ORIGINAL

acrst- 1429

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the matter of:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

303rd General Meeting

Docket No.

Location: Washington, D. C.
Date: Friday, July 12, 1985

Pages: 166 - 285

ANN RILEY & ASSOCIATES
Court Reporters
1625 I St., N.W.
Suite 921
Washington, D.C. 20006
(202) 293-3950

TRO4
d1

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

303RD GENERAL MEETING

Room 1046

1717 H Street, N.W.

Washington, D.C.

Friday, July 12, 1985

The second day of the public meeting of the Advisory Committee on Reactor Safeguards was convened, pursuant to notice, at 8:30 a.m., David Ward, Chairman of the Committee, presiding.

ACRS MEMBERS PRESENT:

D. Ward, Chairman	H. Lewis
J. Ebersole	D. Moeller
W. Kerr	M. Carbon
D. Okrent	H. Etherington
G. Reed	C. Wylie
F. Remick	P. Shewmon

ACRS MEMBERS [Continued]:

C. Mark   C. Siess

R. Axtmann

ALSO PRESENT:

R. Major, Federal Designated Official

R. Fraley, ACRS Staff Member

D. Savio, ACRS Staff Member

PRESENTERS:

E. Adensam	R. Vij
T. Kenyon	D. Hankins
R. Pierce	R. Villa
D. Foreman	C. Thomas
D. Scaletti	J. Knight
C. Cheng	M. Reich
J. Rosenthal	R. Klecker

P R O C E E D I N G S

MR. WARD: The meeting will now come to order. This is the second day of the 303rd Meeting of the Advisory Committee on Reactor Safeguards. During today's meeting the committee will first hear a report of the ACRS Subcommittee on Quality Assurance Activities at the Watts Bar Nuclear Station and meet with representatives of the NRC Staff and TVA.

Second, continue discussions regarding the proposed NRC Quantitative Safety Goals. Three, continue discussions regarding the General Electric Standard Safety Analysis Report, GESSAR-II. Four, discuss EPA standards for high level waste repository. Five, hear reports from ACRS Subcommittees on Air Systems on the ANL West survey of control room habitability practices; and Long-Range Planning for the NRC; Human Factors and Maintenance Subcommittees on aptitude selection procedures for operations and mainten nce personnel. And six, at the end of the day, since the meeting ends about 5:45 we will have a first reading of the sabotage letter. So we can plan our day to extend beyond the 5:45 agenda.

The schedule for Saturday is posted on the bulletin board outside this meeting room. This meeting is being conducted in accordance with provisions of the Federal Advisory Committee Act and the government and the Sunshine Act. Mr Richard Major is the designated federal official for

the initial portion of the meeting.

A transcript of portions of the meeting is being kept, and I request that each speaker use one of the microphones, identify herself or himself and speak with sufficient clarity and volume so that she or he can be readily heard.

We've received no written statements nor requests to make oral statements from members of the public regarding today's sessions.

We'll begin with a subcommittee report from Mr. Reed on the June 26th meeting of the Quality Assurance Subcommittee. Mr. Reed.

MR. REED: First of all, I was somehow stuck into the position of Acting Chairman for the Watts Bar QA Design Construction Review. I don't know how Forrest Remick -- who is head of the QA Subcommittee? Oh. -- how Forrest Remick and Jesse Ebersole pulled this off, but never have I accumulated so much paper in so short a time. So you can imagine that the depth of review suffers a bit.

Well, we had a subcommittee meeting on June 26th and attending were Wylie, Siess and Reed. It seemed that such a subcommittee meeting had to occur because of the ACRS operating license letter on Watts Bar issued on August 16, 1982, which said in part: "Late in the construction program a serious quality assurance breakdown was identified,

principally in the construction area, but also in the design area. The effects of the breakdown persist and corrective work on the plant will continue at least throughout 1982. TVA invoked major quality assurance programmatic changes including plans to have an independent contractor review of the design and construction of a typical vertical section of the plant to confirm the adequacy and safety of the as-completed plant. This issue should be resolved in a manner satisfactory to the Staff, and we wish to be kept informed."

