somewhere around there, so from the source of emission it is spread out. Suppose you have instrumentation at certain points, then the wave would travel pass that point very quickly at a speed of around six feet per second. So you will have a certain very high temperature for a very short period of time, and then it will be gone.

MR. DILWORTH: Do we have an answer for his question?

MR. ZUDANS: The analysis for heat combustion, and yet you are mitigating containment sprays which changes your pressure from steady pressure to some peak.

MR. LAU: Basically, the containment spray acts like a heat sink.

MR. ZUDANS: You said the time constant is one minute. So in one minute you have three times less pressure, and three times less temperature. Correct?

MR. LAU: Yes, sir.

MR. ZUDANS: In two minutes, you have nine times less. Three times less is still very significant.

MR. LAU: It is from 200 degrees, and not from 4000 degrees.

MR. ZUDANS: But then the burn that travels 2000 that strikes the structures anywhere, even if it does not stay there for a long time.

I am not quarreling with you. I am trying to understand what kind of temperatures might the structures see. That is my question.

MR. DILWORTH: You said that the flame front was at 2000 degrees, and travelled at six feet per second. Immediately after this part of the burn it progressed, then we were down to an overall equilibrium temperature, you might say, in the containment of 300 or 400 degrees, that was the point that Mr. Lau was making.

Your question is, would you still be able to raise the temperature.

MR. ZUDANS: I guess I misunderstood what the peak temperature of 2000 meant before. That is the combustion temperature?

MR. DILWORTH: Yes.

MR. ZUDANS: It is not the temperature of the entire containment atmosphere?

MR. DILWORTH: That is correct

MR. ZUDANS: When you mix all the burnt hydrogen in the containment, the equilibrium reduces to 300 degrees

Fahrenheit?

MR. DILWORTH: In the range of 300 to 400 degrees Fahrenheit.

MR. ZUDANS: That is before you are mitigating the containment with the spray --

MR. DILWORTH: That is ignoring the ice bed which is there as well.

MR. ZUDANS: The ice bed is behind the wall.

MR. DILWORTH: Yes. In reality you are probably going to be burning down in the lower compartments. This pressure could be going up through the ice bed as well. But in our calculation of this pressure we had assumed that it would burn in the upper compartment, and then you would not have the ice.

MR. ZUDANS: There the containment sprays would come on within seconds, or within minutes?

MR. DILWORTH: Within seconds.

MR. LAU: The containment spray will come on as soon as we have diesel. You will be thinking of somewhere around 35 seconds, or something on that order of magnitude.

In our study, we assumed that the hydrogen burn is way on out that kind of target. It is not due in the first 30 seconds or so. If you study the first 30 seconds, assuming the hydrogen burn is very unusually normal, because at that time the oxygen in the lower compartment is swept to the upper compartment in the ice condenser design, and you do not have any oxygen in the lower compartment to support combustion.

MR. ESPOSITO: Esposito, Westinghouse. Just a slight correction, if I may. The sprays are on before the burn. It takes five to 10 seconds maximum time for delay time before the sprays are initiated.

MR. ZUDANS: Mr. Esposito, why would they be on?

MR. LAMBERT: That is part of the ECCS design.

MR. ZUDANS: What would cause the sprays to come on before the burn?

MR. LAU: When the containment pressure reaches 3 psi gauge, the containment sprays would start. This is part of the phase B isolation.

MR. ZUDANS:? It will reach that pressure before it begins burning just because of this turns to hydrogen.

MR. GOESSER: You have a mass in energy that was discharged in the start of any of these accidents before you can get into sequence where the core is degraded, which would give you the production of the hydrogen. Therefore, the sprays are on before you begin to see hydrogen in any quantity in the containment.

MR. ZUDANS: The sprays are on before you start burning. So these calculations are conservative.

MR. CATTON: Why weren't the sprays on at TMI, or do you do business differently?

MR. ESPOSITO: The set points are about 50 percent of design, and not 3 psi.

MR. GOESSER: I thought that it was 4 psi, and theirs is 3 psi.

MR. LAU: In the dry containment you have 4 psi. This is the phase B isolation. The containment spray does not come does not on until phase B isolation, which is high pressure. At Sequoyah, the phase A isolation is about 1.2, which is 10 percent of the design pressure, and then phase B comes on at 25 percent, which is 3 psi.

Over here we are not talking about hydrogen pressure. I think we are talking about the blow- dynamics --

MR. ZUDANS: I want to know how much hydrogen you need to start the spray before burning.

You mean 3 psi of hydrogen. What is the answer? Here is a claim that you can take 25 percent of fuel cladding with a reaction.

MR. LAU: I would roughly guess that 3 psi possible pressure of hydrogen would exceed the 100 percent metal water reaction.

MR. ZUDANS: So it would never reach the containment spray initiation with hydrogen.

MR. BALLENTINE: I am Jere Ballentine. The event that initiated the sprays was the same event that ultimately may produce hydrogen, that is a lost coolant, and the subsequent flashing of the coolant to steam, and thereby the containment pressure that produced the signal "start containment spray system."

MR. ZUDANS: That is okay. That is why you did not understand my question, because I was one phase beyond that.

That means that it is spraying there, and it is running, and it starts burning. It should not be able to reach it.

What is the real estimate, if you take the mitigating devices into account, how far your pressure would go and similar combustion if your spray is on from the very beginning?

MR. DILWORTH: As I said earlier, we have not calculated, we took a very conservative approach, we have not sophisticated our calculations enough to take credit for all the conservatism in it to see what we feel like the actual metal-water reaction that we could withstand, but we think it is significantly more than 25 percent.

Are you asking us questions now just a little bit beyond where we are? I guess.

MR. ZUDANS: Well, I am asking for a best estimate. What you gave is like in the moderation model type of answer, right?

MR. DILWORTH: Yes.

MR. ZUDANS: Because I also will ask another question. What if it didn't come on spray?

MR. DILWORTH: Well, if it didn't come on we can take 25 percent for the water reaction.

MR. ZUDANS: Yes, but could you then -- if you didn't come on you would not be able to reduce the pressure and temperature. Your time constant or reduction would not be one minute, right?

MR. DILWORTH: Well, if we had no means to mitigate the pressure, and then we have calculated 25 percent metal-water boil reaction based on the ultimate strength of the vessel. Therefore, the next thing would happen that you would have a failure of the containment vessel.

We feel that looking at the vessel itself if this happened it would be along the spring line of the vessel. Now that is --

MR. BRUCE: Bob Bruce, Offshore Power Systems. I wonder if I could clarify a point here. The 25 percent number is approximately an adiabatic calculation. It takes very little -- -- spray, you might say approximately no -- -- spray at all. If the spray is operating, the final pressure will be a function of how long the burn occurs over. If it is a five-second burn it won't be much different from the adiabatic calculation. If it takes 30 seconds you will get more effect from spray evaporation, and finally, if you have something like a small break loss of current like TMI, say two inches, the release of hydrogen could be such that you could indeed take quite a lot more hydrogen burning.

So the actual pressure is going to be very, very much a function of how the hydrogen is burning. That 25 percent is just about the worst case as I understand it.

MR. DILWORTH: We feel there is an awful lot yet that needs to be done and to understanding the rate of hydrogen burning in different atmospheres of steam, fogging, sprays, combustion products left over from the earlier burns of hydrogen, and so far this has not been done in the industry, but the result of all these we feel will raise the 25 percent to some number significantly higher.

We are here today telling you what the minimum we feel like we could say we could handle.

MR. ZUDANS: Well, I have a feeling that I have no reason to be dissatisfied with what you computed. It is all right now, since I understand it.

But what bothers me a little is at that point, when you reach that point and say, I reached that pressure, I can take 25 percent water-metal reactions, and then for all this calculation you didn't have the spray on, right; it was adiabatic process?

MR. DILWORTH: Yes.

MR. ZUDANS: Now I am saying what if the spray didn't come on and stayed on. You would not take that much water-metal reaction. You would have to reduce it. Of course you rely from that point on, the spray will effectively create something like a one-minute time constant of pressure and temperature, right?

MR. DILWORTH: Well, my answer to you was if you had no other means of cooling after you had burned 25 percent metal-water reaction you would reach the ultimate strength of the containment, and the containment would be on a 50-50 probability of failing at that pressure.

MR. ZUDANS: Okay, that means you cannot take 25 percent unless you can account for an existence of water spray.

MR. BRUCE: George, I would like you to just verify what I am going to say here. I think that if there was spray action at all, the number you were producing was about 20 percent metal-water reaction. So you took some credit for spray but not a lot.

I would also like to make another point. You mentioned earlier the time constant of the sprays of the order of a minute or so.

MR. DILWORTH: Right.

MR. BRUCE: If it is not operating with time constant of the walls, will it be of the order of about 5 to 7 minutes, so you will still get a reduction from the adiabatic pressure peak even if the spray is not operating on a slightly longer time constant?

MR. ZUDANS: Except that I wouldn't be very happy with a time constant of walls, because the walls will start seeing higher temperatures. May not be in trouble, but locally, right.

MR. BRUCE: Yes, but if you look at something like the shell, for instance. Take the case where you are bundling about a 20, 25 percent zirc-water reaction. The equilibrium increase in temperature of the containment shell would be something of the order of about 100 degrees

Fahrenheit. You indeed may get higher temperatures on the surface for a short time, but that will rapidly decay into an equilibrium increase of no more than about 100 degrees Fahrenheit.

MR. BRUCE: Yes, that means that without -- then you say you could likely survive 20 percent, and with spray you can certainly survive more than that, more than 25.

MR. DILWORTH: Well, okay, and in reality we also believe you can stand a little lower because of the fact you do have other heat transfer, as they are spoken to back there.

MR. CATTON: As long as we are talking about heat transfer, as I understand it, your air conditioning or containment cooler system has a turnover time of about 30 minutes. Is that correct?

MR. DILWORTH: Not the -- You are talking about the recirculation fans between the upper deck and the lower compartment that --

MR. CATTON: Whatever is moving the air in the upper volume.

MR. DILWORTH: I believe this is 20 minutes or 30 minutes.

MR. CATTON: I saw 30 minutes somewhere.

MR. DILWORTH: Complete containment air change from lower to upper compartment in what period of time. Due to the recirculation fans operating which then --

MR. LAU: Let's see, 40,000 cfm, which is -- as to every -- rate.

MR. DILWORTH: About 25 minutes.

MR. LAU: Something like that.

MR. DILWORTH: It is in the range of 20 to 30 minutes, as I recall.

MR. CATTON: Your hydrogen source is down somewhere below, and that is going to be very cold air coming up through the ice condenser. It is going to stratify. So you are going to have, if you have 8 percent distributed throughout the upper volume, if you mix it, and you consider that it is not going to be mixed, gee, I would think you would take half the volume in the bottom and put 16 percent in and then detonate it. It would be more reasonable and more in line with what you are doing.

MR. DILWORTH: Okay, you are saying --

MR. CATTON: I am saying the cold air is going to lay in over the top, across the top of the deck.

MR. DILWORTH: On the floor of the deck --

MR. CATTON: Right.

MR. DILWORTH: -- in the upper compartment?

MR. CATTON: And you are going to have a higher concentration of hydrogen in that air coming up through your ice condenser. You have condensed out all the steam. You may have an explosive mixture. And it takes you 30 minutes to mix it, 25 to 30 minutes to mix it, assuming that you have a nice means for getting it all out and stirring it up.

MR. DILWORTH: These fans are taking their --

SPEAKER: It is mixing all the time, Ivan.

MR. DILWORTH: These fans are taking their suction from the top of the containment dome, is this not correct?

MR. LAU: Every ton fan has a skimmer system to go with it, and the skimmer system is designed to circulate and for that particular purpose. It sucks from those houses, what do you call it, the, you know the steam generator and closer, personalized enclosure, and picks up suction from them and blow it out.

So in that area that you are concerned about, we probably have better mixing.

MR. CATTON: Do you take all the air coming up through it, through the recombiners?

MR. LAU: I wouldn't say all the air, but we do take suction from those hard-to-get-to areas.

MR. DILWORTH: And those areas of suction there are also embedded within concrete, all of the duct work is, so it is protected for --

MR. CATTON: Well, how long does it take to pass the contents of the containment through the --

MR. DILWORTH: Between decks?

MR. CATTON: Between the hydrogen recombiners. If you are taking it out and you are taking --

MR. CATTON: Well, the hydrogen recombiners are a prestanding unit that are -- wait a minute.

MR. DILWORTH: I believe a better question is what percentage of the upflow and the ice condenser goes through the skimmers, because that is all that is going to get mixed.

(Pause.)

The hydrogen recombiners does not have any ducted air going to it. There are two hydrogen recombiners and they are essentially a free-standing unit in the upper compartment of the containment, and as Dr. Lau was saying, the recirculation fans are taking suction from all of the places where you have the high points.

So we feel, as we stated, we probably got very good mixing in an ice condenser containment.

MR. CATTON: Well, cold air, having cold air in one place and hot in another is going to stratify, and the hydrogen is not going to separate out of the cold air.

MR. DILWORTH: You are saying the hydrogen is going to stay with the cold air.

MR. CATTON: Yes.

MR. DILWORTH: And remain along the --

MR. CATTON: Well, I don't know, I am asking you. It just seems to me that you have an air cooler, which is what your ice condensers are going to be, that could lead to stratification unless your overall circulation system can be assured of stirring it very well, and just saying that it turns over in 30 minutes doesn't guarantee that it is there.

MR. DILWORTH: Okay. Well, I am not prepared, unless --

MR. CATTON: I am not prepared to pursue it any further either.

MR. DILWORTH: -- unless the Offshore Power people have some information on the velocity along the floor. Has anybody got any --

MR. GOESSER: Goesser, Westinghouse again. There are several things that I guess ought to be considered here. First and foremost is you move into things like stratification. What you must be doing is tracking the sequence of the events in a way that is physically real, that accounts for the fact that you are blowing things down, moving things through the ice condensers. If for example you have the sprays on with the kind of water flows that you get out of the sprays in that upper compartment, that alone is going to tend, if not destroy any chance for stratification.

Beyond that the data that the Germans have produced in some of the experiments over there with respect to hydrogen in containment has shown extremely good mixing with no fluid mechanical driving forces that one would get as a part of a blowdown arrangement.

I think the key is as you begin to consider these things you must do what George had said the TVA people are doing right now, and that is developing physical models of what is going on to show how much additional capability beyond this 20 to 25 percent range that the Sequoyah containment has.

MR. ZUDANS: What physical models are you talking about TVA developing?

MR. DILWORTH: We are going to get in, as we --

MR. ZUDANS: Oh, you are going to this.

MR. DILWORTH: A little further into the end of this we will get into what we plan for development.

MR. ZUDANS: Okay.

MR. DILWORTH: Okay, if there are no further questions on mixing --

MR. CATTON: I guess there are. They are just not answered.

MR. DILWORTH: If there are not any further answers on mixing, we will proceed here to just outline briefly the different concepts that we studied as candidates for mitigation or control or prevention of consequences from hydrogen, and they basically fall into three different categories.

One that would mitigate the consequences, that would mitigate it by venting the containment in some way, one that would control the combustion, and the other would be prevent combustion.

Under the vented containment we looked at filtered vented containment, we looked at added containment volume that you would vent to, and we looked at coupling both unit containments together.

Under control combustion we looked at control ignition sources to burn the hydrogen as it is produced.

Under prevention of combustion we looked very carefully and at much detail at an early containment with nitrogen, and suppressing combustion in the containment with halon. So those are what we have looked at.

I would say a few things about each of these concepts. Vented filtered containment, we found that it was not effective for rapid pressure transients. The size of the vent line that you would have to have out through the filter bed and everything would not be effective for mitigating the fast buildup of pressure.

Furthermore, we estimated the dose in the low population zone with a filtered vented containment could be in excess of 900 rem.

The third item here, there are some essential features of this containment that are not demonstrated yet for filtered vented containment. And then the potential for unnecessary by-pass of the containment by operator actions, under accident situations where he should not have done it, you know, is something you have to face.

And then the high initial cost of this concept is also a factor. However, it does have a moderate operation and maintenance cost.

Additional containment is not effective either for rapid pressure transients. It does minimize the radiation release to the public.

MR. ZUDANS: What do you mean by additional, what is additional containment?

MR. DILWORTH: Another containment vessel built next to the existing containment vessel.

MR. ZUDANS: Is it equivalent to saying a large -- -- to double volume?

MR. DILWORTH: I did not hear you.

MR. ZUDANS: You mean building another building and connecting them?

MR. DILWORTH: That is right.

MR. ZUDANS: With many holes?

MR. DILWORTH: Kind of like the Candid except it is not a vacuum.

MR. ZUDANS: No roof on this one, I guess.

MR. DILWORTH: Just be another containment vessel with nothing in it. You could use it as a . oreroom if you wanted to, I guess, but it would normally be just an open volume. Make up for the fact that ice condenser containment is a low volume containment.d

We feel that this type of approach would minimize radiation release to the public. It is very high in initial cost and time required to implement it. It would have a low operation and maintenance cost.

Coupled containment would be taking the two units at Sequoyah that we now have and building a line to couple the two together, a pipe so to speak. And here again like the other two, it is not effective for rapid pressure transients, and it has the potential of degrading the safety of the second unit because you essentially now have a common containment on the two units.

It would cause us to have a large operational penalty for the second unit. It would minimize radiation release to the public in the event of a hydrogen, pressure buildup as a result of hydrogen.

MR. MATHIS: Pardon me, George. When you say back on the filtered containment, not effective for rapid pressure transients, elaborate on that a little bit if you will. I don't understand why not.

MR. DILWORTH: The time required to transport the -- relieve the pressure would require to keep it below the pressures that could get you up to ultimate strength on the containment are such to be impractical to build a line that large.

We looked through several size lines. I don't know if Wang or Dave can speak to the line size that we used. But it was getting up to the point where it is completely impractical. You know, like you were maybe have to build something 18 feet in diameter or something to be able to --

MR. MATHIS: Were you relying entirely then on the vent, or whatever it is --

MR. DILWORTH: The vent, right.

MR. MATHIS: -- ductwork, to relieve your pressure?

MR. DILWORTH: That is true.

MR. MATHIS: Not your sprays that you are talking about on the other system, as an example?

MR. DILWORTH: Well, the sprays of course --

MR. MATHIS: If you got a LOCA or whatever it is, your sprays are going to go into action, you are going to have a pressure surge just by the flashing of the steam or the water and the steam, so you are going to get a pressure surge of some kind.

Then you could have a vent system, multiple vents, I would think, that would open up and tend to relieve your pressure some more --

MR. DILWORTH: On a slow process it would. If there was a slow pressure buildup it would be effective. On a rapid pressure buildup it would be ineffective. It is the pure hydraulic problem of ability to be able to --

MR. MATHIS: Well, I guess what I am having trouble with, I can see a pressure reduction before you get to this hydrogen burn problem.

MR. DILWORTH: Oh, okay. You are saying your steam breaks --

MR. MATHIS: In other words, you are looking at one pressure spike rather than other things. Okay.

MR. DILWORTH: You are saying you would start with a lower containment pressure because you had vented containment, is the point you are making.

MR. CATTON: Is the pressure spike associated with the vessel failure --

MR. DILWORTH: With vessel failure.

MR. CATTON: -- as large in an ice condenser as it is in the large dry containment?

MR. GOESSER: We are looking now strictly, and the things that George is talking to here are the hydrogen burning phenomena associated with degraded core. We are in the production of hydrogen right now.

MR. MATHIS: Just that incident, and all the other buildup and so forth is irrelevant in this particular analysis?

MR. GOESSER: Well, the buildup to it is very relevant. And the thing that was done within the Sandia study, and I think if Matt is still here he can confirm that, the failure probabilities that were used in that Sandia risk study that was talked to a little earlier with a hydrogen burn were very, very high. They were essentially the residual, like .99 or so, that came out from alpha after you had subtracted out the alpha and beta failure modes, which were low.

So that on production of hydrogen, essentially the assumption that is made is that you would end up with a containment failure leading to one of the categories of releases.

Within the analyses that have been done, if you burn the hydrogen very quickly, and you burn enough such that you are going, substantially being, you know, in the 10 to 20 or 50 percent more than the containment pressure on this fairly simplistic analysis, then you very rapidly get into vent rates for these kinds of containments that run in excess of half of a million cubic feet. And this is for an ice condenser.

In order to chop the peak enough to keep it within the ultimate capability of the containment. Again, remembering the spray time constant being in the neighborhood of a minute, a 5 second to 10 second burn you will not get much chopping of that peak from the sprays, although you will get some.

So that in order to mitigate a very high pressure pulse in this kind of a containment you require a very large vent capacity, which is what George put up there to begin with.

MR. DILWORTH: We assumed, correct me if I am wrong, Dave, that we started out with essentially atmospheric pressure in the containment before the hydrogen burn. So Dr. Mathis' question about taking credit for the venting preceding the hydrogen, we did take credit for that.

I guess I would like to repeat that, Dr. Mathis, that we did start out assuming atmospheric pressure in the containment for the hydrogen burn, which would have assumed that we have already knocked the pressure down, which could have been done by venting in the case of a vented filtered containment, or containment spray or other means.

So in that sense we are not saying that a vented filtered containment would not be affected with a lower pressure on a steam situation. But as far as hydrogen, what we are saying is it is not effective for a rapid hydrogen burn. In fact, none of the additional containment concepts are that we looked at.

MR. MATHIS: Well, I guess the problem I have is that long before you get to the hydrogen burn you have got a pipe break. You have got some water flashing into steam, and you got a lot of other things going on that takes time and quite a bit of time, up to -- what, your core melt here on these charts a while ago was something in the order of 180 minutes. Well, that is three hours. And then you begin to get into core degradation and then you begin to get into the potential of hydrogen burn.