As it worked out, two weeks ago when we had our subcommittee meeting, -- that was June 26th -- the Watts Bar design and construction QA issue did not appear at the time to the three of us to be too murky. But things may have become so since June 26th, and we will try to get that clarified today if we can, or try to find some judgment of our own.

What appeared to be the scene on June 26th was that Black & Veatch had done an independent vertical slice QA review on the aux feed system at Watts Bar on Unit 1, and this was a good system selection in view of its importance and in view of its interfacing, and in view of the design aspects and the complexity, and of course, the safety issues involved.

However, at the time this Black & Veatch vertical slice QA review was done, it was not a completed system. Therefore, based on the Black and Veatch mandate, which was to log and explore any inconsistency, -- I believe these words

that we got out of the subcommittee meeting -- that they were told to look for any inconsistency in the aux feed system. And to me, "any inconsistency" is wording that goes beyond deficiency.

Based on that mandate, Black and Veatch came up with some 428 so labeled findings which were, I guess, any inconsistency, and of course, the process was set in motion for the resolution of this large list of 428 findings.

As we understand it, TVA's inhouse Independent Review Policy Committee worked with Black and Veatch in these resolutions, as they had previously done in horizontal QA slice work that had been performed by other outside contractors, Theodore Barry & Associates and United Engineers.

So what you have here is a case of previous horizontal, slices looking at their QA by independent parties, and now a large vertical slice of these 428 findings.

Now, the findings began to be resolved into 165 construction incompletes. That is sort of bothersome because it almost says, I guess, in our letter, the ACRS letter, that we wanted to know about the as-completed plant. And so there are these 165 which sort of muddied the water -- 165 construction incomplete.

Now, 115 of the inconsistencies or findings were found to be -- by the resolving parties, the TVA committee and Black & Veatch, I assume -- to have no real deviations. So

115 came to be "no real deviations."

There remained 148 which required further generic considerations. So you start out really with 148 that you've got to study and resolve, and all of these large numbers of inconsistencies certainly sounded alarming to the subcommittee. And we had some difficulty at the outset of the meeting trying to decide whether the Watts Bar issue was one of, let's say, one-upmanship of competing parties, or whether it was an issue of quality of paper without respect to quality of equipment installed in the workplace or whatnot, because as we all know, ACRS has been interested in quality as installed in the workplace, and we sometimes wonder that paper is the route to that, necessarily, and that maybe we focus too much on assuring the signatures are on the paper and not the quality of the equipment in the workplace.

Therefore, in view of the large numbers and the fact that we were doing a quick review -- and we are going to do it here in Washington based on people who appeared before us -- we became most interested in the presentation of the Black & Veatch representative. I guess we called him Dr. Zid finally because I couldn't pronounce his name, but I will have a go at it. Zidziunas, I guess.

And the Black & Veatch representative stated that Unit 1 Watts Bar had been designed and is being installed in accordance with the TVA licensing commitments, except for

three identified unresolved issues as follows.

Now, we asked him to review that and he very clearly said yes, things were fine except for these three issues. Okay. The three issues are, one, there are some problems with respect to cable tray loadings and with respect to fire retardant coatings that had been put on cables. So there is some issue of derating factor or something to do with cable trays, and that is an unresolved issue still as far as we know.

The second issue was the loadings on embedments, plates, with respect to restraints and other attachments that are centrally attached, ecentric loadings and pry loadings. So we have the plate attachments.

And the third had to do with seismic design and the way in which they had broadened or used the peaks on the rough seismic spectra.

So those three didn't sound too bad, I think, -- and the other members will speak for themselves -- with respect to the subcommittee review.

In addition to Black & Veatch previous work and the committee of TVA people that worked on QA reviews, there was a named body of people by the TVA Board of Directors which was called the Nuclear Safety Review Staff of TVA. And they had, I guess, a mandate to report to the Board of Directors and they went around to look at all these problems and to make

judgments on that.

Well, the NSRS group assessment concluded that the work that had been done by Black & Veatch and the TVA staff was appropriate, and that there were no issues which would preclude immediate approval for operation of Watts Bar Unit 1.

At the subcommittee meeting on June 26th, the NRC Staff was not prepared to agree or disagree to any extent with the TVA or its NSRS or with the Black & Veatch report. Further, the NRC Staff had just received a draft of an INPO design and construction evaluation report which they were not prepared to comment on, and of course, the status of release of that report I'm sure was also in question at that time.

Now shortly, we will hear from the NRC Staff on where they stand on Watts Bar today, but before that I want to point out another issue related to the Watts Bar situation which has to do with allegations of construction workers -- I don't know on whose payroll -- and allegations of regular employees, perhaps to become operating personnel.

Now it seems to me -- and I will pass around something here if I can separate it. This is part of the activity. Apparently, TVA decided to go out because allegations can play in last-minute problems in licensing. They decided to go out and solicit input from all the employees that are, I guess, to be terminated or new hires or regulars or whatnot, and they hired I guess it would be some

sort of -- well, it was a company called Quality Technology Company -- to go out and conduct a mod management survey and get all kinds of employee allegation inputs, and postings such as this were put up around the plant.

And of course, in my opinion, when one starts to do that, you will get perhaps good allegations and you will get a multitude of chronic dissensions and fun and games type allegations.

So you certainly have to say that TVA was looking for any kind of allegations that they could in a very hard way. You might even say they were looking for Pandora's Box. Because if you give lots of thoughts to these allegations you will find that a lot of it is wattless and it just burns up time. So I don't know they're going to do deal with that.

So that's an issue; that the allegations have not all been boiled and put into a ball of wax. However, it's in the ball of wax, and I guess with respect to this QA issue at Watts Bar the allegations have to be sorted out and handled. And that's a hanging issue at this time. I don't know what we'll hear today.

Also in the last few weeks, and maybe in the last few months, the TVA management issue has ballooned with lots of flack about mismanagement, generic problems affecting all the TVA plants, and the latest is even big shifts in organizational structure. And I even see complaints about low

pay, and maybe that will affect all of us by raising GS-16's to something else.

Now, I don't have a lot of sympathy. I think it's behind these other things. There are probably more generic issues other than that.

MR. WARD: Ray, you haven't told Glenn that everybody else is getting 18, have you?

[Laughter.]

MR. REED: Now, to me, the Watts Bar issue and all this hoopla about TVA in general has muddied the waters of the Watts Bar QA issue. Quite frankly, I think -- and a lot has happened; Wylie and Chester will have to speak for themselves, but I think that the two issues can be kept separate. If we can't keep them separate I think we're in big trouble. If we can't keep on the agenda of Watts Bar QA, we're not going to come up with anything for a long period of time.

The overall issue is a much larger issue for which really no ACRS subcommittee exists, nor assignment of review exists. And we have got to keep the waters from getting too muddy in my opinion.

I have had some input. Ray Fraley I think feels that the water has gotten so muddy that there's no way in which we can move forward on the Watts Bar issue by itself.

And now another complicating issue. I believe that the INPO evaluation report has just been released by TVA to

the public, is that correct?

MR. PIERCE: I am Ralph Pierce, the Watts Bar Project Manager. We received the draft report from INPO on July 5th. We're in the process of evaluating this report. It has not been released to the public per se yet.

MR. REED: Is it released for discussion at this meeting?

MR. PIERCE: We are not prepared to discuss the INPO report at this meeting.

MR. REED: Well, you don't have to discuss it but since it has been handed out in packages to the ACRS members, -- is Ray here? Ray, I thought I got the word yesterday that we could address this report.

MR. FRALEY: We did contact INPO and they indicated that they considered it a public report now and it could be handled as a public report.

MR. PIERCE: That's fine. I will try to answer any questions you may have on it.

MR. REED: I don't expect you to. I just wanted to verify with you that it is not under wraps.

MR. PIERCE: I have a copy of the report with me.