I guess my problem is that there is a time sequence in here an we are jumping from this to an instantaneous kind of thing. At least that is the way it looks to me.

MR. LAU: Well, one other factor we must consider in any sort of venting of containment, whether it be coupled containment, additional containment or filtered vented containment, is that if you burn out the contents before the hydrogen comes into containment, and as Dave said a moment ago, if you burn out half of it, basically what you are doing, you are stifling the hydrogen concentration, because now you have less mass. So the volume fraction you just double because of the venting. So that is not a very good thing to do.

MR. GOESSER: I think there is one other thing in terms of your time sequence, Dr. Mathis, and I think it is important to make sure it is clear.

You have a situation where you have a transient that starts and you build some pressure based on, for example, the small break LOCA. All right, you can have your ---- work and take you back down to your atmospheric condition. If you now predict at the point that you reach the concentration that you choose to use as your burn percentage, or you wait even to core melt, whatever you ignite as long as you are igniting a substantial quantity of hydrogen and using the kinds of assumptions that George had mentioned before, you get a very sharp pressure rise.

Now if that pressure rise is going above the ultimate capability of your containment, you must take out enough in time to chop this peak off that is coming strictly from the hydrogen burn even if you have started it from atmospheric pressure within the containment.

All right, and that is the fundamental point that George is trying to make, that if you have that very, very rapid pressure rise coming from a hydrogen burn, you get into very quickly areas where the amount of venting that you need or the vent rate that you need becomes very, very large.

In the large dries some of the studies were showing that one might need a 20-foot diameter vent. Here you are looking at something that is down considerably from that, but nevertheless in excess of something like a half a million cubic feet per minute.

MR. ZUDANS: I can accept it very easily, even if you have the whole wall open and the other side might be overpressurized. It is the inertia effect again, discharge effect. This is correct.

MR. DILWORTH: And we are not saying that the filtered vented containment might not be definitive for some sort of a Class 9 accident. But we are saying as for hydrogen, the filtered vented containment will not protect your containment, and we feel that the filtered vented containment has risks to the public that are also very important to consider, which is the dose that you would release and the capability you put in the operator's hand now to breach a passive barrier that this plant has been designed for.

We found that the nitrogen inerting would be very effective in preventing hydrogen combustion. It is largely a passive system, but the most significant thing here, it is very difficult, it is not impossible to backfit to an ice condenser. Operationally it is prohibitive because of the frequent maintenance needed on an ice condenser and other containment systems. Has significant potential for degraded through reduced maintenance on equipment that you have in an ice condenser type plant that needs daily operator attention. Increased loss of ice due to sublimation of purging and refilling with nitrogen. Very high initial cost and extremely high O&M costs.

I would like to say a few -- I could say more detailed into inerting if someone wanted me to. We could speak to this probably for somewhat longer, but we did a lot of work in trying to find ways to modify the ice condenser containment.

We were finding we could add a lot of penetrations to the containment. We would take about two years probably to modify it, and then once we modified it we would still be looking at coming down probably 16 times a year for required surveillance and maintenance, which means that many inerting and deinerting processes a year.

MR. MARK: This is to keep the containment, the inertia as a standard thing?

MR. DILWORTH: Right.

MR. MARK: There is another notion of inerting, which I guess you will get to, starting to inert as soon as you think you are in trouble.

MR. DILWORTH: That is what we are looking at the halon system for.

MR. MARK: You could also look at nitrogen.

MR. DILWORTH: Well, there is a lot of bad actors about that as far as health hazard and killing operators and this kind of thing if you did that.

This gets us to the halon. We believe that halon could be potentially effective as preventing hydrogen combustion. Has no significant operational effects with normal precautions taken. It does have a moderate hazard to personnel. It hasn't been technically proven or demonstrated that it could be used as a post-accident spray. We have quite a few concerns about the decomposition products that could be the result after a suppression of a burn and its effect on a post-accident situation, such like you have at Three Mile Island now, material attacks that would occur from halon products.

Active post-accidents with short reasonable times to manually activate it. It is high initial cost and low O&M costs.

But we think that it has enough potential to do further work to find out about some of the uncertainties. And so we intend to do this.

MR. ZUDANS: Is it feasible to think that you could activate it fast enough to be able to accept higher percentage of water-metal reaction than the one computed without it?

MR. DILWORTH: Well, you know we were looking, we were talking a while ago about the time required for the core to get to the point that you would have hydrogen being produced -- 150, 180-minute range -- and so there is a lot of time there to know I should turn on the system.

What we don't know, some of the things we don't know is how long this will stay in the atmosphere effectively, the halon, what effects sprays will have on it. But we will have time, we feel like we would certainly have time to be able to initiate it and get it in.

MR. ZUDANS: In other words, your idea is that it would be introduced as soon as you recognize the hydrogen that would be generated, not after it starts burning?

MR. DILWORTH: That is correct.

MR. ZUDANS: Well, under those conditions you could do it, Dr. Marks has also pumped in the nitrogen.

MR. DILWORTH: That is true. It is time required to get in there with the nitrogen too, that it takes something like 10 purges, I believe, before we get it down to a noncombustible atmosphere.

How much time is that, Wang? The purge we have to go through before we can fill it up completely with nitrogen normally with the system we were looking at?

MR. LAU: I think there are two questions here. One is post-accident use of nitrogen. That does not work because you still have the oxygen in there to support the combustion.

Okay, to inert the containment you have to purge out before the accident, you know that you have to inert the containment with no oxygen in a normal operation, and that, using a containment purge system, it takes about, oh about three and a half hours to inert the containment. I did a test for inerting, and I guess George is going to cover it. You know, a small motor and then you bring the reactor down, bring the reactor up and so on and so forth. It would probably take about 6 to 5 hours, something like that, for each cycle, from full power to full power.

MR. DILWORTH: You have all the fission products inventory and gasses that you got in the containment --

MR. ZUDANS: I think you made good point because the oxygen is still there, even if you introduced the nitrogen.

MR. DILWORTH: That is right.

MR. ZUDANS: However, in a halon case it suppresses the combustion --

MR. DILWORTH: It is a chemical reaction.

MR. ZUDANS: Right. So that is a substantial difference.

MR. DILWORTH: The most promising concepts that we have found after looking at the concepts that we have outlined here and those that we have selected for a rigorous development program are addition sources in halon. We also feel that we have to do significant improvement in the physical models and computer codes needed to understand the accident sequences and the rate of production of hydrogen.

We have stated here that the filtered vented containment is unacceptable because of release dose and other aspects of it not being effective for holding down the pressure.

Inerting is not feasible for an ice condenser containment for various reasons. Risk at Sequoyah we feel is comparable to WASH-1400 reference plant.

MR. ZUDANS: Are you going to talk more about these ignition sources?

MR. DILWORTH: Right, just getting ready to. Before I talk about the ignition sources, I would like, if i can get this on here.

It is a little more detailed than what we have done at TVA and are doing now, is that we have organized an eight-man full-time task force for design and development work on degraded core accidents, and these men are working right now full time on the hydrogen issue primarily.

We are implementing immediately the design and installation of an interim distributed ignition system, which is going to be put in in three phases which I will speak to in a moment.

We hope to have that operational within the next two to three months, but this of course subject to a rigorous safety review by TVA inhouse and by the NRC staff within the next month.

We are implementing also development work now to go parallel with this to upgrade this interim system to what we would call Phase 2 to backfit any improved aspects of the system that can be developed, and we see this Phase 2 system taking about, or work taking about a year from now.

We will complete a long-term study also on the development effect for controlled ignition systems which can lead to further backfitting of the Phase 1 and 2 systems if it is needed; and we think it will probably take two years or maybe a little longer to get this work done.

We are implementing immediately a development effort to understand the potential negative aspects of halon to see whether or not we have to wash it out as a potential candidate for a suppressant system.

Some of the things this task force are going to be looked at again in a little more detail is control ignitions, halon, risk assessment, core behavior, hydrogen generation and transport, hydrogen burning and containment responses, containment integrity, capabilities, equipment environmental qualifications, radiation dose codes, hydride converter, fogging and others. Hydride converter would be something we would hope we might be able to put on the discharge of the reactor coolant drain tank vent on also on the reactor pressure vessel vent, the two most probable paths of hydrogen and prevent it there before it gets into the containment.

And then we intend to follow the rulemaking and the state of the art that everyone else is doing.

MR. MARK: Could I just ask, you speak of hydride converter. What sort of thing crosses the mind -- I don't want you to describe its action -- halon or --

MR. DILWORTH: I will ask Dr. Lau to --

MR. MARK: Is this a metal of some kind?

MR. LAU: Yes. Ever since the early 1950's a lot of people have done research in this area for some, Oak Ridge have done, alumina, platinum alumina. And then they have done some heated copper oxide and things like that.

MR. MARK: Okay.

MR. LAU: What the task force intends to do is not to do everything ourselves. We are probably going into a kind of program management and get a lot of consultants, and we are in the project definition phase right now and trying to define who is going to do what.

This is one of the things they will have to do for us, I think.

MR. MARK: Is look at that work on hydrogenation and see if there is something useful?

MR. LAU: Yes, sir.

MR. MARK: And then the other question, if you go to -- perhaps you said and I missed it -- if you go to an ignition system, how much hydrogen can you live with then?

MR. LAU: This is one of the things we will be studying. Just offhand, I believe that Westinghouse has made a study, and I think if you use an ignition source, you know, and burn it, I think come off you know kind of slowly we can probably take about 75 percent metal-water reaction.

MR. MARK: Right. You still couldn't stand 100, which of course you probably don't feel is necessary anyway.

MR. DILWORTH: We can take everything out to the core melt through the vessel and follow it into the sump.

MR. MARK: And then it wouldn't be hydrogen burn, it would be other gas?

MR. DILWORTH: Yes, you have got another scenario. Actually we don't believe that you can get more than 70 percent metal-water reaction before you get to that due to the water available.

MR. MARK: And so you would be able to at least double and perhaps triple the amount of hydrogen that could be allowed in if you could burn it continually?

MR. DILWORTH: Essentially we are saying --

MR. MARK: Or as it came?

MR. DILWORTH: Dr. Mark, we are saying anything that can be produced in the vessel without it leaving the vessel we feel like we can handle. We are not saying that we can handle a meltthrough and the results of that when you have the remaining --

MR. MARK: Good enough.

MR. DILWORTH: Just saying a little more about the interim distributing ignition system -- we are calling this interim because that is essentially what it is. We found a means that we feel like we can implement in a short period of time, a period of two or three months, a system that can be operational by the time that Sequoyah goes to any significant power, and this system would be made up of commercially available ignition sources of various types, and it would not be a safety grade system.

But it would be operational, and we would within the next year be studying ways to improve that system in what I described a moment ago as Phase 2 system where we are looking at a number of different types of igniters. We are looking for better qualified ways to qualify the system for safety grade, looking at environmental effects on the system, better understanding of the hyirogen burning and suppression characteristics of the sprays and so forth we have in the containment.

So there is a lot of work we have to do in the next year that we feel like we can improve this system with and then there is other longer term R&D work which we call Phase 3 that might have some good benefits to the control ignition system and might also allow us to go ahead with another system, halon, as well.

But to conclude, I would like to say that our view of the hydrogen issue at Sequoyah and our other nuclear plants, we really believe that because of the low risk in comparing Sequoyah with other PWR's, and Sequoyah is safe to operate at full power based on its present capability of its ice condenser containment and the substantial improvements in equipment and training which are being implemented, and we believe that the additional reduction of overall risk may be achieved also by protecting the containment from the consequences involved in the metal-water reaction and hydrogen releases.

And for this reason we have begun a design procurement safety review of an interim distributing ignition system, the details of which will be submitted to the NRC staff sometime this month.

We have further committed to development efforts to improve the performance and safety grade qualifications of this system over the next two years. So that concludes what I am prepared to say today unless you have further questions.

MR. ZUDANS: Do you have more information on the commercially available ignition sources that you plan to install in Phase 1? How do they function? Do they have to throw the gears too then to ignite it?

MR. DILWORTH: No, they are exposed out into the atmosphere. They are very --

MR. ZUDANS: They are just at high temperature point, that is all?

MR. DILWORTH: Dr. Lau can speak to this. I could, but I will let him go ahead.

MR. LAU: We are in a procurement stage. So until we buy the equipment we cannot tell you exactly what we got. Okay? But right now we are thinking of two types of igniters, about half and half. That is what we are thinking of. One type is just a thermal type -- you think of the stove, the electric stove at your home, you flow out about 1500 degree Fahrenheit. Emission temperature is only about 1100 or so. So you have some margin.

It is passive, and all you have to do is put the juice to it and there you go. Okay?

The other type is what we call the spark probe type. That is an example, is the one developed by B&W and they use in, you know, starting an oil-fired furnace. And basically the probe itself, one more that we are looking at, is about a foot long. It is not a spark plug you know. It is about foot long, about, oh, half an inch in diameter, with a central wire.

And what happens is that all you do is -- use 110 volts AC and -- -- to about 2500 volts, and I put through the rectifier and charge up a capacitor, and the capacitor would be charged at about 2000 volts and send a spark across it.

One more that we are looking at is -- spark across about 15 times a second, and each spark will produce about 12 jules.

Now you probably know that in ideal conditions it would take only about one-tenth of a millijule to ignite hydrogen. So you have about a degradation factor for the system of about 120,000 times as far as energy is concerned.

Okay?

MR. ZUDANS: In other words, you will have to then distribute all these ignition sources strategically all over the volume?

MR. LAU: Yes, we intend to put that igniter at various locations in upper and lower compartments.

MR. ZUDANS: I have another question if I may still ask it. How many years has D. C. Cook malfunctioned for the ice condenser?

MR. DILWORTH: You had better ask the staff. I am not sure.

SPEAKER: I believe they started it in 1975.

MR. ZUDANS: Has the ice condenser been ever challenged, and if so to what extent?

MR. SCHWENCER: Al Schwencer, Regulatory staff. I am not aware that the ice condenser system has been challenged, and I can't give you the exact date that D. C. Cook I first went into operation. We could get that for you easily.

MR. ZUDANS: Is the ice condenser continuously monitored for blocked or unblocked situation? Is the air circulating continuously through it or no?

SPEAKER: No.

MR. ZUDANS: Okay, that means that there is a stagnant area --

Is there some regular procedure by which the blockage of the ice condenser channels is verified?

MR. DILWORTH: Are you asking that of staff or us now?

MR. ZUDANS: It doesn't matter.

SPEAKER: We can answer that.

MR. DILWORTH: I think Jerry Ballentine, the superintendent of Sequoyah, could probably --

MR. ZUDANS: Yes.

MR. DILWORTH: Oh, he wants to ask somebody else to. He is doing it. That is the reason I thought I would ask him.

MR. ZUDANS: Are you from D. C. Cook?

MR. DILWORTH: Oh, is this question on D. C. Cook?

MR. ZUDANS: Well, you don't have experience in any other plant, just D. C. Cook.

MR. DILWORTH: We have been operating the ice condenser now for 18 months or longer.

MR. ZUDANS: Oh, but that is a kind of -- -- operation.

MR. DILWORTH: That is the only thing we have been operating, but we have been operating it.

MR. ZUDANS: You can go off in the louvres and look down, but I am wondering how it is done in D. C. Cook.

MR. BUTLER: I was involved in D. C. Cook but I don't remember.

MR. ZUDANS: Dr. Butler?

MR. BUTLER: Yes. Well, on D. C. Cook it was very much like Sequoyah. There will be a requirement to once a week enter the containment and examine the ice condenser louvres, assure that they are functional; that is, that they are not frozen shut.

MR. ZUDANS: Well, I am more concerned about what happens to 60 or so feet of ice. There could also be complete -- partial melting, complete -- of all that cross-section. It is like a cork, it could be like a cork. It doesn't have to be.

MR. BUTLER: Well, there are the programs for periodically measuring the amount of ice per basket. So what they do is move from one basket to the next in each inspection sequence.

But I think this is very much the same as Sequoyah, and they should be able to describe their plan for weighing the ice in each of the baskets.

MR. ZUDANS: Okay. I saw that done in McGuire, right. That is okay. But what I am concerned, I know that the bets have been made and there is a great deal of assurance that the ice will not collect itself and block the cross-section any place, which will force the discharging steam to form some channels, right, and much of ice may not be used to hold down energy release.

Now in order to be sure that none of the baskets is blocked in any of the cross-sections for all those high peaks, there has to be some way of circulating something through the ice, see whether it goes through.

Closing and opening the louvres only prevents freezing up the louvres, right?

MR. BUTLER: Yes.

MR. ZUDANS: It doesn't tell you anything what happens and what it is, 60 feet longer -- stack? What happens if any level in the stack? Is the cross-section free or is it blocked?

MR. BALLENTINE: Perhaps I could answer the question. I am Jerry Ballentine. We do have a periodic visual examination. You actually from the intermediate deck doors on top can view the air space or the air passage between the baskets and can see from top to bottom from the top or the intermediate deck doors.

We also periodically weigh the baskets by lifting them, physically lifting them, and putting them on a scale and weighing them for ice weight.

MR. ZUDANS: So you can see the outside diameter of the baskets. You can see visually that the baskets are not connected?

MR. BALLENTINE: That is correct.

MR. ZUDANS: But you can't see what is inside the basket, but I guess you don't care because this has been shown previously that even if he ice was solid in a basket it would still suppress the pressure.

MR. BALLENTINE: Yes, and we know that there is the inventory of ice required there by the weighing.

MR. ZUDANS: Okay, all right. Well, thank you. That is nice to have the ice, yes.

MR. MARK: Thank you. I am a little relieved that the description of, not the igniter, but the things considered for igniters. I was thinking of a sanctuarial in Mexico where they would have all of the candles rounded up from Brown's Ferry burning all the time.

(Laughter.)

Mr. Stahle, should NRC look at the prospects here on this subject?

MR. STAHLE: Dr. Butler will give his presentation.a

MR. BUTLER: I just have a brief presentation here to follow up what was said last month at the subcommittee and full committee meetings.

There are two parts to what I have to say this afternoon. The first part deals with what the staff intends to do as a result of the upcoming TVA proposal to design and install their interim distributed ignition system.

The second part will describe in greater detail than previously the contents of the user's request where the Nuclear Reactor Regulation Office intends to request that the Office of Research undertake some supporting study to provide the data base for the upcoming rulemaking proceeding on degraded cores as well as provide specific accelerated studies on the ice condenser plant and the Mark III BWR plant.

As the TVA representatives earlier indicated, they intend to very shortly propose the design and installation of the distributed ignition system. The staff's view is that the Sequoyah station as well as other ice condenser plants are acceptable for full power operation pending the upcoming rulemaking proceeding.

Nevertheless, they do represent a special situation on the basis of the information contained in the Commission Report SECY 80-107, in that they do not have very large margins in their capability for accommodating substantial metal-water reactions as a result of degraded core.

Nevertheless, with the proposed addition of the distributed ignition system, the staff feels that it has the potential for improving the capability for accommodating more metal-water reaction.

The staff will have to undertake this program of studies to assure itself that if installed the distributed ignition system will in fact enhance safety, will in fact improve the margin of safety.

In that regard we have a program here to extend over the next three months or so, the object being to develop a data base or an information base for assessing the distributed ignition system.

It is broken into two parts, the first being the experimental tasks. We intend to have either the Sandia Laboratories or the Lawrence Livermore Laboratories assist the staff in this effort.

We expect to acquire the ignition sources that TVA plans to use and ignite various lean mixtures of hydrogen-air systems and if possible steam systems as well to assure that the ignition characteristics are as we would want them to be.

We also intend to evaluate the capability of hydrogen instrumentation system.

Sorry. Dr. Sudans?

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NRC ACRS
Sequoyah
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Babineau/
Hatfield

MR. ZUDANS: Isn't TVA going to do exact same thing? Why do we have to duplicate? Why not join forces and watch what they do?

MR. BUTLER: Well, it's my understanding that TVA does not intend an experimental verification program.

MR. ZUDANS: Oh. They just will install it, right?

MR. BUTLER: They will assess the design based on available --

MR. DILWORTH: That's not correct. We do intend -- I thought I spoke to that earlier, that we do intend a longer-term verification program than -- and so -- but we definitely would like to work with the staff on anything we do or they do that we can use their information or they can use ours, and we will stay close enough to them that we won't --

MR. ZUDANS: You know, Dr. Butler, I am not critical of what you might do. I thought that you could increase the scale if it's a joint effort, you could do a better job.

MR. DILWORTH: It's possible they can do some things and we can do other things that --

MR. ZUDANS: Complementary, yeah.

MR. BUTLER: You know, the difficulty in these discussions is, we expand and contract the scope of discussion and we often get into difficulty. I meant to make that comment earlier, that when TVA was describing all the alternative mitigation devices they were talking in terms of the longer term, and yet sometimes the questioning was with respect to the immediate concern of full power licensing of the Sequoyah station.

Now, in this case here, the staff has to be satisfied with confirmatory testing --

MR. ZUDANS: Okay.

MR. BUTLER: -- in the near term, that is, the next three months --

MR. ZUDANS: All right.

MR. BUTLER: -- that, in fact, these ignition systems will help.

The staff has to review the proposed design and approve its installation and use. It being a non-safety system at this time, that is, not required for full power licensing, TVA can, in fact, go ahead and install it, but they will not be allowed to operate it until they first get the NRC staff's endorsement for use.

MR. ZUDANS: So it, actually the installation, then, will have no effect on their full power license?

MR. BUTLER: That's correct. However, they propose to have it installed and available for use in the near term, that is, within the next couple of months, and it will then be a responsibility of the staff to make a finding that, yes, it does improve the safety margins and it is appropriate to use them.

MR. ZUDANS: But there is no urgency on staff's part to do that. They could just follow the long-term programs and

TVA.

MR. BUTLER: Well, we recognize that there is a desire to increase the safety margins, and we're prepared to expend the resources for this early review and approval of this interim system.

MR. ZUDANS: That sounds reasonable.

MR. BUTLER: Thank you.

As I indicated earlier, 1.3 -- we intend to have one of these contractors ignite various lean mixtures of hydrogen-air systems, and if possible including steam, and probably later on in the program include sprays, to determine whether -- when you ignite it, how is the ignition process and the combustion process.

Associated with that is part two, the analytical tasks. We intend to study various ignition strategies for different accident scenarios. One can well imagine that in some scenarios you wouldn't want to set off the ignition devices, and what we'd like to do is examine what kind of release rates under what conditions one might encounter and have -- work with TVA in the development of procedures for use of the ignition system.

The second part of the analytical task is to come up with some assessment of how effective the interim distributed ignition system might be, how much does it in fact improve the capability to take metal-water reactions.

JO-4

That is a near-term program over the next three months or so.

Now, a longer-term program, which also has two parts, will be the subject of this upcoming user's request. We expect it to be issued within the next -- within the current week. Its purpose will be to evaluate -- or to have the research people sponsor studies to evaluate systems for mitigation of degraded core/core melt accidents, confined to the issue of hydrogen control. The scope will be to develop information on mitigation systems for all LWR containment for use in the upcoming rule-making proceedings.

The short-term phase will extend over six to twelve months; and the longer-term phase, over twenty-four months.

Greater detail on the short-term phase is in this present slide, where the scope is confined to degraded core accidents, deferring the core melt aspect of it to the longer term. And it will cover the ice condenser and the Mark III containments, which have an intermediate capability for accommodating metal-water reaction.

MR. ZUDANS: Why don't you extend it to Mark I? You may not need inerting.

MR. BUTLER: We are going on the assumption that the Mark I's and II's will, in fact, be inerted. If it turns out that they are not required to inert, then we will have to put them in this same category of short-term review.

JO-5

MR. ZUDANS: Oh, this is short-term.

MR. BUTLER: This is within the next six to nine months -- wait a minute, six to twelve months.

The three major tasks involved here is to evaluate the hydrogen generation rates for a range of degraded core accidents. Our rough-cut studies as reported in the SECY paper 80-107 indicates that metal-water reactions might take between 15 and 20 minutes to go from zero to 100 percent metal-water reaction. What we'd like to do is examine different scenarios and see how fast can you generate the hydrogen and then after that how fast can you transfer it from the primary system to the containment system.

We need to determine the containment pressure response and temperature response, both for the case with and without the various mitigation devices.

And item three, we will try to assess the various promising hydrogen control systems.

For ice condensers, we are scheduling it over the next six to nine months. And for the Mark III, a little longer time scale -- 12 months.

And then my last slide here is a brief description of the long-term phase. It's intended to provide the staff with the information it requires for this upcoming rule-making proceeding on degraded core and core melt situations for all classes of LWR containments. It will include studies of the venting

IO-6

systems and hydrogen control systems.

These various hydrogen control systems, you've seen them before in prior presentations. And we are requesting that research schedule it for conclusion in about a two-year period.

That's all I have to say on the subject.

CHAIRMAN MARK: Does the staff have its independent idea as to how much hydrogen the containment of the size of the ice condenser could accommodate if you had distributed ignition burners in operation? Effective ones.

MR. BUTLER: We believe it is not possible to answer that with one simple number, because it depends very much on the rate at which the hydrogen evolves from the primary system into the containment. Clearly, if it is a slow evolution, then the containment heat removal systems play an important part. Whether you have a lot of ice in the ice condenser at the time, at the onset of hydrogen release and you can take advantage of the heat capacity of the ice, that's another factor that affects it. Containment sprays and whether it's available. These all -- all these factors, depending on their availability, have a strong impact on the ability to take metal-water reaction. So it's very difficult to pin down a number.

CHAIRMAN MARK: I can see that, because if you get the whole 25 percent hydrogen in and your igniter works, then you -- or you get 30 percent -- then you've had it.

MR. BUTLER: I think so, yes.

JO-7

CHAIRMAN MARK: But if you can burn it in three chunks of ten, then you say you will count upon, depending on what the time steps are, the cooling you could accomplish in between the different flares.

MR. BUTLER: Yes, that's true.

CHAIRMAN MARK: Very good.

Are there any other questions on the hydrogen study program proposals?

If not, I think there's only one remaining major item on the agenda, which has to do with the staff's thoughts at the moment on vented filtered containment studies. That's something which is put down for 60 minutes. Is that likely to require that long?

MR. STAHLE: Well, Mr. Meyers from the reactor systems branch is here. I think he allocated maybe on the order of that amount of time. But let me ask him if it can be reduced, or just his discussion --

MR. MEYER: I can tailor my presentation to a considerably shorter time span, if that's the subcommittee's desire.

CHAIRMAN MARK: Well, I wish you'd give a little thought to cutting it so that it was, if possible, not worse than half an hour or less. But I would suggest we take a ten-minute walkabout before that.

(A brief recess was taken.)

CHAIRMAN MARK: We will proceed with the meeting. Mr.

Meyer will report on NRC's looking at filtered vented systems.

MR. MEYER: My name is Jim Meyer, from NRR. The purpose of my presentation this afternoon is to bring the ACRS subcommittee up to date on the filtered vented containment system study for Zion and Indian Point.

I had intended to give a brief history of the study. I will only state one or two key points, to give you a feeling of completeness regarding the program.

The Zion and Indian Point action got under way late last year with a request to the utilities to perform certain interim actions on Zion and Indian Point that Matt Taylor referred to earlier today. In addition, there was a request to do studies of mitigation features, to see what features would be practical, what features could substantially reduce risk from these two plants -- or actually four plants.

The part of that program that I would like to review with you today is the NRC portion in the area of mitigation features. I would like to briefly review the status of the program, give you some feeling of the approach to the program that is the basis on which we are proceeding to determine whether in fact mitigation features will reduce risk at these two plants.

I'd also like to spend a few minutes giving you some examples of conceptual designs that have been proposed by Sandia Laboratories under contract to the office of research. They had a filtered vented containment system program under way and adjusted it to the specific Zion and Indian Point facilities to accommodate this action.

The purpose of the NRC's severe accident mitigation features study is to determine how immediate and practical technical fixes can be implemented on the Zion 1 and 2 and the Indian Point 2 and 3 units that assure a real and significant reduction in societal and individual risk due to severe accidents, including core melt.

The general approach is to pursue actively those design features that contribute favorably toward the mitigation of the consequences of a severe accident.

I will skip over the second Vu-graph, which presents the same statement in the form of a goal, and proceed with the third Vu-graph, which breaks down the program.

The three mitigation features under study are the filtered vented containment system that I will talk about further today, core retention devices, and hydrogen control methods. In addition, we are performing analyses in the area of steam explosion evaluation and in the area of accident risk calculations, so that we can pull the mitigation features study into an overall assessment of reduction in risk.

My main -- for the main part, my discussion will be, then, on the one aspect of the mitigation features, namely, the filtered vented containment system.

In approaching the study of filtered vented

O-10

containments for Zion and Indian Point, we can approach it by dividing it into three or four parts. We are interested in having a good handle on key input to determining requirements and criteria for the formulation of conceptual designs. In particular, as the Vu-graph shows, we're interested in what are the key loading terms for the containment. Pressure loading is the most important. But we're also interested in temperature loading, aerosol loading, and of course the radiological source term histories.

These containment loadings and source terms are generated by making use of the MARCH/CORRAL codes and also independent analyses. The MARCH/CORRAL codes, early versions, were used in the WASH-1400 analysis of the Surry plant.

We are also interested in variations in these loadings of the containment, based on a number of things, some of them indicated on the Vu-graph, for example, core retention devices, the presence of hydrogen control in the reactor system, in the containment system, and -- and other considerations.

Based on these loading histories, the Sandia Laboratories proceeded with a variety of conceptual designs and considered such factors as practical layouts, presence or lack of AC power, decontamination factors achievable, practical design flows, actuation levels, operator/automatic controls, venting to atmosphere versus special buildings, and also environmental requirements.

JO-11

F llowing the development of conceptual designs, we are then in a position to perform consequence analysis, very similar to the WASH-1400 type of analyses that Matt Taylor reported on earlier this afternoon.

Another important component of the filtered vented containment system program is a realistic assessment of the containment failure pressures and modes.

I'd now like to very briefly run through where we are in this program. I reported to the full committee on March 9th on the program. And this Vu-graph is intended to bring the subcommittee up to date on what has transpired.

We have held five technology exchange meetings, one of which was dedicated to addressing filtered vented containment systems. Those meetings have been concluded, and the meeting reports are being completed.

We intend to issue yet this month a sampling of what we feel are important functional requirements for the mitigation features, including filtered vented containment. Our important milestone is a staff report due late this fall which will give the staff recommendations regarding filtered vented containment systems as well as other mitigation features for Zion and Indian Point. The key question to be answered in that report is whether the staff feels that there can be substantial reduction in risk by incorporating these features into the plants.

The licensees have had a major parallel program, that was started December 5th, and have participated in the technology exchange meetings. They will be completing a report, I understand, to be issued late in September.

There are ongoing research and NRR programs that will be continuing throughout the summer to address key issues that surfaced as a result of the technology exchange meetings.

Another item that should be included on this Vu-graph is that Sandia and LASL (?) have published reports on mitigation features that have been made available to the ACRS. They are presently pre-publication copies of NUREG CR-1409 and -1410.

I'd like now to spend a few minutes giving you some indication of how we are proceeding with the filtered vented containment system designs, the conceptual designs, and, more importantly, how we're determining the risk reduction that's potentially available from such systems.

The first what one might refer to as a very general functional requirement is that the filtered vented system ends up having the net effect of reducing risk. In order to achieve this, the secondary requirement is to prevent containment failure. This is containment failure by overpressurization either due to hydrogen burn, steam generation, or a combination of the two, or by base melt through. Here we are not concerning ourselves with the other containment failure modes that you're familiar with from WASH-1400.

The next step is to define the functional requirements

JO-13

for systems which will prevent containment failure. An important ingredient in determining the functional requirements are to determine the sequences that dominate the risk, that challenge the containment, for which one can then establish functional requirements. And I will have a few Vu-graphs and make a few comments concerning the dominant sequences.

The next step then is to define and design systems that meet the requirements that have been established, requirements that came out of a understanding of the dominant accident sequences. Here I'd like to present a few specifics and give some examples.

The fifth step is to assess the consequence mitigation capabilities of the system. I would like to say a few words in terms of an example of a CRAC analysis that gives you some indication of the mitigation potential of a filtered vented containment system.

And finally, a very important item is to assess competing risks of the filtered vent containment system. Even if it offers considerable promise for reducing risk from high-risk contributors, it may also introduce new paths for release of radiation that may very well cancel out the beneficial effects for which it was designed. This is an important area of study.

So I'd like to take a few minutes just to very briefly run through the characteristics of the accident sequences that the NRC staff feels dominate risk. And I should qualify this that -- that these are ones that dominate risk for the large dry containments.

The first item is that they are initiated by small breaks and transients. This is in contrast to the double-ended pipe rupture that, of course, we're all familiar with.

The second is that human error contributes significantly to the risk from these sequences.

A third characteristic: that the core melts.

A fourth: that large amounts of hydrogen are generated, of course, as a consequence of the former.

Containment safety features are unavailable.

And the containment fails above ground via overpressure.

The actual sequences that the Sandia Laboratory program used in order to develop conceptual designs are indicated on this Vu-graph. The numerical indication is from the notations in WASH-1400. The description, I think, suffices for the various accident sequences.

The first one is the TMLB' -- loss of offsite and onsite AC power for an indefinite period. So it's a transient event with loss of all offsite power and onsite power.

The AB burn is a large LOCA, loss of electric power; and because the conditions according to the analyses indicate the potential of hydrogen burning, hydrogen burning is included in this scenario. A better word than -- you should substitute

JO-15

for hydrogen explosion rapid hydrogen burning in this Vu-graph.

The third and fourth are both small breaks, one with a hydrogen burn and one without a hydrogen burn, the S2D being a failure of the ECCS system, and the S2G being a containment heat removal failure.

The -- the last two are of less importance, one being the TMLB'', which is the same as the first transient but the AC power comes on after six hours instead of 16 hours. This introduces some complications to the containment loading as well as to the design of the filtered vent.

The last item is a -- a transient that would not have failed the containment if the systems that -- the engineered systems worked properly. It's a double-ended pipe rupture but with the assumption that the vent installed. It's a case where the original containment would not have failed but there would have been a release through the filtered vent had the filtered vent been installed with certain criteria for venting.

END TAPE 11

Tape12

To give you a little flavor for what these transients look like, I have shown here three of the ones that present enveloping challenges to the containment.

The solid-line transient is for the -- the solid-line pressure history is for the TMLB' accident. And I think you can see from your copies the distinction between the lower-dotted-line S2D burn and the higher-dotted-line AB burn.

Two important points that should be noted from the

JO-16

pressure histories. The first was mentioned earlier this afternoon. The characteristic in all three of these of a pressure surge or what's been referred to as a pressure spike. And the other characteristic that if you assume the failure pressure range indicated on the Vu-graph, then even without the pressure spikes for the two, for these two scenarios, that you enter into the failure pressure range.

I might add that the AB burn is considered a scenario which challenges the containment but which is a -- it is also an accident sequence which is considered very unlikely relative to the other accident sequences.

I'd like to give you a little flavor for what these conceptual designs actually look like. The approach that Sandia used was to think in terms of four filtered vent options and two modes associated with those four options.

The first mode is a venting-to-atmosphere mode, shown at the top without AC power. And the second is an option of venting to the atmosphere as well as the potential for controlled recirculation back into the existing containment building.

These two modes, as indicated on the next two Vu-graphs that you have, the vent relief mode and the recirculation mode, I think are pretty well self-explanatory. I'll be glad to present those Vu-graphs and comment on them if you wish. But the vent relief mode is basically, the purpose is to prevent containment rupture due to overpressurization by venting to the atmosphere. And the second mode is -- provides alternate containment heat removal, depressurization, and atmospheric clean-up.

MR. ZUDANS: What size of a vent lines would you require to effectively reduce the pressure in the case of hydrogen burn?

MR. MEYER: The -- I could -- I can give you -- it's a difficult question to answer from --

MR. ZUDANS: Well, why don't you just remember what we talked half an hour ago: we said that they could not be practically made to reduce the pressure.

MR. MEYER: Well, I'd have -- those --

MR. ZUDANS: They are just simply too big.

MR. MEYER: Those particular comments were in reference to a different kind of containment. I can tell you, I can give you a feel for the type of openings that would be required in a large dry containment to accommodate certain transients.

If it is required, for example, to remove 150,000 cfm in order to keep the containment below dangerous pressures, a three-foot-diameter penetration has been calculated to be sufficient. This would accommodate a large portion of the transients that I showed earlier, assuming that the rate of rise in the several spikes are conservative. If, in fact, the rate of rise that's been calculated in the Sandia report are, in fact, real, then we need a much larger opening in order to

JO-18

accommodate --

MR. ZUDANS: Well, that's what the understanding was. From your scale, the other scale in minutes it's very difficult to how fast it rises.

MR. MEYER: It's very difficult to --

MR. ZUDANS: And we heard just before that about five seconds to do the combustion.

MR. MEYER: The -- for the -- the most rapid rise considered for steam pressure type of spikes is in the range of 15 seconds. And that is considered by some to be very conservative. I don't have a number for you on the hydrogen burn, but it's in that range, I believe.

The -- what I'd like to do now is to present a Vu-graph that presents the most complicated or the most complete of the four options I referred to. And as we remove various options, you can get an idea -- remove various features, you can get an idea of the various options.

This is also in your packet.

The first major component in the filtered vented system, according to the Sandia study, is a suppression pool, in many ways not unlike the suppression pool in a boiling water reactor. The suppression pool has estimated decontamination factors of 50 for particles, 50 for inorganic iodine, and 2 for organic iodine. The -- if one assumes 150,000 cfm flow rate and a release into the filtered vented system at a pressure of about

JO-19

75 psi, then the size of the system is approximately 150,000 cubic feet of water and 150,000 feet of free volume. The Sandia report estimates that if this is a Seismic Category 1 system, it will cost about nine million dollars.

If we go right from here to the stack, this constitutes basically option one in the Sandia report. The same with the presence of AC power if we take this and by-pass these two components, we basically have option one.

If we had the second option of a sand filter indicated here, we pick up, as you might expect, considerable advantage in terms of decontamination factor improvement. The decontamination factor for the sand filter proposed is 500 for particles. The estimated cost for this system would be -- again, Seismic Category 1 -- would be five million dollars.

The final major component -- or I should add that then if we by-pass this final component this constitutes option two. If we include further zeolite and impregnated charcoal with HEPA filters present, we then are considering the third option. Here we pick up decontamination factors of 200 for inorganic iodine and 100 for organic iodine.

The final option is to also include a large bit of activated charcoal, in order to accommodate the, some of the xenon activity.

To put these various options into perspective, I have included here one of the several consequence calculations that were performed. You note that these are conditional probabilities normalized to the assumption that the accident occurs. And here I've shown -- or here the Vu-graph shows latent cancer fatalities.

With no venting, that is, with the assumption of containment failure, there is, of course, as you expect, a large number of latent cancer fatalities.

For the option number one, which just included the suppression pool, there is a -- about an order of magnitude improvement in the reduction of -- an order of magnitude improvement in the reduction of consequences.

And going to option two almost doubles that, or gives you two orders of magnitude.

It's important to note that fine-tuning by going to options three and four do not appear, at least, from these CRAC analyses, to buy very much in terms of risk reduction.

The thinking now is that, depending on the impact of the competing risks that I will talk about in a minute, the viable options are either option one or option two.

I present this Vu-graph which was also -- a number of these Vu-graphs were presented at the technology meeting I referred to earlier. I think this Vu-graph very effectively portrays the problem in designing a filtered vent and containment system. Here we have a whole number of different kinds of vent strategies that are -- that are described in some detail in the report I referred to earlier.

They go from a rating of effectiveness for severe accidents of low to high. So what we'd like to have to reduce the consequences from severe accident is systems in the lower portion of the Vu-graph. However, the rub comes when considering interference with less severe accidents, system interactions, a number of things that turn out to give you high contributions -- or potentially high contributions to risks from these more complicated systems. Therefore, one must draw some kind of a balance between the two in making a judgment and liquidation regarding mitigating systems.

I should add here that one of the activities this summer is to pursue this question in some detail, both through the research program at Sandia and through the NRR program at Brookhaven National Laboratory.

But these are some of the system interactions that are of concern. And they're basically of concern because when you -- for many of the conceptual designs, when you vent the containment, you reduce the pressure in the containment considerably and, therefore, might have interactions resulting in cavitation of recirculation pumps due to sump flashing, severe vacuum caused by delayed sprays or coolers, degradation of reflood caused by premature venting, and other interfaces.

There are both administrative procedure and design change solutions to these problems, which will be looked at.

JO-22

And finally -- and again I -- we can probably skip over this, but what this Vu-graph portrays is essentially the last Vu-graph but in more detail, indicating some possible problems, some possible interactions of the filtered vented system with other systems in the Zion and Indian Point containments.

Unless there are specific questions on this Vu-graph, I will just summarize and state that for various components, both the control, the performance, and possibly the function of these systems are jeopardized by certain conceptual designs.

That concludes my remarks. If there are any questions I will be glad to answer them.

CHAIRMAN MARK: I take it the study has the objective of firming up the estimates that you have given, freezing on an option between zero and four and then taking serious attention to the stuff on this last graph, are you really making money and if so how much by giving similar consideration to the negative terms?

MR. MEYER: Well, I -- I presented the four options as examples of what's being considered by Sandia.

CHAIRMAN MARK: Yes.

MR. MEYER: There are a group of other options --

CHAIRMAN MARK: I didn't mean that that was limiting, of course. On that graph you wouldn't perhaps have thought it reasonable to go to four. But you might have to go to two, for instance, possibly, and then you would look at this last page and see whether you have bought trouble with that.

MR. MEYER: Exactly. That -- that's exactly what we are attempting to do during the summer, to firm that up so that we can not only make a judgment regarding the reduction in risks from the severe accidents but we can have a, we can fold into that the competing risks that are potentially there, that may, in fact, for several cases, like I said before, negate the benefits that were originally intended.

CHAIRMAN MARK: In one of your early Vu-graphs, you included basemat melt through. Is that of concern because you're really concerned with the risk resulting from fission products getting under the basemat or the gas from the basemat raising the pressure in containment?

MR. MEYER: Well, there -- as far as the filtered vented containment program is concerned, the -- the concern is the latter: core melt interactions with concrete not only have the potential for generating pressures in and of themselves but also copious amounts of hydrogen, other combustibles, and aerosols that may, in fact, clog some of the filtering systems.

CHAIRMAN MARK: But you're really paying attention to what would interact between the containment and the atmosphere rather than to the sequel of having gone through the concrete and wondering what happens down there?

MR. MEYER: Well, there is another component to the

JO-24

program, the core retention device component, which is addressing the questions of whether, in addition to having a core retention device ›ep containment pressures and aerosols and hydrogen generation down, tc either terminate or delay penetration. So that aspect is being considered.

CHAIRMAN MARK: I would have thought the risk, however, that you're talking of there would be a very much smaller term than the ones that come to site upstairs.

MR. MEYER: In terms of the atmospheric pathway, that's correct. In fact, the WASH-1400, the more probable containment failure path was basemat penetration, but it contributed so little to the risk that it wasn't of major consequence.

However, we are taking a new look at the basemat penetration problem for these two sites, not only in terms of atmospheric release but in terms of the question of liquid pathway, to -- to make sure that -- that there are no implications there for -- for problems.