MR. REED: As we all know, these INPO reports are kept under wraps, and unless the company involved releases them or allows that they may be released -- and I think it would be good judgment on the part of TVA and probably they

have done this at the top -- to release this, because the waters are so muddy and the accusations are so flying you might just as well get it all out. We all live in a glass house in nuclear things, so to speak. A glass house gets so misinterpreted anyway.

So this INPO report is out and I have reviewed it, and I am sure the Staff has reviewed it.

MS. ADENSAM: Mr. Reed, I would like to clarify what report we are talking about. The only document the Staff has available is a set of INPO field notes which are part of an exit interview dated June 20, 1985. We do not have a copy of the draft INPO report.

I would like to also add we were not prepared to discuss it at this meeting today.

MR. REED: Oh, I can understand that. This is a June 20, 1985 exit meeting, construction project evaluation at Watts Bar Nuclear Plant.

MS. ADENSAM: Yes, sir, that is the document we have. We provided it to the ACRS Staff yesterday.

MR. REED: I will make the point on how these things go through their mechanisms anyway.

So we have this exit meeting report, and I am familiar with INPO. INPO in the past has, I think, been guilty of superchicken, I will call it. They have many times laid down a lot of their ideas, and their activity is

excellence, not minimum standards. Okay? Sometime their people become very enthusiastic and do criticize very heavily.

Now, I have gone through this INPO report, and based on what I have seen in the past, this one is not, in my opinion -- I can't pick out a lot of superchicken items. It is harsh, which is typical of INPO. They are trying to cause excellence in the workplace, so they lay it on the line.

However, you people have a copy. You should look at it a little bit, and it will perhaps affect what we come up with here today. But certainly the next step is that TVA -- this is not a final report. TVA now has a response activity to this report, and I assume that they are working on that, and their responses will either blow away a number of these things or they will stand as the criticism is.

So that's a very confusing situation with respect to us writing any position or taking any position because if you read those things, you will find that the INPO exit report, their planned report points out a lot of things that Black & Veatch didn't include, apparently, and perhaps the TVA committee didn't include, their Board of Directors committee.

So I am in a big state of confusion, and I am going to ask Chester and Chuck Wylie to make their comments, and then I guess we are going to leave it up to Ms. Adensam to straighten us all out with respect to how the Staff feels because they haven't volunteered, and didn't on June 26th,

their positions, and see where we go from there.

Chuck, do you have any comment?

MR. WYLIE: Well, I think Glenn very accurately and adequately described the situation, and as far as -- I think he is right. We must separate the quality assurance issues from the other problems and stick to the quality assurance issues in order to get a resolution of this.

As far as boiling down to the three items, the cable tray loading and the coverage on the cable trays is the one that I am the most competent in. That is a common problem on all construction jobs. It is not unusual, if you don't give attention to the installation of those cables on a continuing basis, that they very quickly become overfilled by the field installers, and it requires close control by the quality people on the job in order to get those installed properly.

It is not a safèty issue in that the cables are within their design loadings and they are not going to overload. It is just the fact that they seemingly overfill the cable trays because of sloppy installation. But as I say, that is not peculiar to TVA; I have seen it lots of places.

MR. ETHERINGTON: Overfilling volume or weight?

MR. WYLIE: Volume, I guess you could call it, because basically what the designers do, they take something like 40 percent of the available volume of the cable tray system, and they set a limit that if you take the actual

calculated cross-sectional area of the cables themselves compared to the available cross-sectional area of the cable tray, they say, well, we will fill them 40 percent.

What happens, they put them in like spaghetti and they end up stacking on top of each other, and they very quickly overfill the trays because they are sticking above the rungs of the trays.

From the safety standpoint it's not really a problem, but --

MR. EBERSOLE: Chuck, there we may be allowance for the sloppy work. I remember 25 or 30 years ago those cables were trained, and they laid flat in the cable trays and they came out in beautiful, oriented arrays. Now they throw them up like they have flung them from 20 feet away.

MR. WYLIE: I think it is the numbers involved. You take a typical nuclear unit that has 20,000 cables in it. Back in the days when you saw those, you were talking typically maybe 2000 cables in a plant, so it is a matter of the volume and the amount of cable that has to be put in those trays.