CHAIRMAN MARK: Fine.

Any questions?

Thank you, Mr. Meyer.

I think that completes the agenda. The meeting will shift now into executive session and I don't think we need a record. We can terminate the record at this point.

END TAPE 12

(Whereupon, at 5:30 p.m., the committee went into closed session.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBC. ON SEQUOYAH NUCLEAR PLANT

Date of Proceeding: July 9, 1980

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.

Joan Gill

Official Reporter (Typed)

Joan Gill

Official Reporter (Signature)

NUCLEAR REGULATORY COMMISSION

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Suzanne Babineau

Official Reporter (Typed)

Suzanne Babineau

Official Reporter (Signature)

Tape 4
Timmons

REACTOR VESSEL NOZZLE UNDERCLAD CRACKING

BACKGROUND

WESTINGHOUSE FRENCH LICENSEE DETECTED CRACKING:

- IN BASE MATERIAL OF REACTOR VESSEL NOZZLES
- IN BROAD AREA OF NOZZLE BORE - MORE PREVALENT IN THICKER SECTION
- CONFINED TO HAZ OF SECOND LAYER OF CLADDING
- ORIENTED PERPENDICULAR TO CLADDING DIRECTION
- 1.0 INCH IN LENGTH, 0.28 INCH IN DEPTH
- BY DESTRUCTIVE AND NON-DESTRUCTIVE (UT) EXAMINATIONS

CRACKING BELIEVED TO BE:

- HYDROGEN-INDUCED
- RESULT OF WELDING PROCESS/HEAT TREATMENT USED IN CLADDING

Tape 4

CHRONOLOGY OF EVENTS RELATED TO REACTOR VESSEL NOZZLE UNDERCLAD CRACKING

Early October 1979

- NRC and Northern States Power Company (NSPCo) advised of cracking found by French licensee and that Prairie Island Units 1 and 2 (operating plants) have French-manufactured reactor vessels

October 26, 1979

- W/NSPCo Meeting

November 26, 1979

- NRC/W/NSPCo Meeting
- W presented status of ongoing efforts:
  - survey of vessel manufacturers
  - examination of French-manufactured nozzles/boat samples
  - Prairie Island fracture mechanics analyses
  - development of UT technique
- NSPCo committed to do 70° UT ISI of nozzles:
  - Unit 1 - July 1980 outage
  - Unit 2 - February 1981 outage
- NRC saw no immediate concern related to continued operation of Prairie Island Units and concluded that W proceeding in an appropriate manner

December 13, 1979

- W transmitted letter to NRC:
  - documenting information presented at November 26 meeting
  - indicating that Rotterdam-manufactured vessels (Sequoyah Unit 1, Watts Bar Units 1 and 2, McGuire Unit 2, Catawba Unit 1) under investigation and that cladding processes/heat treatment used by CE, B&W, CB&I should preclude cracking

Late December 1979

- All customers advised of survey results/W efforts
- Decision made to inspect Watts Bar Unit 2

Early January 1980

- Watts Bar Unit 2 nozzles inspected

January 31, 1980

- W transmitted letter to NRC
  - documenting results of Watts Bar Unit 2 inspection - no underclad cracking

Early February 1980

- Decision made to inspect one Sequoyah Unit 1 nozzle
- Sequoyah Unit 1 nozzle inspected - reheat cracking found

February 22, 1980

- NRC/W/TVA Meeting
- Results of Watts Bar Unit 2 nozzles and Sequoyah Unit 1 nozzle inspections presented
- NRC required inspection of other Sequoyah Unit 1 nozzles
- NRC stated that all Rotterdam-manufactured nozzles should be inspected
- NRC concern related to satisfying ASME Code Section XI acceptance criteria

Late February 1980

- Other Sequoyah Unit 1 nozzles inspected - underclad cracking found
- Acceptability of all indications in terms of Section XI criteria demonstrated
- NRC granted Sequoyah Unit 1 5% Operating License (February 28, 1980)

Mid-March 1980

- NRC requested detailed information about cladding process/heat treatment used in fabrication of North Anna Unit 2 nozzles in order to perform an independent evaluation (NOTE: North Anna Unit 2 vessel manufactured by Rotterdam, nozzles clad by Sulzer.)
- Virginia Electric and Power Company committed to inspect North Anna Unit 2
- NRC inquired about condition of Salem Unit 2 nozzles (NOTE: Salem Unit 2 vessel manufactured by CE.)
- Public Service Electric & Gas Company committed to inspect Salem Unit 2

INCOMPLETE (FULL-POWER) TMI ISSUES ON SEQUOYAH UNIT NO. 1

1. Shift Tech Advisor - 1/81
2. Immed. Upgrade of SRO & RO Qual. - 8/80
3. Admin. of Training Program for Licensing Exams - 8/80
4. Rev. Scope & Critiera for Normal Licensing Exams - 8/80
5. Rev. Scope & Criteria for Simul. Exams
6. Proc. For Veirification of Correct Perf. of Op. Activities
7. Control Room Design Review
8. Reactor Coolant Systems Vents - 1/81
9. Post-Accident Sampling - 1/81
10. Training for Mitigating Core Damage
11. Analysis of Hydrogen Control
12. Degraded Core - Rulemaking
13. Relief and Safety Valve Test Req. - 6/81
14. AFW Reliability Evaluation
15. AFW Initiation and Indication - 1/81
16. Containment Dedicated Penetratic - 1/81
17. Containment Isolation Dependability
18. Add. Acc. Monitoring Instrumentation - 1/81
19. Inadequate Core Cooling Instruments - 1/81
20. Final Recom. of B&O Task Force
21. Upgrade Emergency Prepardness
22. Upgrade Emergency Support Facilities - 1/81
24. Communications
25. Impl. of NRC and FEMA Respon.
26. Offsite Dose Measurements
27. In-Plant Radiation Monitoring - 1/81
28. Control Room Habitability
29. Power - Ascension Test

SEQUOYAH
FLOOD SCENARIOS

RAINFALL(PMF)-3 day storm over 21,400 sq.mi.
watershed (16.8in.) preceeded by
3 day storm (6.7 in.) 3 days earlier

SEISMIC-Failure of 4 dams coincident with
crest of 1/2 PMF

FLOOD PROTECTION PLAN

STAGE I

- Call additional personnel
- Controlled shutdown and cooldown
- Make load adjustments to DG system
- Boration to 5% shutdown margin (Xenon free)
- Move supplies and distribute equipment

FLOOD PROTECTION PLAN

STAGE II

- HPFP would replace AFWS
- ERCW would replace Component Cooling Water
- Liquid radwaste tanks filled to prevent flotation
- "Non-essential" loads are de-energized
- Batteries below DBF are disconnected
- Drains are sealed in DG and ERCW buildings

WARNING PLAN

PROVIDES AT LEAST 27 HOURS NOTICE TO SEQUOYAH

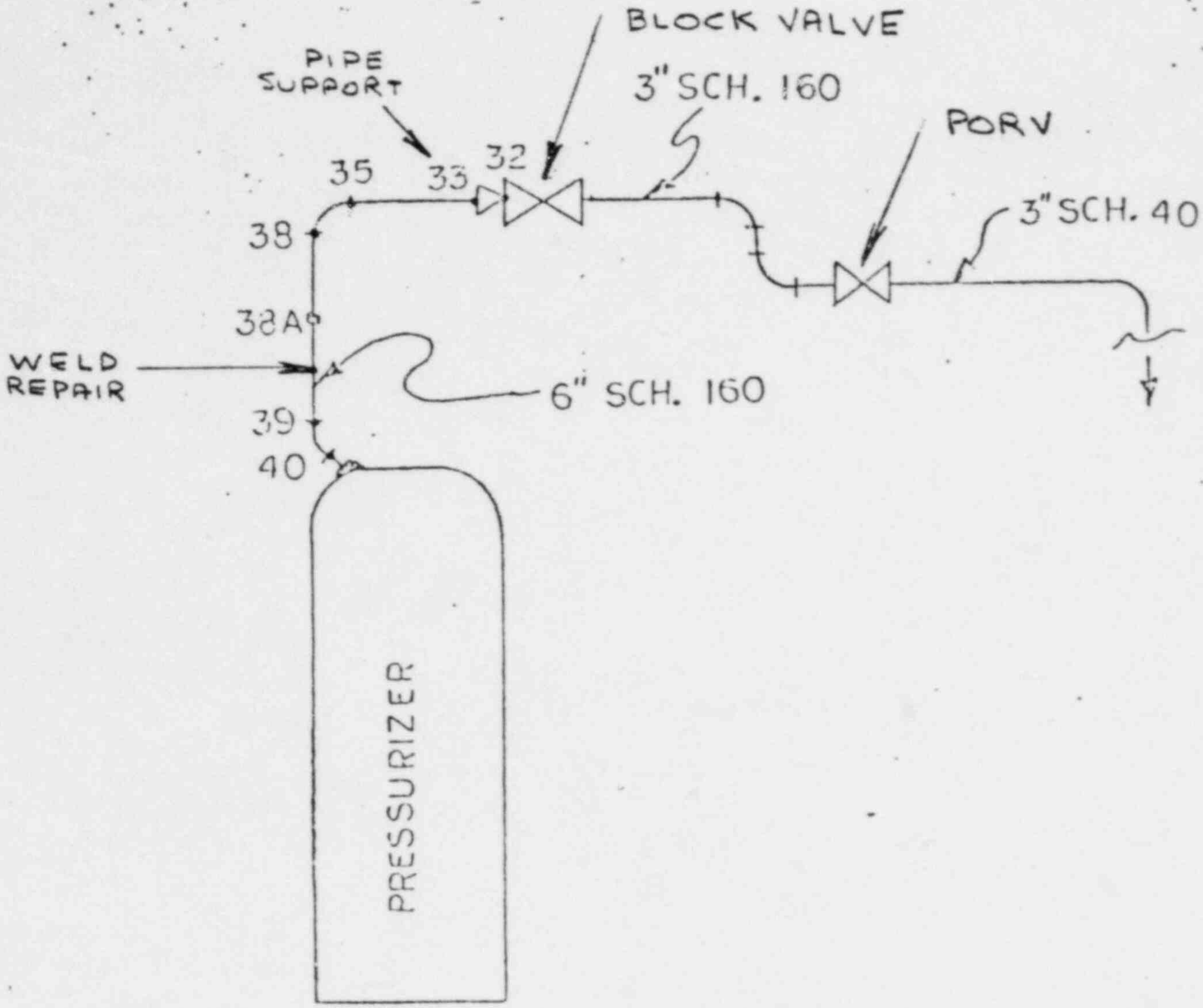
- 10 hours to implement Stage I
- 14 hours to implement Stage II
- 3 hours contingency

COMPLETE (FULL POWER) TMI ISSUES ON SEQUOYAH UNIT NO. 1

1. REACTOR INSPECTOR AT OPERATING REACTORS
2. SHORT TERM ACC. ANALYSIS AND PROC. REVISION
3. NSSS VENDOR REVIEW OF PROC.
4. PILOT MONITORING OF SELECTED EMERG. PROC. FOR NTOL APP.
5. LOW POWER TESTING TRAINING
6. PLANT SHIELDING
7. EMERG. POWER FOR PRESSURIZER HEATERS
8. PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT

INCOMPLETE NON-TMI ISSUES ON SEQUOYAH UNIT NO. 1

1. Seismic Audit per ACRS Letter
2. Position Required Regarding Foudation Monitoring on Settlement
3. Position Required on Containment SUMP Debris
4. ECCS Evaluation Model Concerning Fuel Clad Swelling
5. Position Required Regarding Process Control Program
6. Equip. Qualifications Comply With the Guidelines of NUREG-0588
7. PAD 3-3 Performance Code - Complete Evaluation Regarding Restriction In the Use of this Code
8. ATWS - Review and Approve Operating Procedures
9. Compliance of IE Bulletin 79-27, Loss of Non-Class IE Instrumentation & Control Room System During Operation
10. Diesel Generator Reliability - Compliance with R.G. 1.108 and NUREG/CR-0660
11. Topical Reports WCAP-9226, 9230 and 9236 Related to Main Steam & Feedline Break Accidents
12. Q-List Complete Review of "Q-List" Requirements
13. Compliance of OIE Bulletin 80-06 Related to By-Pass. Override, Reset Circuits



PRESSURIZER RELIEF PIPING SKETCH

Tape 5

TVA RELIABILITY STUDIES

A) SYSTEMS INTERACTION METHODOLOGY APPLICATIONS PROGRAM

B) REACTOR SAFETY STUDY METHODOLOGY APPLICATIONS PROGRAM

C) AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

D) PLANT FULL SCALE SAFETY AND AVAILABILITY ANALYSIS

SYSTEMS INTERACTION METHODOLOGY
APPLICATIONS PROGRAM

SUMMARY

- AN OBJECTIVE WAS TO DEVELOP A METHODOLOGY INDEPENDENT OF THE STANDARD REVIEW PLAN (SRP) FOR IDENTIFYING AND EVALUATING SYSTEMS INTERACTIONS IN LIGHT WATER REACTOR COMMERCIAL POWER PLANTS
- WATTS BAR NUCLEAR PLANT (WBNP) WAS CHOSEN AS THE EXEMPLARY FACILITY FOR DEMONSTRATING THE METHODOLOGY
- ALTHOUGH IT WAS NOT THE PURPOSE OF THIS STUDY TO JUDGE WBNP, IT WAS CONCLUDED THAT THE FACILITY IS GENERALLY WELL PROTECTED AGAINST INTERACTIONS CONSIDERED WITHIN THE SCOPE OF THIS STUDY

SYSTEMS INTERACTION METHODOLOGY
APPLICATIONS PROGRAM

OVERVIEW

OBJECTIVE

- DEMONSTRATION OF METHODOLOGY

METHOD

- IDENTIFICATION OF COMMONALITIES EXISTING AT WBNP THROUGH EXAMINATION OF FAULT TREES
- DETERMINATION OF POTENTIALLY INTERACTIVE CUT SETS WITH 3 OR LESS INDEPENDENT FAILURES
- REVIEW AND ASSESSMENT OF POTENTIAL INTERACTIONS

LIMITATIONS

- RCPB MITIGATING SYSTEMS WERE NOT MODELED
- FAULT TREES WERE DEVELOPED FOR ANSI N18.2 CONDITION I AND II OCCURRENCES ONLY
- FUNCTIONS RELATING TO THE CONSEQUENCES OF RELEASE OF RADIOACTIVITY WERE NOT MODELED
- FIRE, EARTHQUAKE, HURRICANES, TORNADOES, FLOOD, SABOTAGE EXCLUDED

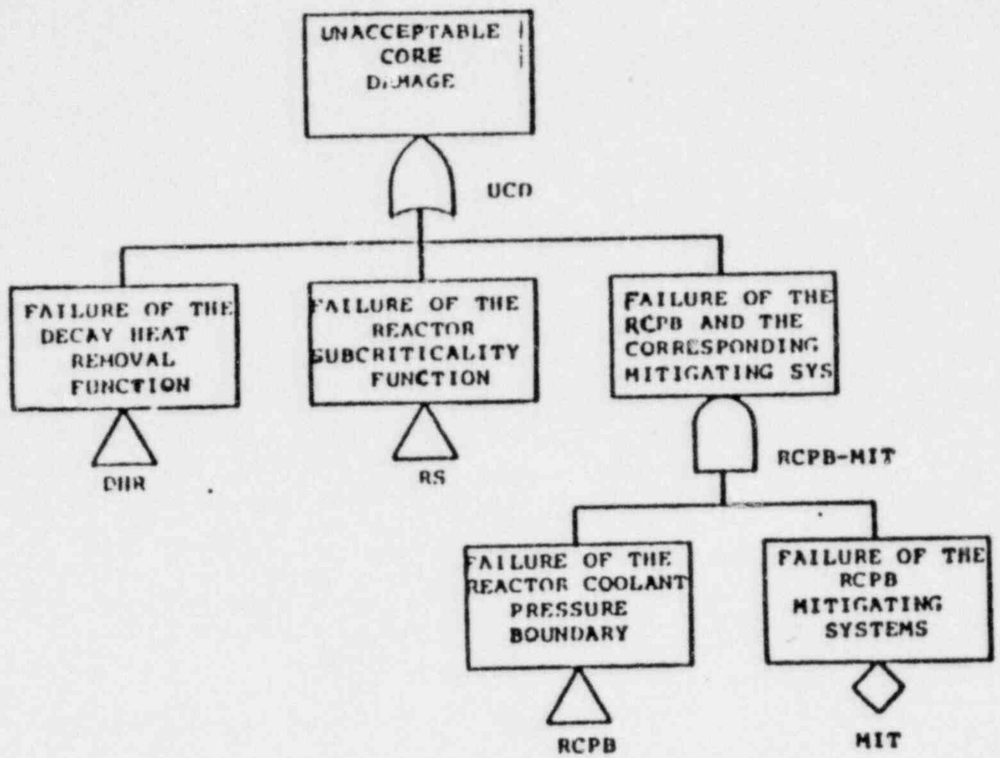


FIGURE 2.1. TOP OF THE FAULT TREE.

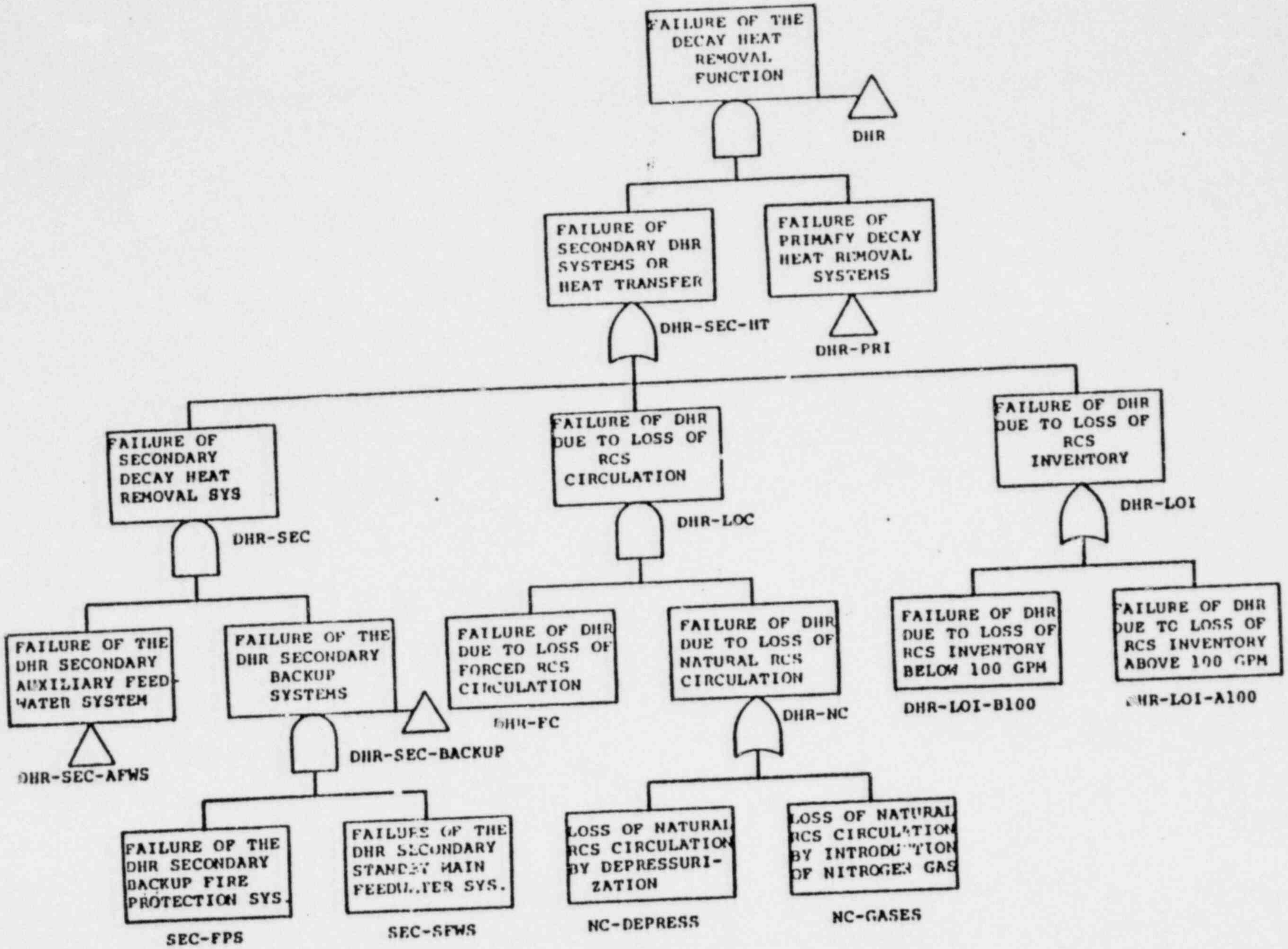


FIGURE 2.5. DECAY HEAT REMOVAL

SPECIFIC ANALYSIS

DATA OBTAINED ON ALL COMPONENTS WHICH APPEAR IN CUT SETS

LINKING CHARACTERISTICS

- AC POWER - TRAINS A AND B
- DC POWER - TRAINS A AND B
- ACTUATION - INPUTS AND OUTPUTS TO AUTOMATIC CONTROL CIRCUITS
- LUBRICATION - INTERNAL AND EXTERNAL
- COOLING
- HYDRAULIC
- COMPRESSED AIR
- LOCATION - ROOMS, PIPE CHASES, GENERAL AREAS

REACTOR SAFETY STUDY
METHODOLOGY APPLICATIONS PROGRAM

OVERVIEW

OBJECTIVE

- DETERMINATION OF DOMINANT ACCIDENT SEQUENCES

METHOD

- SYSTEM EVENT TREES CONSTRUCTED FOR WASH-1400 INITIATING EVENTS
- SIMPLIFIED FAULT TREES DEVELOPED FOR MITIGATING SYSTEMS

RESULTS

- ICE CONDENSER PLANTS HAVE DIFFERENT DOMINANT ACCIDENT SEQUENCES
- RISK IS SIMILAR TO LARGER DRY CONTAINMENT PLANTS

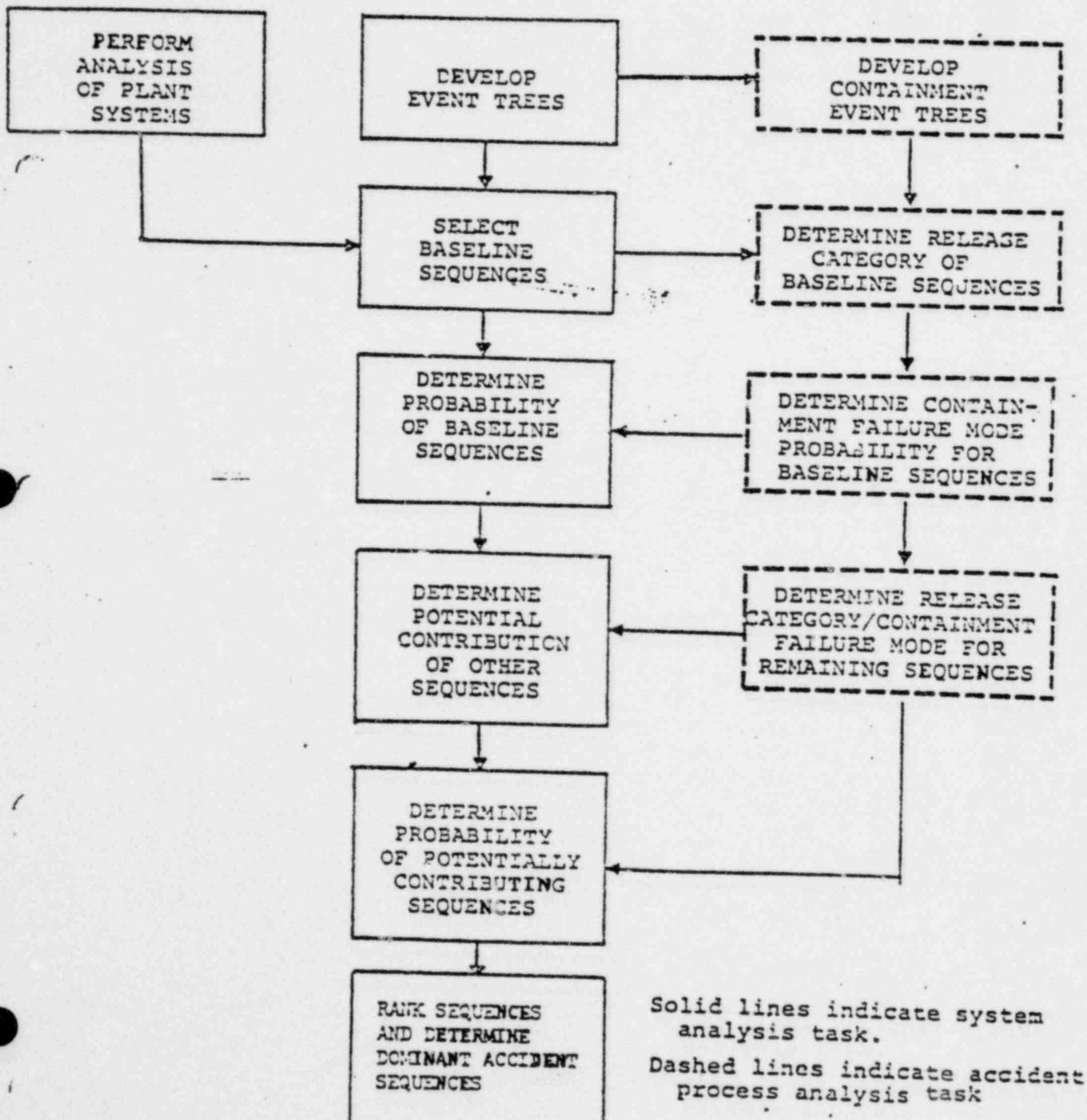


Figure 3-1. RSS Methodology Applications Program Systems Analysis Task

SEQUOYAH AUXILIARY FEEDWATER SYSTEM
RELIABILITY EVALUATION

SUMMARY

Kaman Sciences Corporation was contracted by the Tennessee Valley Authority to conduct a reliability evaluation of the Sequoyah Unit #1 Nuclear Power Plant Auxiliary Feedwater System (AFS). Kaman employed the GO computerized event tree methodology to perform the analyses.

Results indicate that the probability of successfully starting the auxiliary feedwater system upon demand and providing adequate water flow and pressure to at least two out of four steam generators is 0.99999 where the initiating event is both feedwater pumps tripped. In event of loss of offsite power (blackout) with diesel generators and battery back-up available the AFS start-up success probability is 0.99997. Other excursions were also evaluated.

The analysis revealed that there are no first order faults in the Sequoyah AFS for the initiating event both feedwater pumps tripped. A total of 116 second order faults were identified for this case. The largest contribution of unavailability resulting from a pair of faults is 10^{-7}. Most second order fault sets contribute to start-up unavailability on the order of 10^{-10}.

SEQUOYAH AUXILIARY FEEDWATER SYSTEM
RELIABILITY EVALUATION

INTRODUCTION

In November 1979 the Tennessee Valley Authority (TVA) contracted with Kaman Sciences Corporation (KSC) to conduct an evaluation of the reliability and availability of the auxiliary feedwater system of Sequoyah Unit 1 nuclear power plant located at Daisy, Tennessee.

This analysis had multiple objectives. The initial objective was to determine the probability of successful start-up of the AFWS under the following conditions:

1. Both main feedwater pumps tripped and all power sources available.
2. Loss of offsite power with diesel generators and battery backup available.
3. Loss of all ac power, batteries only available.
4. Break on steam generator #1 water inlet line causing low steam generator #1 level.
5. Loss of Train A 120 volt ac inverter electrical power and trip of both main feedwater pumps to cause AFS initiation.

Additional objectives included: an assessment of the long term reliability of the AFS, assuming successful start-up; and the identification of all first and second order fault sets which could fail the system in the start-up mode for the initiating event both main feedwater pumps tripped.

PROCEDURE

TVA provided KSC with complete engineering drawings and operational descriptions of the Sequoyah AFS including requisite support systems such as compressed air, electrical power, etc. KSC used these data to construct a comprehensive model of the AFS employing the GO methodology which

KSC has developed. GO is a collection of computer codes and modeling instructions which have been developed during the past 15 years to treat complex reliability and availability problems in a rigorous yet economical manner. In recent years its continuing development has been sponsored by the Electric Power Research Institute.

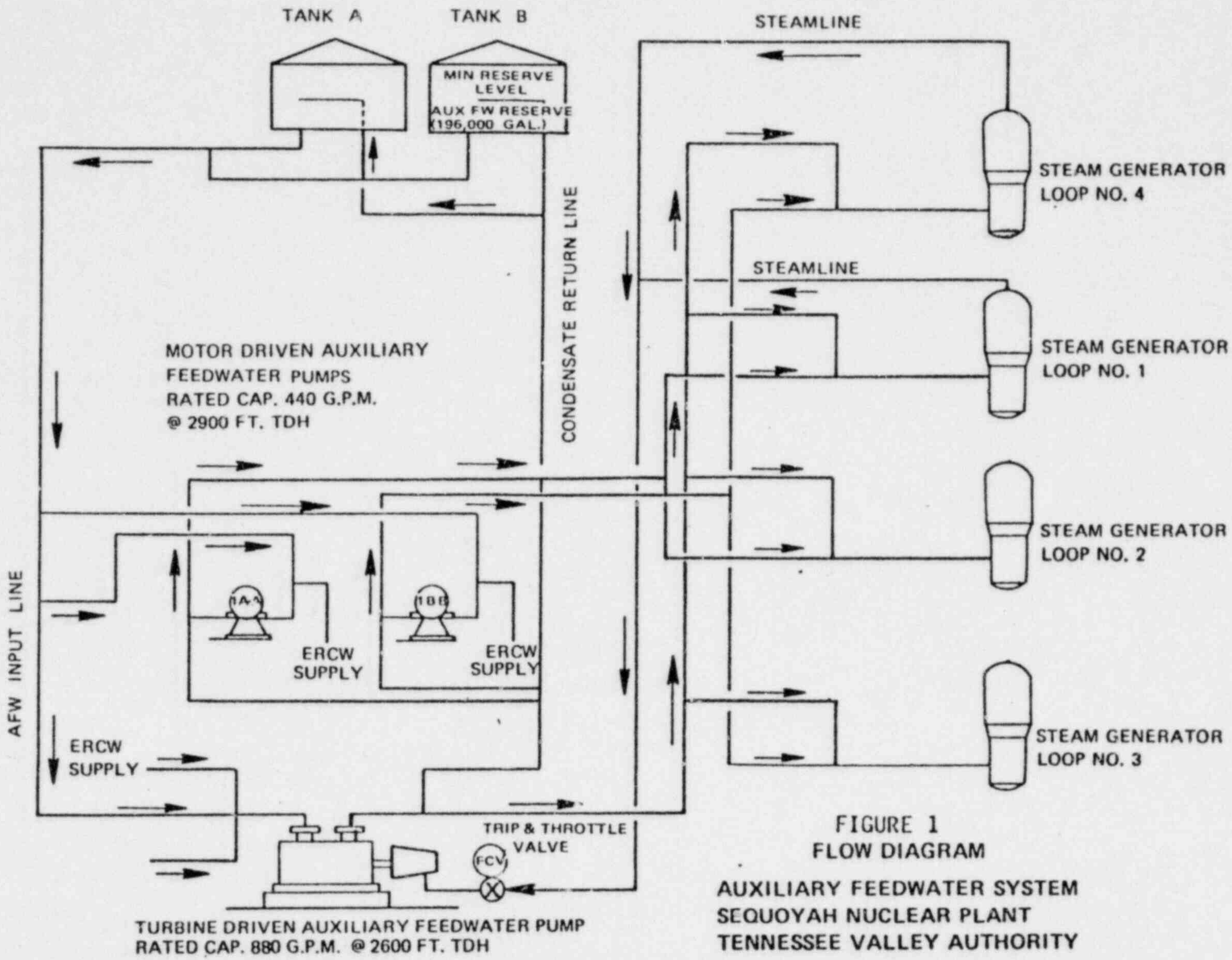


FIGURE 1
FLOW DIAGRAM

AUXILIARY FEEDWATER SYSTEM
SEQUOYAH NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY

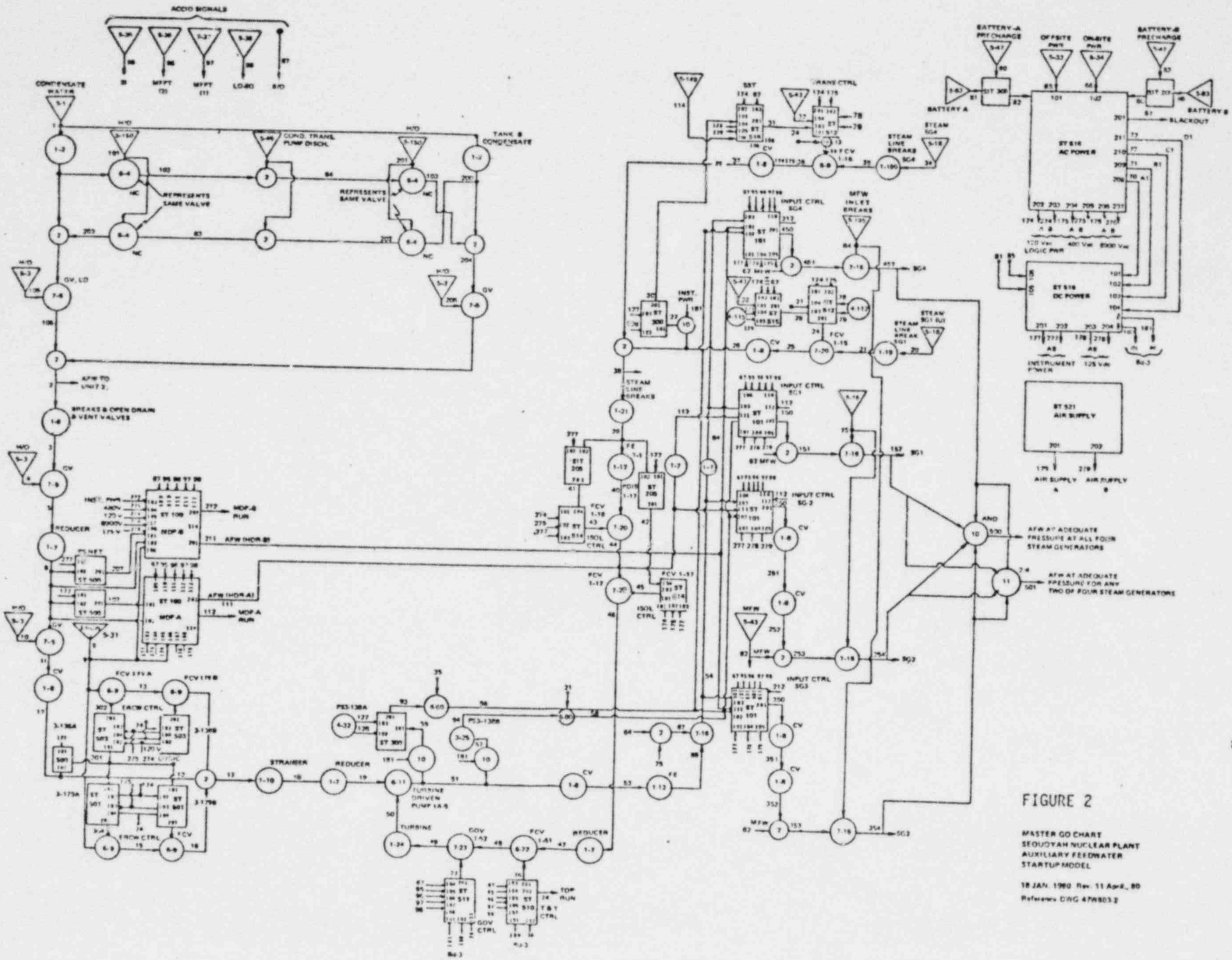


FIGURE 2

MASTER GO CHART
SEQUOYAH NUCLEAR PLANT
AUXILIARY FEEDWATER
STARTUP MODEL

18 JAN. 1980 Rev. 11 April, 80
Reference DWG 47W803-2

SEQUOYAH NUCLEAR PLANT
FULL SCALE SAFETY AND AVAILABILITY ANALYSIS

OBJECTIVE:	To develop two plant models, one to assess plant safety and one to evaluate plant availability		
METHOD	GO methodology developed by Kaman Sciences Corporation with funding from EPRI		
MANPOWER	KSC	=	80 man-months
	TVA	=	30 man-months
SCHEDULE	Phase 1		July 1, 1980 - Dec. 31, 1980
	Phase 2		Jan. 1, 1981 - Dec. 31, 1981
SCOPE	Phase 1		Simplified plant model Detailed plant models of selected systems (Electrical Power, Central Air, Reactor Protection, Safety Injection, Main Steam, Main Feedwater) Preliminary safety and availability assessments
	Phase 2		Expansion of simplified model Data collection Final safety and availability assessments Incorporation of operator, test, and maintenance actions Determination of critical components Investigation of abnormal scenarios

Tape 6 +7

REACTOR ACCIDENT RISKS

SOME INSIGHTS ON RISKS OF AN ICE CONDENSER
DESIGN VS. WASH-1400 PWR AND BWR DESIGNS

JULY 1980 PRESENTATION TO ACRS

BY

M. TAYLOR
PROBABILISTIC ANALYSIS STAFF
OFFICE OF NUCLEAR REGULATORY RESEARCH

BACKGROUND

- o WASH-1400 (RSS) (LATE 1975)
- o RSS - METHODOLOGY APPLICATIONS PROGRAM (KICKOFF, LATE 1976)
  - LONGER TERM - 4 LWR'S
  - CODE DEVELOPMENT - MARCH, CORRAL
- o LEWIS REPORT RECOMMENDATIONS (E.G., SMOOTHING, UNCERTAINTIES)
- o FRG RISK WORK
- o TMI-2 INVESTIGATION RESULTS
- o IREP
- o I.P. INTERIM OPERATIONS (SECY-80-283)
  - REBASELINING OF WASH-1400 PWR AND BWR DESIGN
  - COMPARISON OF DESIGNS, SITING, PROTECTIVE MEASURE OPTIONS

ESTIMATED PROBABILITY OF SEVERE CORE DAMAGE

REACTOR NAME	TYPE	PROBABILITY* OF SEVERE CORE DAMAGE PER REACTOR-YEAR
SURRY	3-loop PWR	$6x10^{-5}$
PEACH BOTTOM	BWR (Mark I)	$3x10^{-5}$
SEQUOYAH	4-loop PWR (Ice Condenser)	$4x10^{-5}$
OCONEE	2-loop PWR	$2x10^{-4}$
CALVERT CLIFFS	2-loop PWR	$2x10^{-4}$
CRYSTAL RIVER-3	2-loop PWR	$3x10^{-4}$
BIBLIS	4-loop PWR	$4x10^{-5}$
INDIAN POINT	4-loop PWR	$1x10^{-5}$

* Reflects median values

- SECY-80-283 -

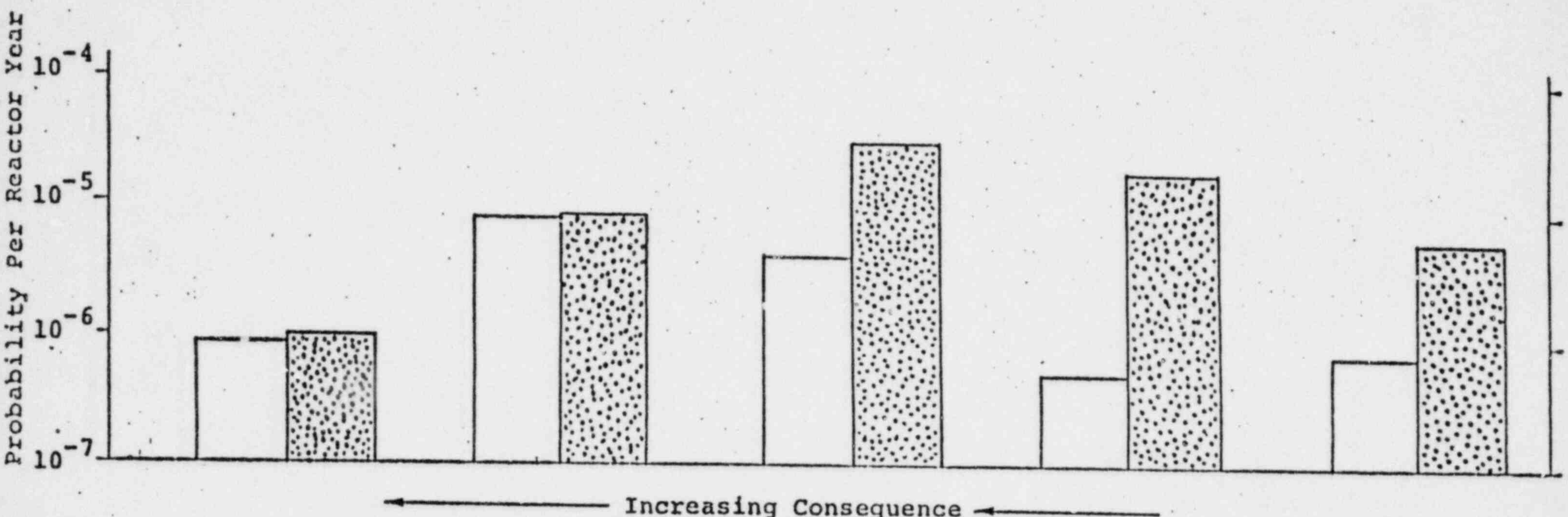
DOMINANT ACCIDENT SEQUENCES

Accident Scenario	Sequence Code From WASH-1400	Offsite Consequences Expected
LOCA and failure of ECCS in injection mode	AD S_1D S_2D	Low to modest
LOCA and failure of ECCS in recirculation mode	AH S_1H S_2H	
Transient scram and loss of feedwater or serious failure and no feed and bleed on primary side (X)	TMLX TMKX	
LOCA and loss of containment heat removal with subsequent interactions with ECCS	AG S_1G S_2G	Intermediate
LOCA and failure of ECCS and containment ESFs in recirculation phase due to common cause	AHF S_1HF S_2HF	High
LOCA and coupled damage to ECCS and potential bypass of containment	Event V	
Transient involving loss of all AC power (or possibly DC) and failure of auxiliary feedwater	TMLB'	