MR. EBERSOLE: If they are thrown up there loosely, they still get cooled. It doesn't matter. As a matter of fact, they get cooled better.

MR. WYLIE: Oh, yes. It's not a safety problem; it's just an appearance problem. They are within the cable --

well, they noted here that some of them were not within the confines of the cable trays, but also they noted that some were coiled up in the cable trays, and that shouldn't be. You shouldn't coil them up in cable trays. But again, it takes close field control to straighten that out.

On jobs that I have been involved in, we have had to do the same thing. We have had to straighten those out.

MR. EBERSOLE: Let me ask a question on the matter of the hangers. I remember back also some 20 or 23 years ago, none other than Chairman Palladino was poking through the gas reactor at Oak Ridge, and he points with his finger to the variety of the hangers on pipes, some of which carried 1200 degrees steam, and asked a series of questions: I want you to assume that any given hanger falls or it cuts loose, and what is the consequence?

So, what is the rationale here? Do we know about the failure of any given hanger on any pipe? I certainly hope that that doesn't mean any catastrophic consequence, that that pipe will fail because one of its hangers came down. And I can't get excited very much if that is the case.

On the other hand, if a hanger comes loose and we have a pipe failure, that is a horse of a different color. Does anybody know what the design logic is here about failure of single hangers?

MR. WYLIE: I don't.

MR. EBERSOLE: What is the requirement in design on the hanger in the hanger picture?

MR. REED: Well, Jesse, to keep us on track here --

MR. EBERSOLE: I am trying to assess the second embedment plate problem.

MR. WYLIE: Maybe I can get the answer from my experience as to how they design those, but they actually design the cable trays for not only the dead weight of the cable but for the seismic design, you know, associated with it; but they also design it for the workers getting up in there and pulling cable trays, walking through those cable trays. So generally, they are well overdesigned.

MR. EBERSOLE: Well, I certainly hope we will not find that failure of a pipe hanger results in a pipe break, whether it is on a single embedment plate or whether there are 15 pipes hanging from it.

MR. REED: Well, that is one of the things that is yet unresolved. I was going to ask you, since you dumped this acting chairman job on me, I was going to ask you for your input before -- well, Chet will be back shortly, but do you have any other input in general that you want to put into this?

MR. EBERSOLE: You know we had an intermediate meeting and all things seemed in order. About mid-spring they were working on the fire protection problem, and then these

other matters came up later.

I would like to ask the Staff to look at this question about the singularity of failure and its consequence in pipes at large, not just necessarily just at Watts Bar; anywhere.

MR. KERR: Mr. Chairman, I think I heard comments that said we should concentrate on this safety on the QA program here and not get involved in other problems, and then I heard comments that said one of the big problems is that the cable trays -- this is not a safety problem, it's an appearance problem. I may be oversimplifying.

If what we are talking about are QA problems and we don't know what the safety implications are or we may be convinced that they aren't relevant, and if the principal QA problem is an appearance problem, not a safety problem, I can't believe that ACRS should spend a lot of time on this. So I must be missing something.

MR. REED: Well, the Black & Veatch report, I guess, boils down to those three issues, one of which is the cable tray loading issue, and the other, as Charlie has defined it, is more of an appearance problem. The volume requirements for QA, the volume in the tray has been met.

I don't think those are the big things we are facing today with respect to resolving this. I think it is these peripheral things that have come in, the INPO document, the

needed Staff input, and I had forgotten one, and I had asked the Staff to address that today, and that is the two memos, I believe, that Henry Meyers has written to someone in the Staff.

So after we get Chet's initial input here, we will let the Staff --

MR. KERR: What is it that we should be concerned about? I mean which one of the many issues?

MR. REED: I think what we have to be concerned about is did TVA meet the requirements that the ACRS laid down in the April 16, 1982 letter, and is the QA activity at Watts Bar satisfactory, and to stay on that issue.

MR. KERR: You mean satisfactory to the ACRS?

MR. REED: Right.

MR. KERR: Thank you.

MR. AXTMANN: Am I right in believing that the QA problem is now beyond the vertical slice?