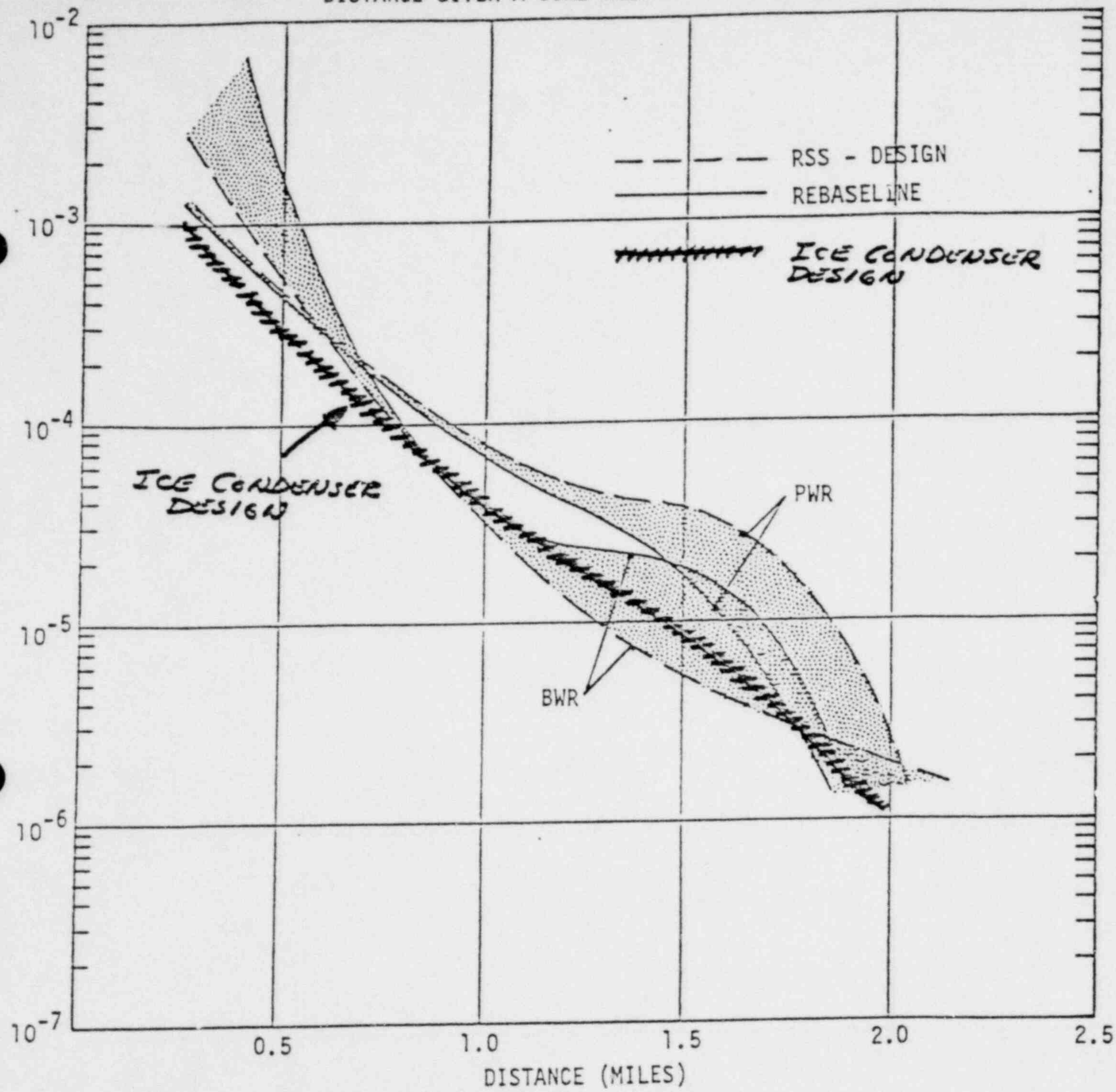
— Secy-80-283 —

Comparison of Dominant Accident Sequences: RSS PWR and Ice Condenser PWR

Category 1		Category 2		Category 3		Category 4		Category 5	
RSS	IC	RSS	IC	RSS	IC	RSS	IC	RSS	IC
None	$S_1H-\alpha$	V	$S_2HF-\gamma$	$S_2C-\delta$	$S_2H-\gamma$	None	$S_1H-\gamma$	None	$S_1D-\gamma$
Dominant	1×10^{-7}	4×10^{-6}	5×10^{-6}	2×10^{-6}	2×10^{-5}	Dominant	1×10^{-5}	Dominant	4×10^{-6}
		TMLB'-δ	V		$S_1HF-\gamma,\delta$		$S_2D-\gamma$		
		2×10^{-6}	9×10^{-7}		3×10^{-6}		6×10^{-6}		
		TMLB'-γ			TML-γ				
		7×10^{-7}			3×10^{-6}				



DISTANCE GIVEN A CORE MELT*



ASSUMPTIONS: *CORE MELT PROBABILITY ASSUMED TO BE 10^{-4}/REACTOR YEAR

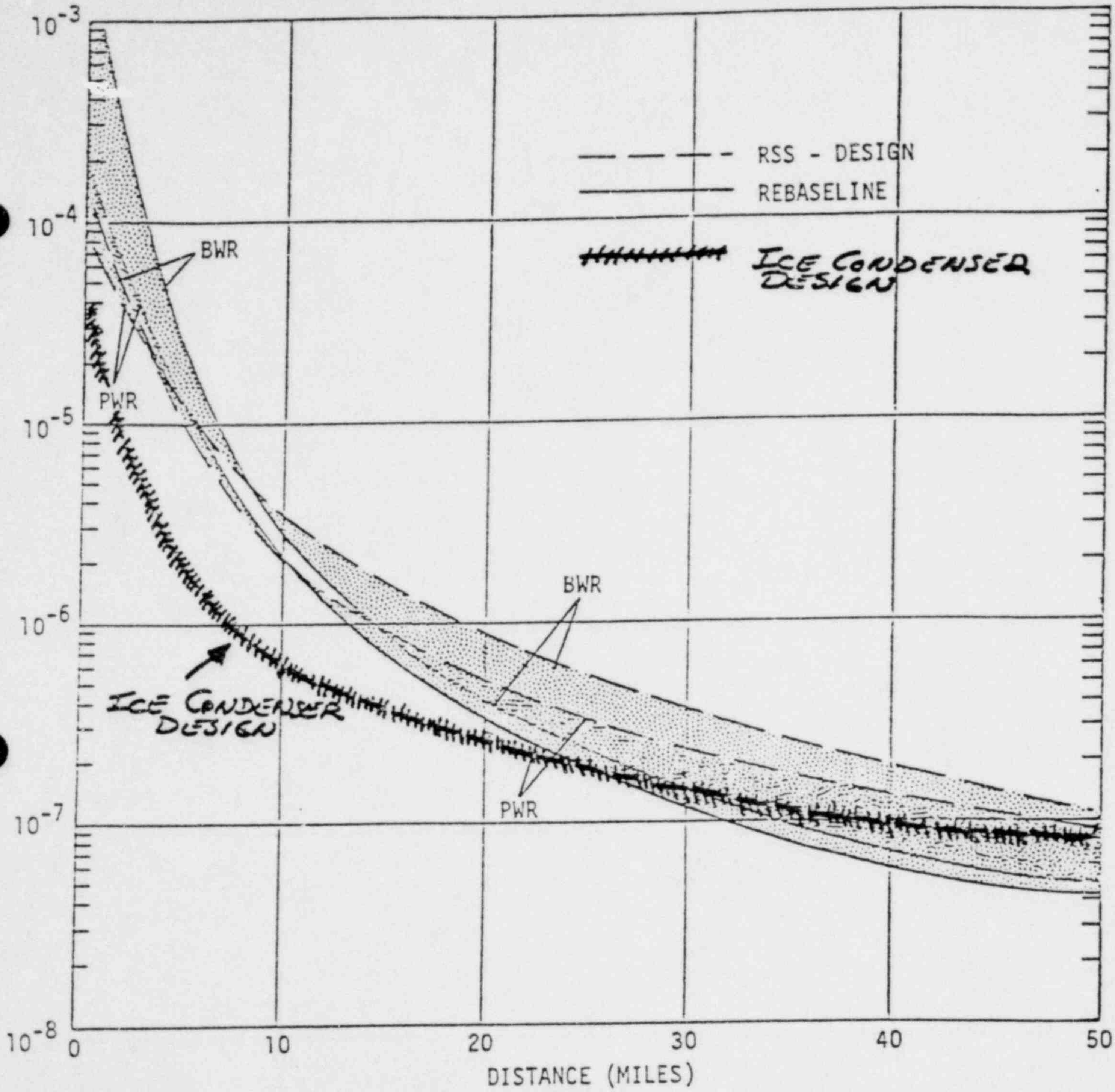
RSS-DESIGN
1. ALL RSS CORE MELT ACCIDENT RELEASE CATEGORIES
2. ALL RSS ASSUMPTIONS (E.G., SMOOTHING)

REBASELINE
1. SMOOTHING ELIMINATED
2. EXPLICIT ACCIDENT SEQUENCES
3. NEGLIGIBLE PROBABILITY OF VESSEL STEAM EXPLOSION

EXPECTED CONSEQUENCES FROM 91 WEATHER SEQUENCES WITH 3200 MWT POWER LEVEL

ENTIRE CLOUD EXPOSURE + 24 HOUR GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY

RISK OF LATENT CANCER FATALITY TO AN INDIVIDUAL VERSUS DISTANCE GIVEN A CORE MELT*



ASSUMPTIONS: *CORE MELT PROBABILITY ASSUMED TO BE 10^{-4}/REACTOR YEAR

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1. ALL RSS CORE MELT ACCIDENT RELEASE CATEGORIES
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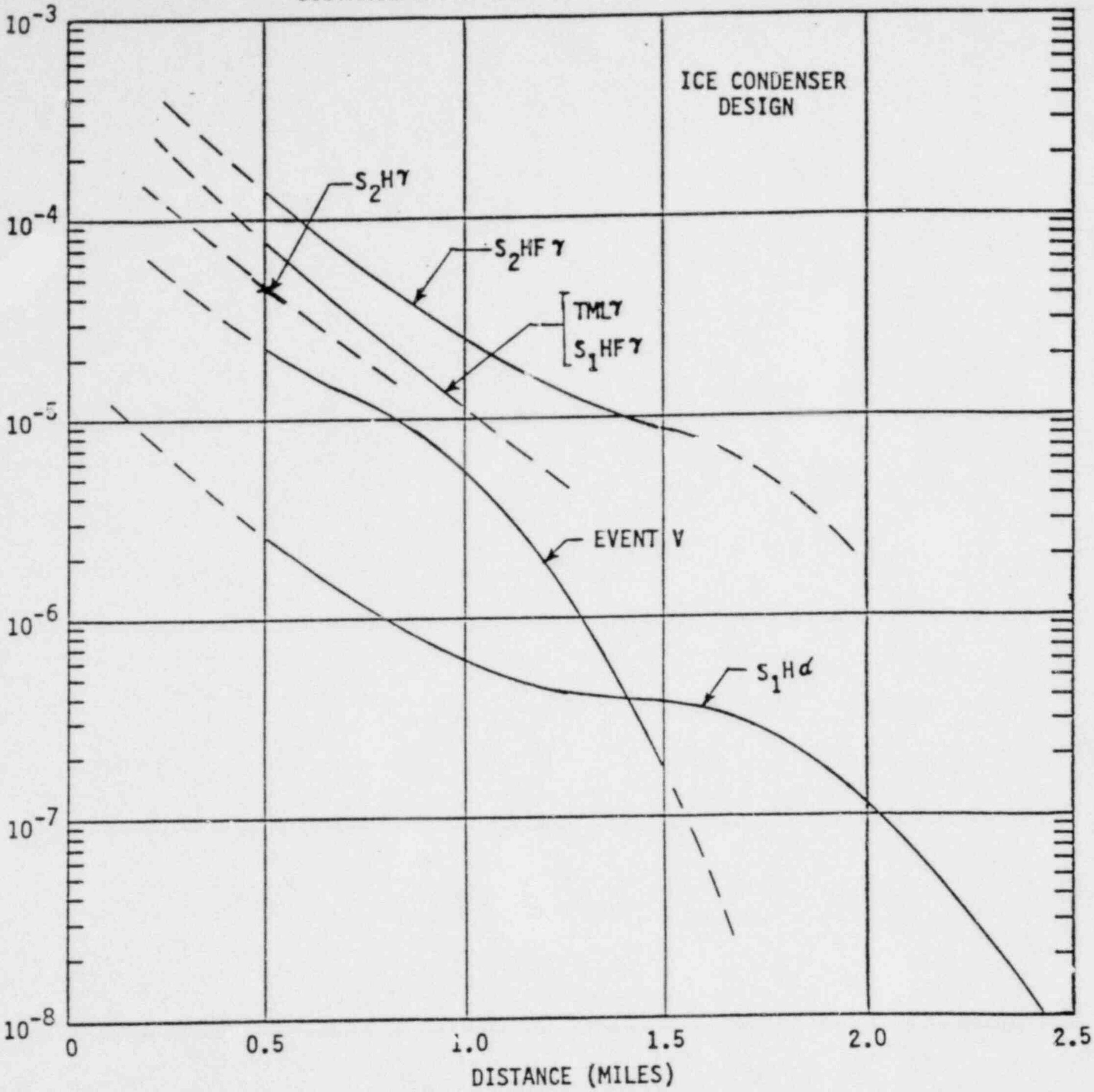
ENTIRE CLOUD EXPOSURE + 24 HOUR GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY

RISK OF EARLY FATALITY TO AN INDIVIDUAL VERSUS DISTANCE GIVEN VARIOUS CORE MELT SEQUENCES



ASSUMPTIONS:
- OVERALL CORE MELT PROBABILITY ASSUMED TO BE 10^{-4}/ REACTOR YEAR
- SMOOTHING ELIMINATED
- EXPLICIT ACCIDENT SEQUENCES
- EXPECTED CONSEQUENCES FROM 91 WEATEHER SEQUENCES WITH 3200 MWT POWER LEVEL
- ENTIRE CLOUD EXPOSURE + 24 HOUR GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY

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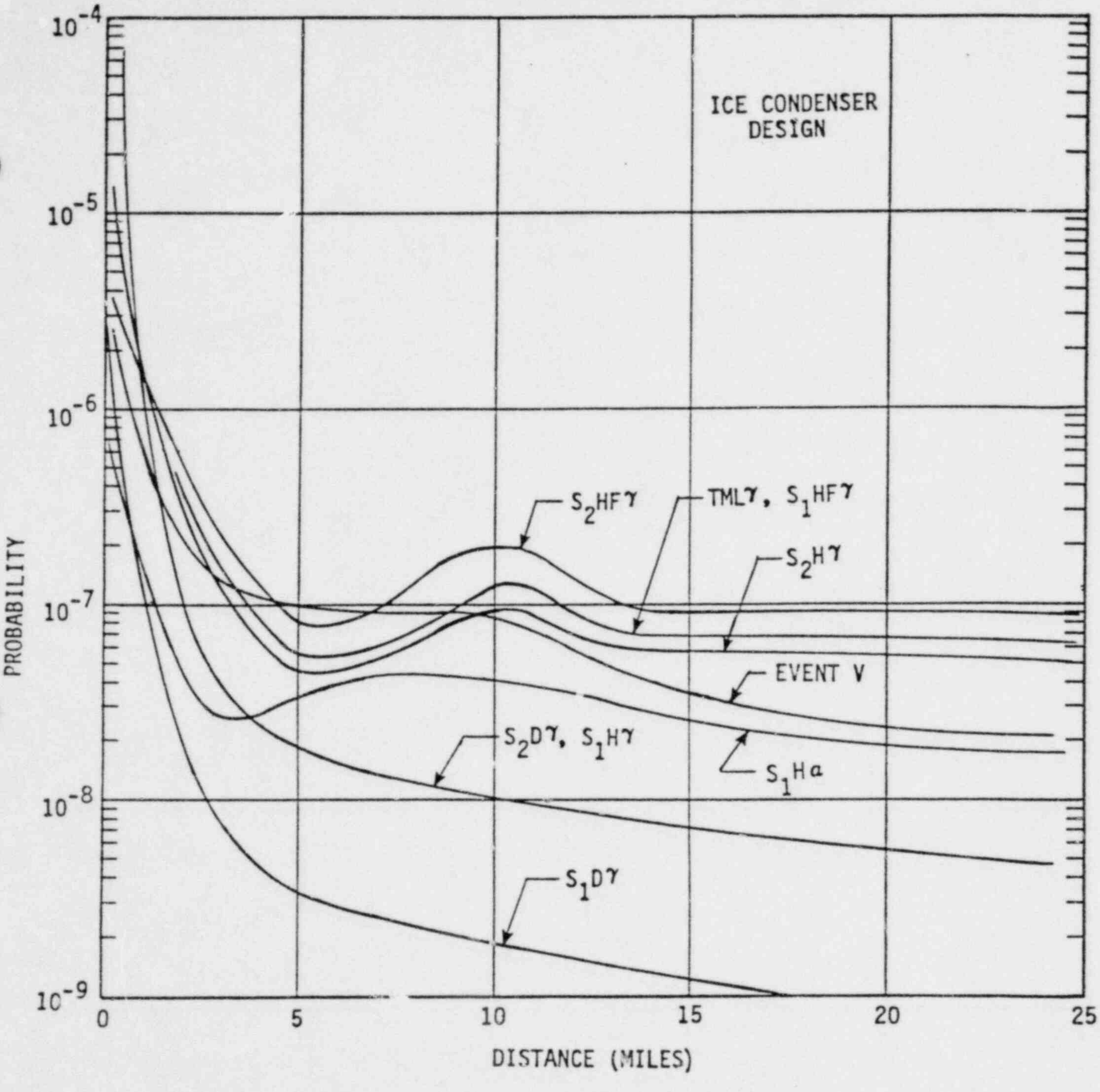
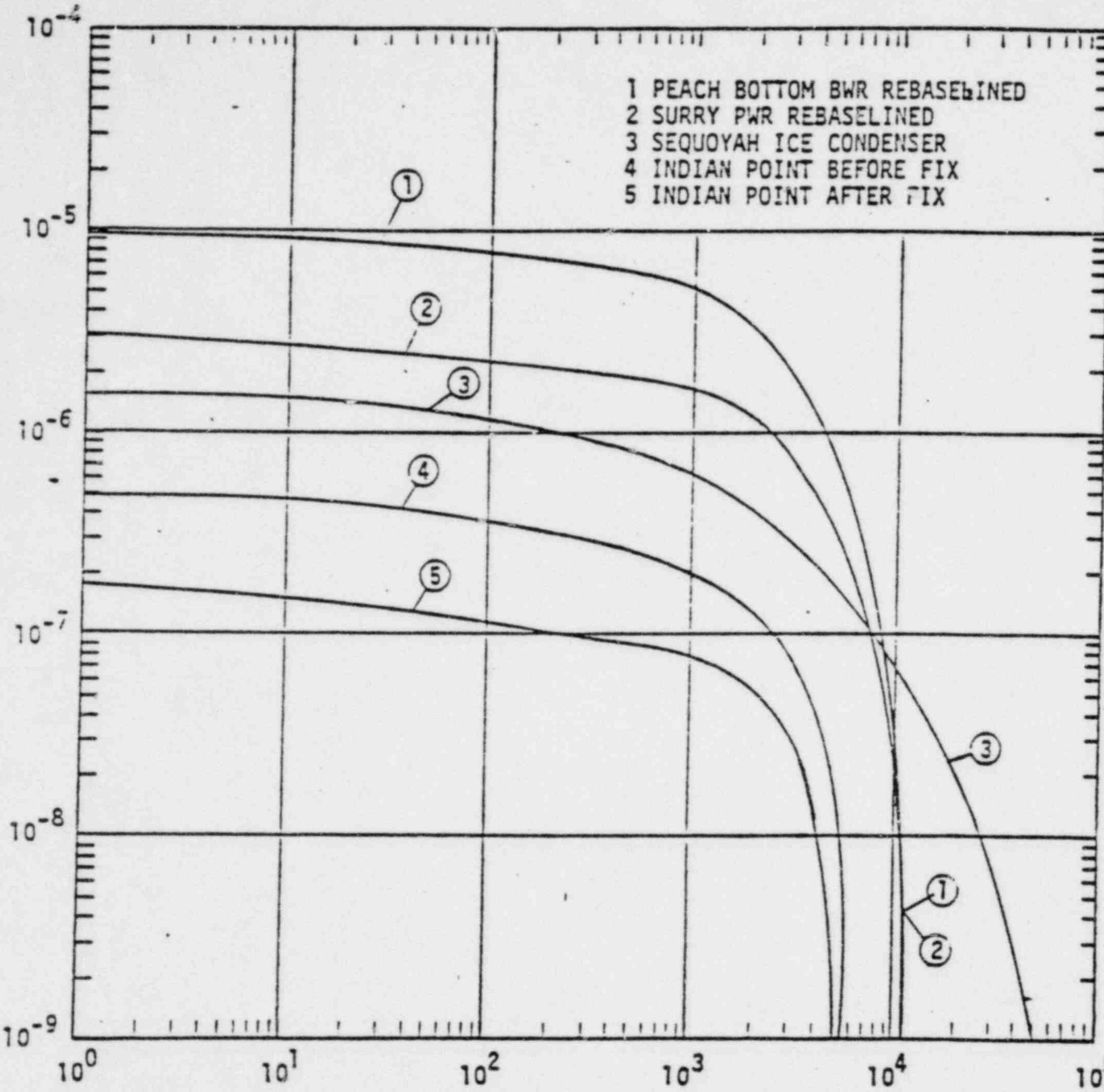


ASSUMPTIONS:
- OVERALL CORE MELT PROBABILITY ASSUMED TO BE 10^{-4}/ REACTOR YEAR
- SMOOTHING ELIMINATED
- EXPLICIT ACCIDENT SEQUENCES
- EXPECTED CONSEQUENCES FROM 91 WEATEHER SEQUENCES WITH 3200 MWT POWER LEVEL
- ENTIRE CLOUD EXPOSURE + 24 HOUR GROUND EXPOSURE SHIELDING BASED ON NORMAL ACTIVITY

VARIATION OF DESIGN AND OPERATION

- o INDIAN POINT SITE
- o SAME PUBLIC PROTECTION MEASURES
- o DIFFERENT REACTORS AT 3025 MWT
  - SURRY
  - PEACH BOTTOM
  - SEQUOYAH
  - INDIAN POINT BEFORE
  - INDIAN POINT AFTER