MR. REED: I believe it may be. We will listen to the Staff. But since the vertical slide of Black & Veatch looked fairly easy, fairly clear that there wasn't a big problem, and the TVA inputs are saying it's not a big problem, and now we have this INPO thing which seems to have muddied he waters from the QA issue.

MR. AXTMANN: How would you describe the INPO effort? It's a vertical horizontal slide?

MR. REED: Oh, it's an overall onsite many-numbered personnel walking around activity, interviewing activity, checking activity, checking paper activity. So I have to say it's an overall look at the general quality assurance and operational and construction -- design and construction. No operational.

Chet, do you have any comment?

MR. SIESS: Not an awful lot. I have some difficulty in focusing on a QA problem versus a quality problem, and I do believe that the Black & Veatch study and the internal study both went down to the quality level. They looked at the product as well as at the QA, and I think that did give me some comfort because the Black & Veatch study jus did not find any deficiencies in the design or the execution that I thought were significant, and I think that gave me some comfort.

But that is a fault of mine. I tend to look at the bottom line rather than the QA, and I don't put that much faith in the QA one way or the other. If it's bad, I can't quite convince myself that the plant is bad, and if it's good -- we had a good example yesterday where the paper was all there but the supports were not. So I have that problem.

But the Black & Veatch study did go down right to the plant itself, and I got some comfort from that.

I'm not completely up to date on the INPO situation,

but I think the Staff has the thing in hand.

MR. REED: Elinor, are you ready?

MS. ADENSAM: Yes.

MR. SIESS: Incidentally, all our letter said was that we want to be kept informed. We said this issue should be resolved in a manner satisfactory to the Staff and we wish to be kept informed, and I would certainly say we have been kept informed.

MR. WARD: And also, keeping the ACRS informed is an obligation of the Staff, not of TVA. There seemed to be some implication that TVA somehow had a duty to keep us informed of something or to respond to our letter.

MR. SIESS: I didn't make it.

MR. WARD: Oh, okay.

[Slide.]

MS. ADENSAM: My name is Elinor Adensam. I am Licensing Branch Chief in charge of licensing on Watts Bar.

I guess I had a little misunderstanding with Mr. Reed. He has covered a good bit of the first part of my presentation, so instead of the Vu-graphs that you have, I would like to start a little further down. The one that starts with Staff review.

I apologize, gentlemen. I gave you the page instead of the flimsy.

The reason I want to talk on the Staff review a

little bit was to characterize a little better where we are. There are two pieces to what we call the Watts Bar independent design verfication, or IDVP review. The first part was the Black & Veatch and review, which was a vertical slice on the auxiliary feedwater system only.

MR. KERR: Excuse me. Tell me what a vertical slice is, please.

MS. ADENSAM: As I understand it, and I have not been personally involved in one of these, it is the detailed review starting with what the original concept and commitments of a particular system were supposed to be and following that through the design, and I think in the case of Black & Veatch, they went in and looked at how it was being constructed, looking at process controls and how the licensee is developing procedures and construction specifications to implement it.

MR. KERR: Thank you.

MS. ADENSAM: But getting into the details of different disciplines of one system, which was what I understand Black & Veatch did, rather than looking at a programmatic review, which I think is usually referred to as a horizontal slice instead of vertical slice.

With the Black & Veatch review, the Staff, back as early as '82, had decided when TVA came in and made the proposal that Black & Veatch was both a competent AE and independent of TVA, at least as far as we understood, and we

proceeded then with the Black & Veatch reports, which were filed both in April of '83 and February of '84, to conduct an audit of those reports, in which we reviewed 97 out of the 428 findings.

Of those, we are satisfied that Black & Veatch did what they said they were going to do, that the findings are reasonable and that there is one of the three which is still under review and has a bearing on Bulletin 79-02, which has to do with baseplat flexibility and anchor bolts.

MR. MOELLER: Excuse me. On the ones that you reviewed, were those the ones that you considered more important, or did you randomly select them?