EARLY FATALITY RISK FOR DIFFERENT DESIGNS

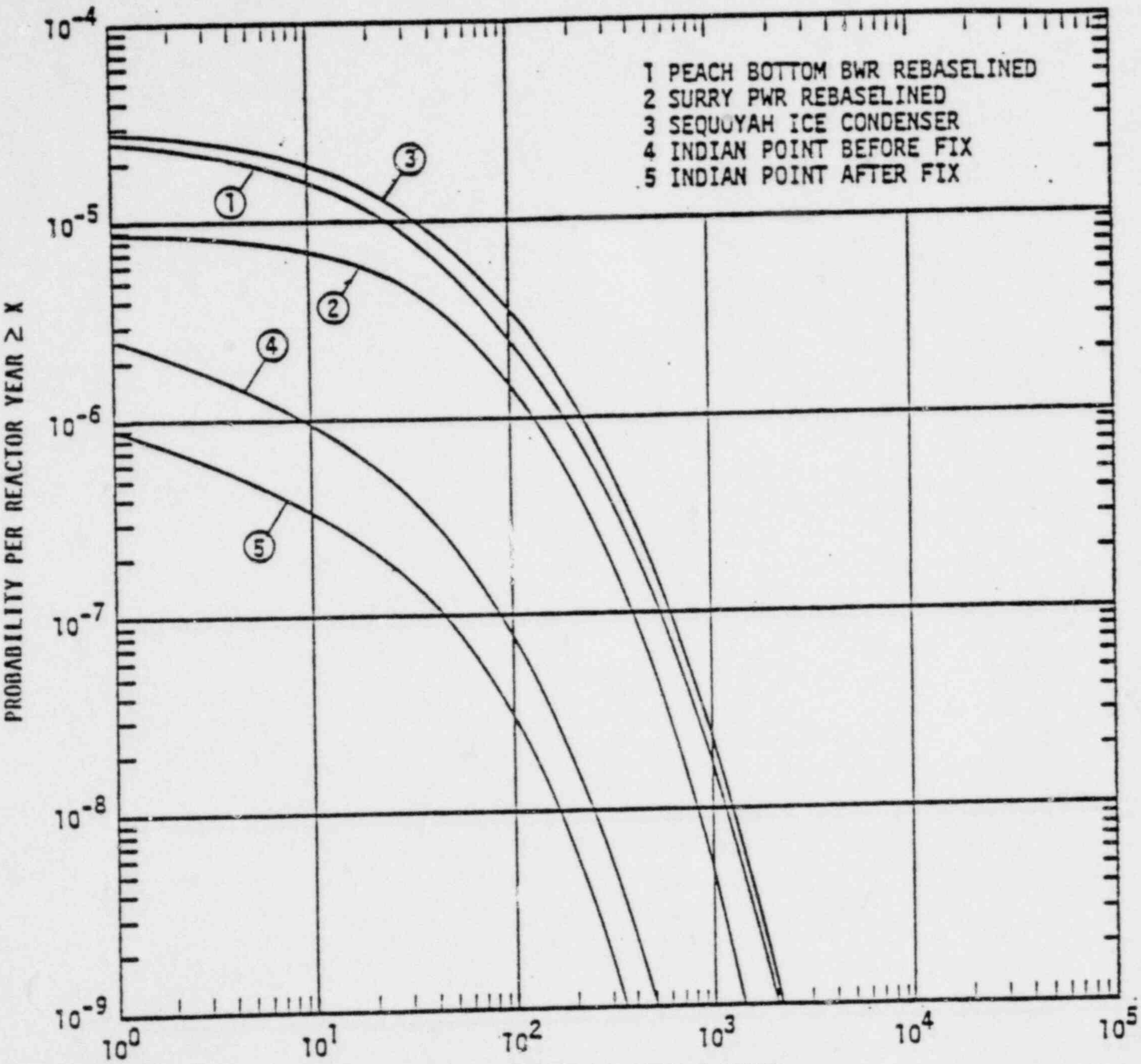


NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

ASSUMPTIONS: 1) INDIAN POINT SITE
METEOROLOGY - 91 WEATHER SEQUENCES
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
UNIT 3 POWER LEVEL (3025 MWT)

2) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
SHIELDING BASED ON NORMAL ACTIVITY

LATENT CANCER RISK FOR DIFFERENT DESIGNS



*TOTAL LATENT CANCERS WOULD BE 30 TIMES HIGHER

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE.

ASSUMPTIONS: 1) INDIAN POINT SITE
METEOROLOGY - 91 WEATHER SEQUENCES
WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION
UNIT 3 POWER LEVEL (3025 MWT)
2) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
SHIELDING BASED ON NORMAL ACTIVITY

SOME CONCLUDING OBSERVATIONS AND INSIGHTS

- CORE MELTDOWN ACCIDENTS DOMINATE LARGE CONSEQUENCE ACCIDENTS AND OVERALL RISKS
- ICE CONDENSER PLANT DESIGN HAS SOMEWHAT DIFFERENT RISK DOMINANT SEQUENCES RELATIVE TO WASH-1400 PWR AND BWR DESIGNS
- ICE CONDENSER CONTAINMENT FAILURE MODE MOST AFFECTING RISK IS OVERPRESSURE FAILURE
- OVERALL ACCIDENT RISKS FOR AN ICE CONDENSER DESIGN APPEAR COMPARABLE TO THOSE PREDICTED FOR WASH-1400 PWR AND BWR DESIGNS
- LARGE UNCERTAINTIES EXIST IN THESE RISK-BASED ANALYSES BUT SHOULD NOT NEGATE ABOVE INSIGHTS

Tape 8
slide 1 149

SUMMARY

- HYDROGEN STUDIED ABOUT NINE MONTHS

- SEQUOYAH CAN WITHSTAND SUBSTANTIAL AMOUNTS OF HYDROGEN ABOVE DESIGN BASIS

- SIGNIFICANT MODIFICATIONS HAVE BEEN OR ARE BEING INCLUDED TO REDUCE POTENTIAL FOR DEGRADING EVENTS

- LIMITED RISK ASSESSMENT SHOWS SEQUOYAH COMPARABLE TO THE WASH 1400 STUDY REFERENCE PLANT

- PROPOSED CONCEPTS FOR RESOLUTION OF HYDROGEN ISSUE EVALUATED

- INTERIM DISTRIBUTED IGNITION SYSTEM CHOSEN FOR IMPLEMENTATION AT SEQUOYAH. DEVELOPMENT WORK ON CONTROLLED IGNITION IS PROCEEDING FOR FINAL IMPLEMENTATION AT SEQUOYAH. HALON SUPPRESSION IS ALSO BEING STUDIED.

SLIDE 2

CAPABILITY OF THE SEQUOYAH CONTAINMENT

– MINIMUM CONTAINMENT PRESSURE CAPABILITY

YIELD – 33 PSIG
ULTIMATE – 42.5 PSIG

– VOLUME – 1.2×10^6 FT^3

– CONTAINMENT CAPABILITY TO WITHSTAND HYDROGEN COMBUSTION

ASSUMPTIO[illegible]

– BURN IS INSTANTANEOUS AND COMPLETE

– BURN IS ADIABATIC

– NO RADIATIVE TRANSFER

RESULT:

– SEQUOYAH CAN WITHSTAND A HYDROGEN BURN EQUIVALENT TO APPROXIMATELY 25 PERCENT METAL-WATER REACTION (USING ULTIMATE STRENGTH OF MATERIALS)

CONCEPTS STUDIED FOR MITIGATION, CONTROL, OR PREVENTION OF CONSEQUENCES FROM HYDROGEN

– MITIGATE THE CONSEQUENCES OF HYDROGEN BURNING

VENTED CONTAINMENT:

1. FILTERED
2. ADDITIONAL
3. COUPLED

– CONTROL COMBUSTION

CONTROLLED IGNITION SOURCES

– PREVENT COMBUSTION

1. INERT CONTAINMENT WITH NITROGEN
2. SUPPRESS COMBUSTION WITH HALON

CONCEPTS – ASSESSMENT

– VENTED CONTAINMENT

FILTERED

1. NOT EFFECTIVE FOR RAPID PRESSURE TRASIENTS

2. ESTIMATED DOSE IN LOW POPULATION ZONE IS IN EXCESS OF 900 REM

3. SOME ESSENTIAL FEATURES NOT DEMONSTRATED

4. POTENTIAL FOR UNNECESSARY BYPASS OF CONTAINMENT

5. HIGH INITIAL COST, MODERATE O/M COST

ADDITIONAL CONTAINMENT

1. NOT EFFECTIVE FOR RAPID PRESSURE TRANSIENTS

2. MINIMIZED RADIATION RELEASE TO THE PUBLIC (VESSEL LEAKAGE ONLY)

3. VERY HIGH INITIAL COST, LOW O/M COST

SLIDE 5

CONCEPTS – ASSESSMENT (CONT.)

COUPLED CONTAINMENT

1. NOT EFFECTIVE FOR RAPID PRESSURE TRANSIENTS
2. POTENTIAL FOR DEGRADING SAFETY OF SECOND UNIT
3. LARGE OPERATIONAL PENALTY FOR SECOND UNIT
4. MINIMIZED RADIATION RELEASE TO THE PUBLIC

SLIDE 6

CONCEPTS – ASSESSMENT (CONT.)

– CONTROL COMBUSTION

IGNITION SOURCES

1. HIGH POTENTIAL FOR EFFECTIVENESS DURING MOST ACCIDENTS LEADING TO CLAD OXIDATION
2. NO EFFECT ON PLANT OPERATION
3. TECHNICAL DEVELOPMENT REQUIRED
4. REQUIRE LOCAL HYDROGEN MONITORING
5. MODERATE INITIAL COST, LOW O/M COST

CONCEPTS – ASSESSMENT (CONT.)

– CONCEPTS WHICH PREVENT COMBUSTION

NITROGEN INERTING

1. EFFECTIVE IN PREVENTING HYDROGEN COMBUSTION
2. LARGELY A PASSIVE SYSTEM
3. DIFFICULT BACKFIT TO ICE CONDENSER
4. OPERATIONALLY PROHIBITIVE BECAUSE OF FREQUENT MAINTENANCE NEEDED ON ICE CONDENSER AND OTHER CONTAINMENT SYSTEMS
5. SIGNIFICANT POTENTIAL FOR DEGRADED SAFETY THROUGH REDUCED MAINTENANCE OF EQUIPMENT
6. INCREASED LOSS OF ICE
7. HIGH INITIAL COST, EXTREMELY HIGH O/M COST

SLIDE 8

CONCEPTS – ASSESSMENT (CONT.)

HALON SUPPRESSANT

1. POTENTIALLY EFFECTIVE IN PREVENTING HYDROGEN COMBUSTION

2. NO OPERATIONAL EFFECTS WITH NORMAL PRECAUTIONS

3. MODERATE HAZARD TO PERSONNEL

4. TECHNICAL FEASIBILITY NOT DEMONSTRATED

5. DECOMPOSITION PRODUCTS MAY PRODUCE SEVERE CONSEQUENCES

6. ACTIVE POST ACCIDENT WITH SHORT BUT REASONABLE TIME TO MANUALLY ACTIVATE

7. HIGH INITIAL COST, LOW O/M COST

RESULTS AND CONCLUSIONS

– MOST PROMISING CONCEPTS FOR HYDROGEN CONTROL SELECTED FOR A RIGOROUS DEVELOPMENT PROGRAM ARE:

1. IGNITION SOURCES

2. HALON SUPPRESSION

– SIGNIFICANT IMPROVEMENT IN PHYSICAL MODELS AND COMPUTER CODES ARE NEEDED

– FILTERED VENTED CONTAINMENT IS UNACCEPTABLE FROM RELEASED DOSE

– INERTING IS NOT FEASIBLE FOR AN ICE CONDENSER CONTAINMENT

– RISK AT SEQUOYAH COMPARABLE TO WASH 1400 REFERENCE PLANT

PROGRAM FOR DEALING WITH DEGRADED CORE CONDITIONS

- WE HAVE ORGANIZED AN EIGHT-MAN FULL TIME TASK FORCE FOR DESIGN AND DEVELOPMENT WORK ON DEGRADED CORE ACCIDENTS.

- WE ARE IMPLEMENTING IMMEDIATELY THE DESIGN AND INSTALLATION OF AN INTERIM DISTRIBUTED IGNITION SYSTEM (PHASE 1) TO BE OPERATIONAL WITHIN TWO TO THREE MONTHS.

- WE ARE IMPLEMENTING IMMEDIATELY DEVELOPMENT WORK TO UPGRADE THE INTERIM DISTRIBUTED IGNITION SYSTEM (PHASE 2) AS IMPROVED ASPECTS OF THE SYSTEM CAN BE DEVELOPED.

- WE WILL COMPLETE A LONG-TERM STUDY AND DEVELOPMENT EFFECT FOR CONTROLLED IGNITION SYSTEMS WHICH WILL LEAD TO BACKFITTING THE PHASE 1 & 2 SYSTEMS, IF NEEDED. (PHASE 3) THE LENGTH OF THE STUDY SHOULD BE APPROXIMATELY TWO YEARS.

- WE ARE IMPLEMENTING IMMEDIATELY A DEVELOPMENT EFFORT TO UNDERSTAND THE POTENTIAL NEGATIVE ASPECTS OF HALON AS A HYDROGEN BURN SUPPRESSION.

slide 11

Degraded Core Task Force Program

Major Tasks

1. Controlled Ignition
2. Halon
3. Risk Assessment
4. Core Behavior, Hydrogen Generation and Transport
5. Hydrogen Burning and Containment Responses
6. Containment Integrity
7. Equipment Environmental Qualifications
8. Radiation Dose Code
9. Hydride Converter, Fogging, and Others
10. Rulemaking and State of the Art

Tape #
+12

NRC SEVERE ACCIDENT

FILTERED VENTED CONTAINMENT SYSTEM

STUDY

July 9, 1980

ACRS SUBCOMMITTEE ON THE SEQUOYAH NUCLEAR PLANT

James Meyer, NRR

PURPOSE OF NRC'S SEVERE ACCIDENT MITIGATION FEATURES STUDY

TO DETERMINE HOW IMMEDIATE AND PRACTICAL TECHNICAL FIXES CAN BE IMPLEMENTED IN THE ZION 1/2 AND INDIAN POINT 2/3 UNITS THAT ASSURE A REAL AND SIGNIFICANT REDUCTION IN SOCIETAL AND INDIVIDUAL RISK DUE TO SEVERE ACCIDENTS, INCLUDING CORE MELT

THE GENERAL APPROACH IS TO PURSUE ACTIVELY THOSE DESIGN FEATURES THAT CONTRIBUTE FAVORABLY TOWARD THE MITIGATION OF THE CONSEQUENCES OF A SEVERE ACCIDENT.

GOAL #1

TO ANSWER THE QUESTION:

WILL THE PROPOSED FEATURES SIGNIFICANTLY MITIGATE THE CONSEQUENCES OF SEVERE (CORE-MELT) ACCIDENTS?

FOR THAT SET OF SEVERE ACCIDENTS THAT BOUND THE EXPECTED DOMINANT SEQUENCES FOR Z/IP: WHAT DO PRACTICAL MITIGATION FEATURES LOOK LIKE AND WHAT DO THEY ACHIEVE IN TERMS OF ATTENUATION AND DELAY OF RELEASE?

SEVERE ACCIDENT FEATURES/STUDIES UNDER CONSIDERATION

- FILTERED-VENTED CONTAINMENT SYSTEM
- CORE RETENTION DEVICES
- HYDROGEN CONTROL METHODS
- STEAM EXPLOSION EVALUATION
- ACCIDENT RISK CALCULATIONS

4

FILTERED VENTED CONTAINMENT SYSTEM

INPUT: PRESSURE, TEMPERATURE, AEROSOL, AND RADIOLOGICAL SOURCE TERM HISTORIES FROM MARCH/CORRAL AND INDEPENDENT ANALYSES. VARIATION OF HISTORIES DUE TO THE PRESENCE OF: CORE RETENTION DEVICES, HYDROGEN CONTROL, ACCUMULATOR WATER CONTROL, RESTORATION OF AC POWER.

CONCEPTUAL DESIGNS: CONSIDERATION OF PRACTICAL LAYOUTS, PRESENCE OR LACK OF AC POWER; DECONTAMINATION FACTORS ACHIEVABLE; PRACTICAL DESIGN FLOWS; ACTUATION LEVELS; OPERATOR/AUTOMATIC CONTROLS; VENTING TO ATMOSPHERE VS TO SPECIAL BUILDING; ENVIRONMENTAL REQUIREMENTS.

CONSEQUENCE ANALYSIS: CRAC (CONSEQUENCE) ANALYSES OF IMPACT OF MITIGATION FEATURES WITH AND WITHOUT: NOBLE-GAS ATTENUATION; AC POWER TO FVCS.

OTHER CONSIDERATIONS: REALISTIC (BEST ESTIMATE) EVALUATION OF CONTAINMENT FAILURE PRESSURES AND MODES.

UPDATE ON Z/IP MITIGATION FEATURES STUDY

1. TECHNOLOGY EXCHANGE MEETINGS CONCLUDED.

2. MITIGATION FEATURE REQUIREMENTS AND CRITERIA -- ISSUED BY NRC FOR COMMENT (JULY 1980).

3. STAFF REPORT DUE LATE FALL.

4. LICENSEE REPORT DUE LATE SUMMER.

5. RES AND NRR PROGRAMS PROCEEDING THROUGH SUMMER ON KEY ISSUES.

6

BASES FOR DEVELOPING Z/IP MITIGATION FEATURES: REQUIREMENTS AND CRITERIA

1. REDUCE RISK.
2. PREVENT CONTAINMENT FAILURE (BY OVERPRESSURE OR BASEMAT MELT THROUGH).
3. DEFINE FUNCTIONAL REQUIREMENTS FOR SYSTEM WHICH WILL PREVENT CONTAINMENT FAILURE.
4. DEFINE/DESIGN SYSTEM THAT MEETS REQUIREMENTS.
5. ASSESS CONSEQUENCE MITIGATION CAPABILITIES OF SYSTEM.
6. ASSESS "COMPETING RISK" OF SYSTEM.

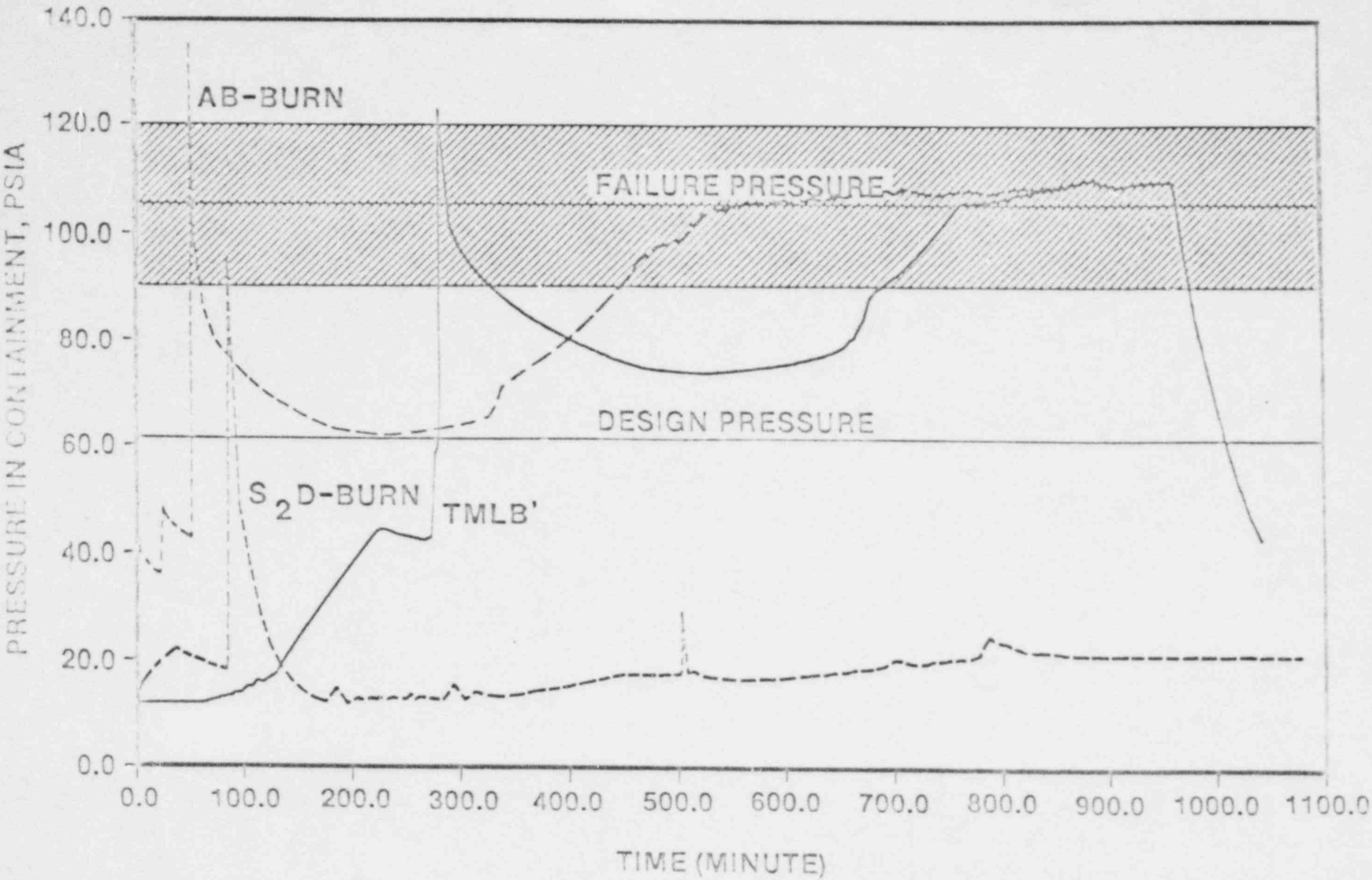
CHARACTERISTICS OF ACCIDENT SEQUENCES DOMINATING RISK

- INITIATED BY SMALL BREAKS AND TRANSIENTS
- HUMAN ERROR CONTRIBUTES SIGNIFICANTLY
- CORE MELTS
- LARGE AMOUNTS OF HYDROGEN ARE GENERATED
- CONTAINMENT SAFETY FEATURES UNAVAILABLE
- CONTAINMENT FAILS ABOVE GROUND VIA OVERPRESSURE