MS. ADENSAM: Actually, they were biased a little bit, Mr. Moeller. The first report that Black & Veatch came in with, they had a considerable number of unresolved items. They called them confirmed or open. But they hadn't settled with TVA what the proper resolution was, and the Staff basically sent them back to the drawing board and said, you know, this is too big a job for us, you guys iron it out and come back again.

That was the reason for the report back in February of '84 where they resolved all but the three that were still left open.

We reviewed out two-thirds -- I would say about half of our sample was from those unresolved ones in that year, and

then the other ones were more or less randomly selected, I believe, from the other ones.

The other half of this effort is TVA's effort, and this is a policy committee that Mr. Reed referred to. TVA had a policy committee that was put together to oversee this whole effort from the very beginning. The policy committee was responsible for doing the generic review, for taking the Black & Veatch findings and considering the implications for the rest of the plant, of the other safety systems that might be impacted by these findings.

Initially, in the '83-84 time frame, Region II people were involved in looking at this work. They did do an inspection of the TVA's generic effort, which clearly followed the Black & Veatch work. However, with some of the concerns that have been raised recently with regard to the report, with regard to the closeout of the findings and the fact that TVA was the one doing this generic review, the Staff has determined that we need to do some further work, and we have established a Dedicated Review Group to continue with this effort.

This is a group composed of representatives both from Region II and NRR under my direction and is made up of a cross-section of the engineering disciplines that we would consider necessary for the site.

[SLIDE]

The objectives of this review group are going to be to determine if TVA did a good job, basically, in addressing the findings of the Black & Veatch report, did they do an adequate job of addressing the generic applicability of findings and corrective actions on plant design and construction.

The other thing we have asked this review group to do is not only close out the IDVP review but to address the allegations that we have received regarding Black & Veatch, and I will get into that a little more later.

[Slide]

Basically, the steps that the Dedicated Review Group are going to take will be to review the Black & Veatch reports, more for familiarity, and to address the allegations rather than to rereview the findings, to look at the Policy Committee report and related documents. This includes a rather voluminous set of evaluations that were conducted by a Task Force.

The Policy Committee set up a task force to go into each finding and review each finding and evaluate its impact on the balance of the plant. The Policy Committee itself was a high-level management group --

MR. KERR: Excuse me. I am getting lost in committees now. There is a Black & Veatch report and there is a TVA Policy Committee report, and now there is another

committee that has been set up to review the work of these other two groups?

MS. ADENSAM: No, sir. Okay, let me take -- it does get confusing, Dr. Kerr. I'm sorry.

The Policy Committee was overseeing the whole effort, both interaction with Black & Veatch as well as establishing a TVA task force to report to them, a kind of a working group for them, to take the Black & Veatch work and actually do the job of evaluating those findings on the rest of the plant.

The Task Force report was the technical input to the Policy Committee and formed the basis for the Policy Committee report.

Does that help?

MR. KERR: Well, I thought you were now telling me waht some group was going to do with a combination of the B&V reports and the Policy Committee report, and I wasn't sure what that group was. Maybe I am off track.

MS. ADENSAM: Okay. I'm sorry.

This is an NRC Staff group. We are referring to them as the Dedicated Review Group, composed of individuals from both NRR and Region II, and our objective here is to develop the Staff's evaluation of the whole IDVP, not just the Black & Veatch work, most of which we have done, but also look at the generic effort that TVA put in to determine how those

findings should be utilized and what the impact was on the rest of the facility.

MR. KERR: Okay. Now, when you say the rest of the facility, do you mean Watts Bar or all of the nuclear plants?

MS. ADENSAM: As it turns out, they did address some of their other facilities, but we are focusing here on Watts Bar.

MR. KERR: Thank you, ma'am.

MS. ADENSAM: Another set of documents that we are asking this group to look at, largely because of the allegations but partially as part of the Black & Veatch review, is this Nuclear Safety Review Staff, which is an independent TVA review staff that reports to their general manager and board of directors.

The Nuclear Safety Review Staff took all of the documents of Black & Veatch, the Policy Committee report and the work the Task Force had done, and did a review of it and reported and made recommendations to the TVA management as to what corrective actions should be taken.