ACCIDENT SCENARIOS

- TMLB'
  - LOSS OF OFFSITE AND ONSITE AC POWER FOR AN INDEFINITE PERIOD (16 HOURS)
- AB-BURN
  - LARGE LOCA PLUS LOSS OF OFFSITE AND ONSITE AC POWER FOR AN INDEFINITE PERIOD. H_2 EXPLOSION WHEN CORE HITS CONCRETE
- S_2 D-BURN
  - SMALL LOCA, FAILURE OF ECCS INJECTION, H_2 EXPLOSION WHEN CORE HITS CONCRETE
- S_2 G
  - SMALL LOCA, FAILURE OF CONTAINMENT HEAT SINK
- TMLB"
  - LOSS OF OFFSITE AND ONSITE AC POWER FOR 6 HOURS, FOLLOWED BY ACTIVATION OF CONTAINMENT COOLERS, CONTAINMENT SPRAYS, AND ECCS INJECTION
- A-VENT
  - LARGE LOCA CAUSING ACTUATION OF CONTAINMENT VENTING

CONTAINMENT PRESSURE VERSUS TIME



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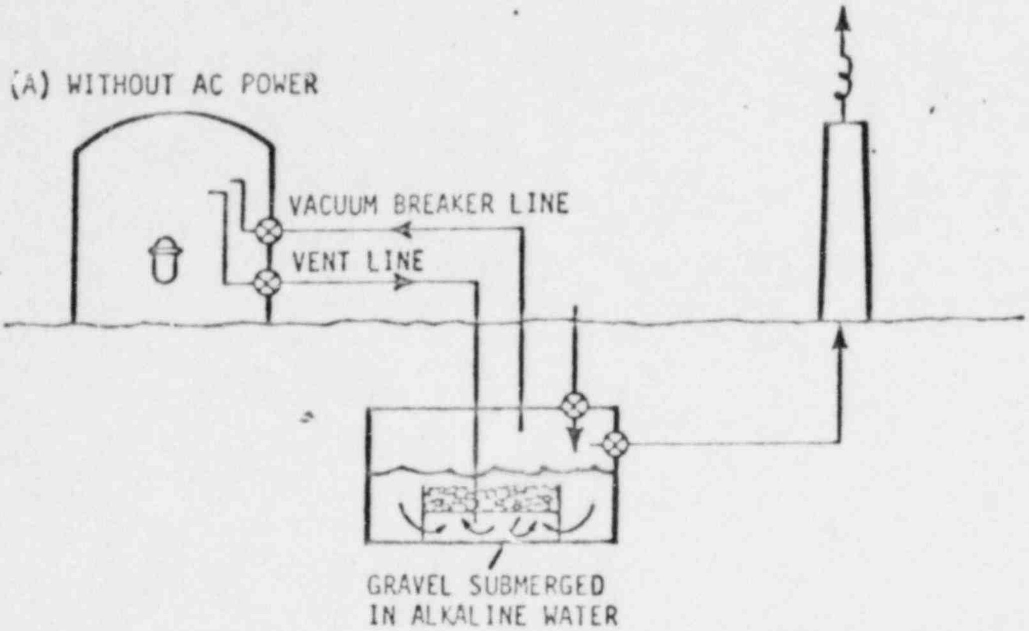
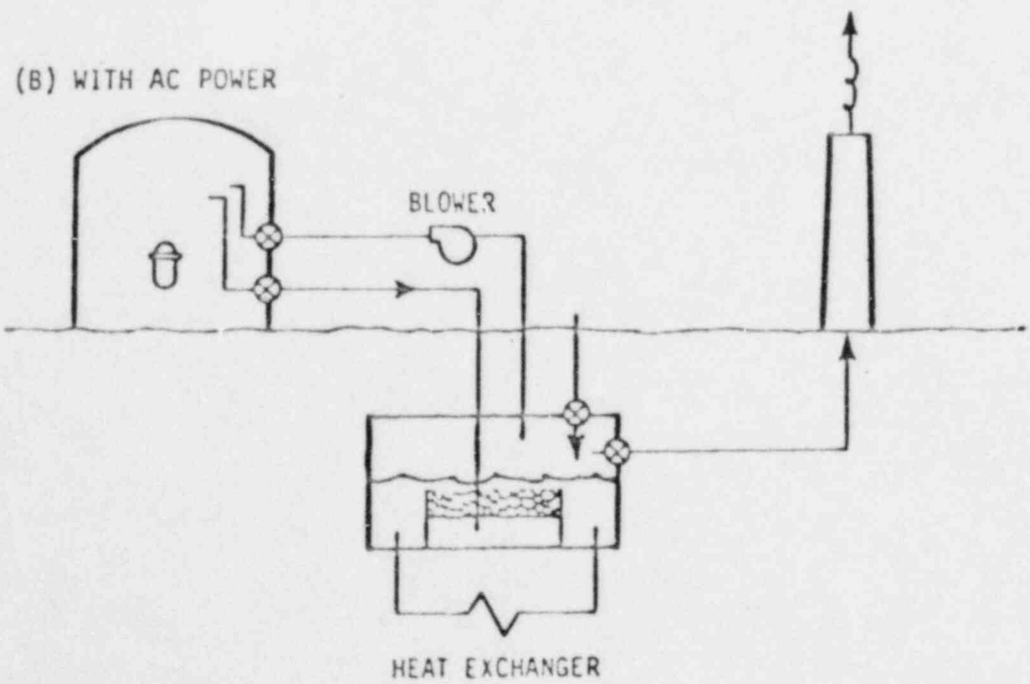
(A) WITHOUT AC POWER



(B) WITH AC POWER



VENT RELIEF MODE

PURPOSE: TO PREVENT CONTAINMENT RUPTURE DUE TO OVERPRESSURE.

PASSIVE ACTUATION: RELIEF SETPOINT BASED ON CONTAINMENT PRESSURE OF 75-85 PSIA.

PASSIVE OPERATION: CONDENSING AND FILTERING COMPONENTS DO NOT OVERHEAT FOR AT LEAST 16 HOURS.

ACTIVE OPERATION: UTILIZES HEAT EXCHANGERS AND/OR BLOWERS FOR LONG-TERM HEAT REMOVAL.

MANUAL OVERRIDE: INHERENT LOGIC TO DISCOURAGE OVERRIDE UNLESS NECESSARY.

FLOW VOLUME: 160,000 CFM TOTAL AT DESIGN CONDITIONS; 40,000 CFM NONCONDENSIBLE FLOW.

28/18

RECIRCULATION MODE

PURPOSE: ALTERNATE CONTAINMENT HEAT REMOVAL/DEPRESSURIZATION/ ATMOSPHERE CLEANUP SYSTEM.

OPERATOR ACTUATION: CAN BE ACTUATED AT ANY TIME. DOES NOT OVERRIDE VENT RELIEF ACTUATION AT CONTAINMENT PRESSURE SETPOINT.

ACTIVE OPERATION: VARIABLE SPEED RECIRCULATION BLOWER, HEAT EXCHANGER.

FLOW VOLUME: MAXIMUM 160,000 CFM TOTAL FLOW; 40,000 CFM NONCONDENSIBLE FLOW.

POSSIBLE PROBLEMS: HYDROGEN FLAMMABILITY.

13 85

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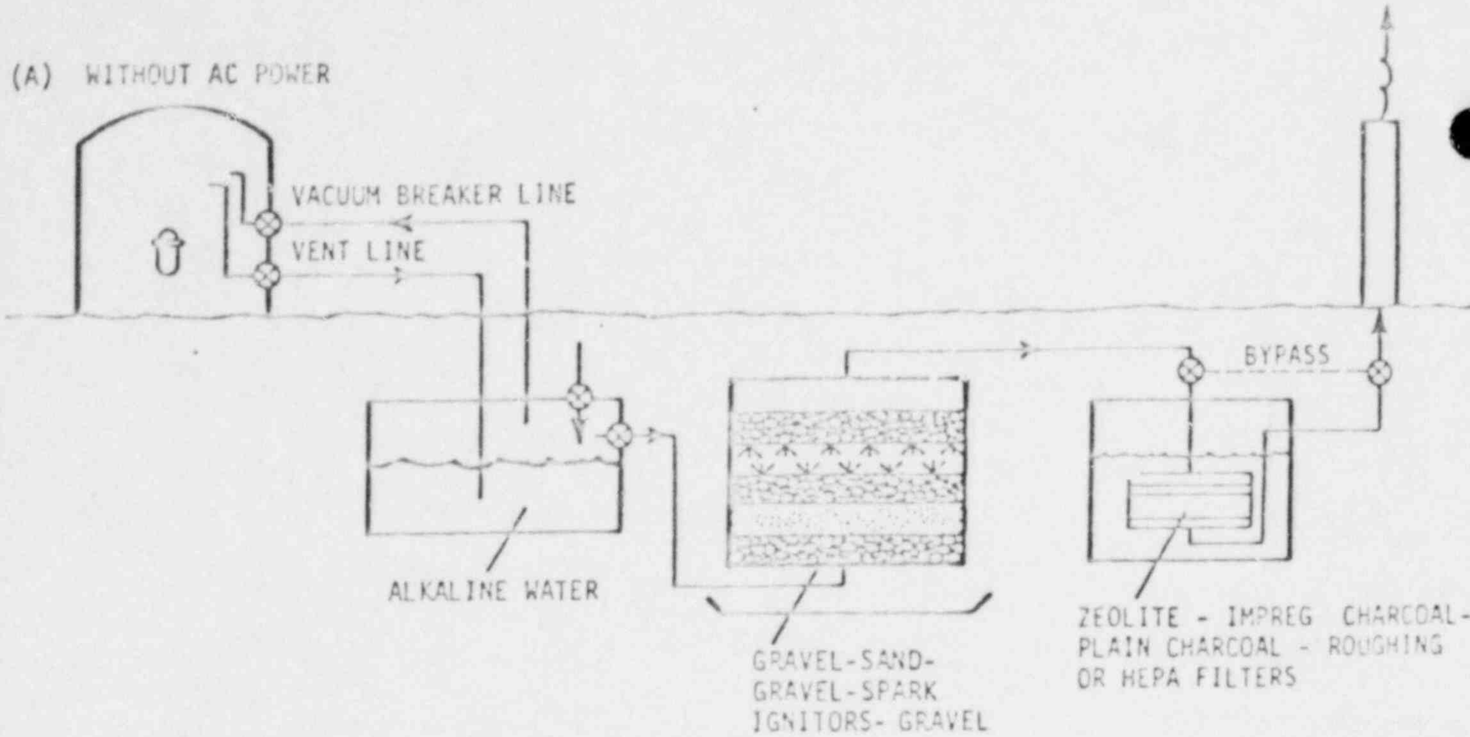
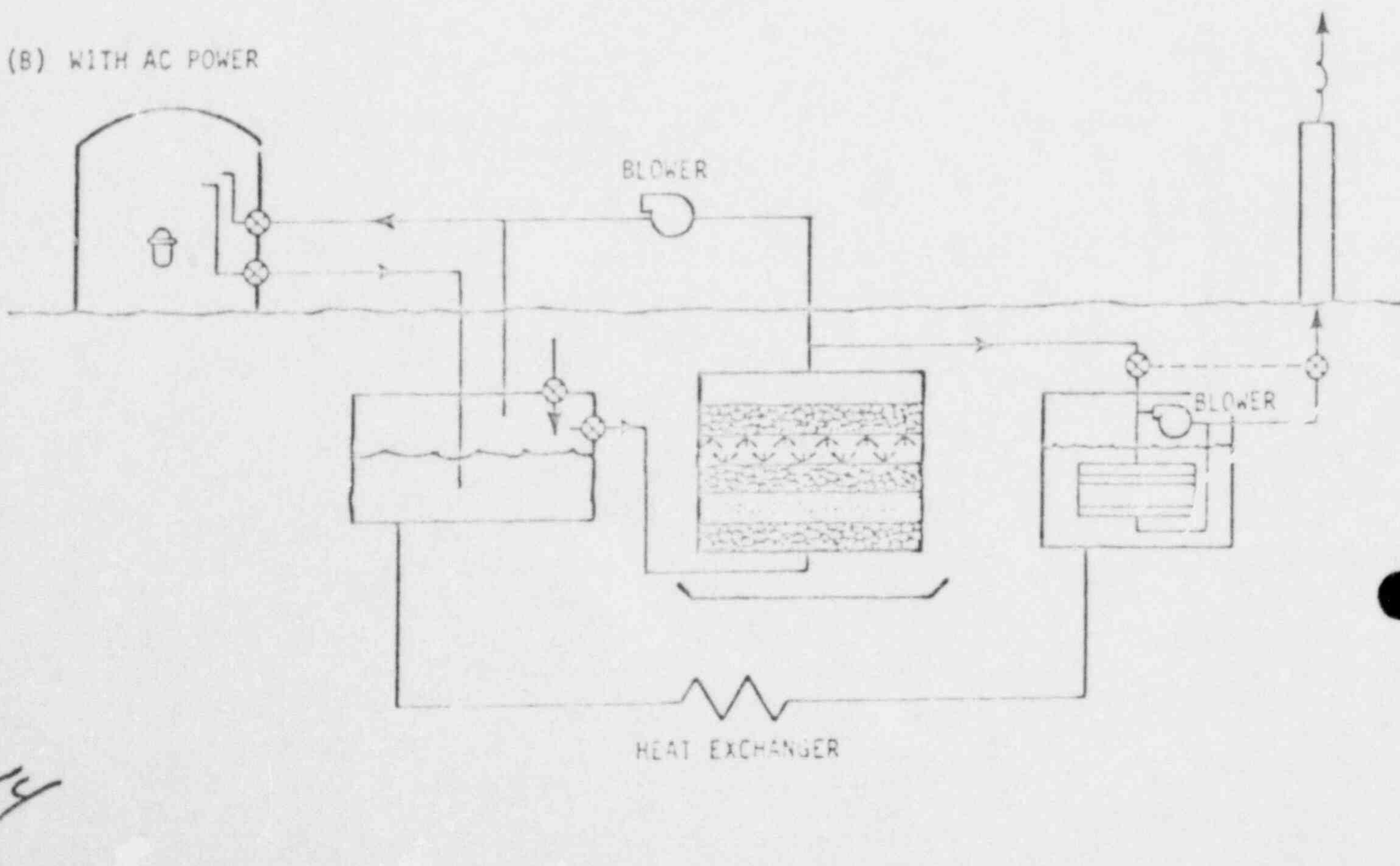
(A) WITHOUT AC POWER



(B) WITH AC POWER



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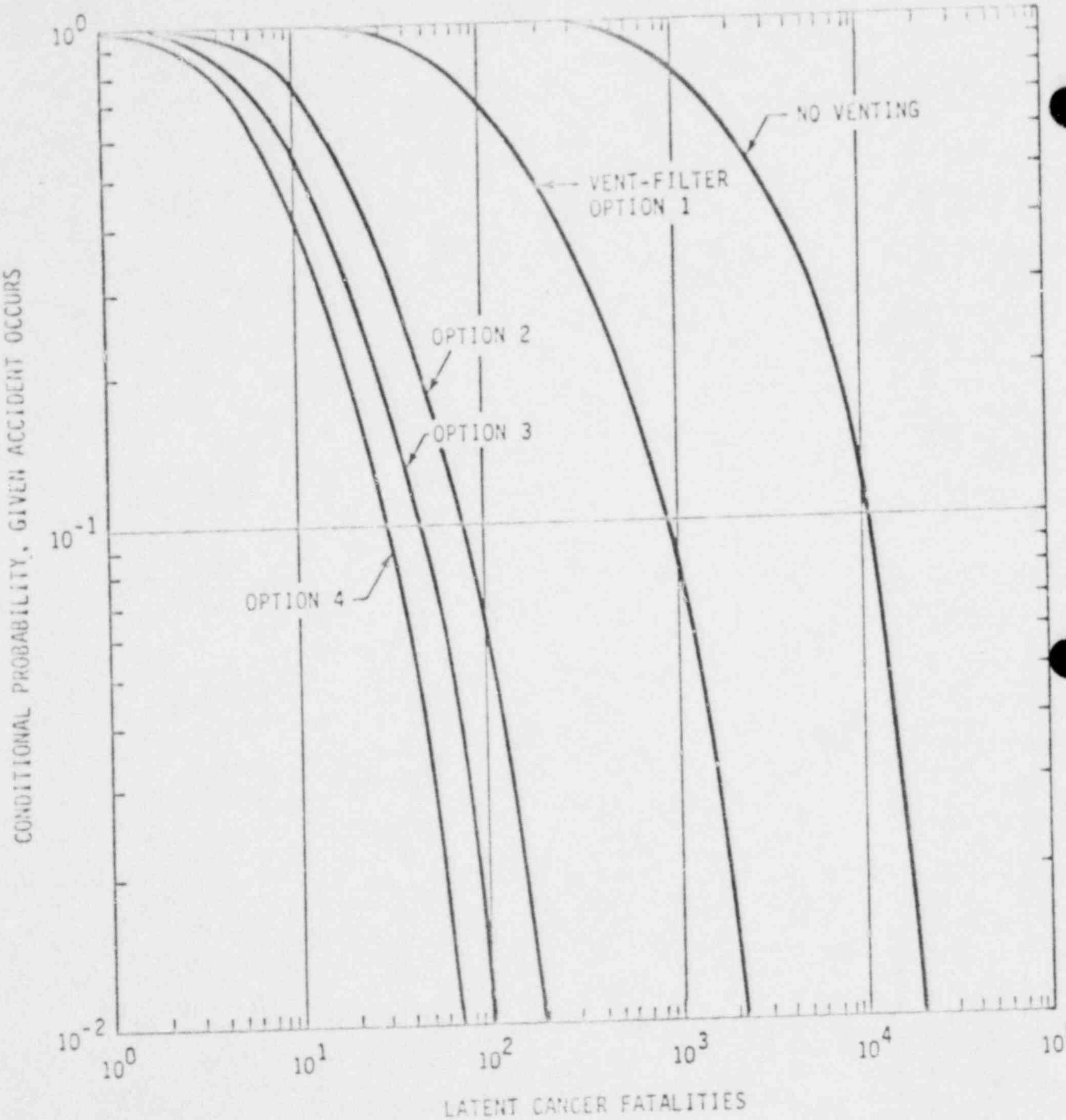

Figure I-24 Probability of Latent Cancer Fatalities for Various FVCS Options, Given Occurrence of Accident TMLB', Indian Point Site.

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POSTULATED VENT STRATEGIES FOR EXISTING REACTORS

VENT STRATEGY	EFFECTIVENESS FOR SEVERE ACCIDENTS	INTERFERENCE WITH LESS SEVERE ACCIDENT
• LOW VOLUME CONTAINMENT PRESSURE RELIEF	LOW	LOW
• LOW VOLUME CONTAINMENT DEPRESSURIZATION.	↓	↓
• HIGH VOLUME CONTAINMENT PRESSURE RELIEF	↓	↓
• HIGH VOLUME CONTAINMENT PRESSURE RELIEF PLUS LOW VOLUME CONTAINMENT DEPRESSURIZATION	↓	↓
• PASSIVE CONTAINMENT FLOODING PLUS LOW VOLUME CONTAINMENT RELIEF OR DEPRESSURIZATION	↓	↓
• DIVERSION OF SUMP AND ACCUMULATORS PLUS LOW VOLUME CONTAINMENT RELIEF OR DEPRESSURIZATION	↓	↓
• ANTICIPATORY PRIMARY SYSTEM DEPRESSURIZATION PLUS LOW VOLUME CONTAINMENT RELIEF OR DEPRESSURIZATION	↓	↓
• ANTICIPATORY CONTAINMENT DEPRESSURIZATION	HIGH	HIGH

(10)

SYSTEM INTERACTIONS

- CAVITATION OF RECIRCULATION PUMPS DUE TO SUMP FLASHING
- SEVERE VACUUM CAUSED BY DELAYED SPRAYS OR COOLERS
- DEGRADATION OF REFLOOD CAUSED BY PREMATURE VENTING
- OTHER INTERFACES

SOLUTIONS

- DESIGN CHANGES
- ADMINISTRATIVE PROCEDURES

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Table B.1 Possible FVCS Systems Interactions

System	No Effect	Interactions Control	Performance	Function	Comments
Electric Power	X				
Spray Injection				X	FVCS action could negate function or cause vacuum.
Coolant Injection		X	X	X	FVCS action could negate trigger or degrade performance.
Spray Recirculation			X	X	FVCS action could negate function; or create vacuum; action could cavitate pumps.
Containment Cooling		X	X	X	FVCS action could trigger change from emergency to normal operation or cause vacuum.
Coolant Recirc.			X		FVCS action could cavitate pumps.
NaOH Addition	X				
Reactor Protection	X				
Power Conversion	X				
Auxiliary Feedwater	X				
Pressurizer S/R Valves			X		FVCS action could degrade seals.
Chemical Volume			X		FVCS action could degrade seals.
Residual Heat Removal			X		FVCS action could degrade seals.