So we are asking the Staff to look at those reports and related documents largely to address the allegations. We are currently scheduled to go down Monday for a site visit. We have given the Licensee a request for additional documentation and certain questions that we need to have answered, and we are going to go down there on a site audit to try and look at

these other documents and get resolution to our concerns so we can close this issue out.

This will be followed by preparation of an SER, which will close out the Black & Veatch review and, hopefully, the allegations related to Black & Veatch.

[Slide]

I would like to go on a little bit into the allegations. This isn't terribly definitive. The letter that you have been given, dated May 16, 1985, to TVA has two enclosures, 1 and 3, which outlines the allegations that we have received that we have given to TVA. Most of these are very general. They are very similar to issues that have been identified in NSRS reports, in nonconformance reports and in construction deficiency reports, other TVA memoranda and some NRC inspection reports and meeting summaries.

MR. MOELLER: Excuse me. You say these are allegations that you have received from whom?

MS. ADENSAM: We don't know.

MR. MOELLER: Okay. These were then in response to this poster that Mr. Reed passed around?

MS. ADENSAM: No, sir. The poster is in response, I think, to some extent to the allegations. The allegations came first.

MR. MOELLER: Oh, and they were anonymously submitted to NRC?

MS. ADENSAM: Yes, sir. Some were by anonymous Telecon, and some were simply shipped to us in the mail.

MR. KERR: Well, this may not be the time to interrupt you, but in looking at Enclosure 1, for example, and I just picked No. 5, which says several concerns have been raised regarding the Independent Design Verification Program, and it says a concern regarding the closeout of about 500 items, a concern that only one construction specification was looked at by Black & Veatch in their review, a concern that Black & Veatch did not know how the plant was actually built, and a concern that Black & Veatch only compared the system's design and construction to its design criteria, not to the underlying regulatory criteria.

Now, if I got something like this from the NRC, I wouldn't have any idea what to do with it. What is the Licensee supposed to do with this?

MS. ADENSAM: Well, Enclosure 2 includes some questions that we asked regarding those issues, and that is not the total answer, clearly. I guess the Staff is asking Licensee to look at those and to address them as best they can to see if there is anything there that they should look at and reconsider.

I agree with you they are very general. This is one of the problems the Staff has had in dealing with these, also.

MR. KERR: So the Staff does not attempt to make any

judgment as to whether something is irrelevant; it just ships everything off to TVA and says, tell us whether these things are irrelevant, and if they are relevant, what you are going to do about them? I'm not trying to be critical. I just --

MS. ADENSAM: In this particular case, in this particular case we felt that it was appropriate to advise TVA of what these concerns were and ask them to address them as best they could. Since these allegations were received by the Staff, the Staff is also evaluating them, and we will also reach a determination as to whether we consider them irrelevant or not.

I think our concern was at the time we received them they were general enough, and our understanding, frankly, of the details that TVA had gone into with regard to their Black & Veatch review and the generic implication review that they had done was such that we felt they were in a better position to address these general concerns than we were. At least we could get the benefit of what they knew.

MR. KERR: Then there is one that says a concern has been expressed that FSAR Amendment No. 53, TVA lessened the experience requirements for the plant manager. The reason I raise the issue is because my impression is that there are some important problems involved with Watts Bar. I may be wrong about this. And if somebody is spending time on this sort of thing, it would seem to me that the talent might be

better concentrated on important problems.

I'm fishing because I don't know what the important problems are, but it does not seem to me that what I see here has been well enough defined that I would know whethr a problem exists or not. Maybe that is the position in which you find yourself: you don't know whether there are problems or not.

MS. ADENSAM: I think that is a pretty good characterization, yes, sir. We have determined that we are going to look into it and see if there is anything there. We have found in some instances, for instance, on the embedded plate issue, that there have been all sorts of inspections and inspection reports, there are many documents within TVA in terms of nonconformance reports and construction deficiency reports, and there has been a lot of inspection effort put in to reviewing and evaluating some of the problems related to this issue. We find that there are still some loose ends with regard to that.

One of the better-defined allegations had to do with lack of material control, and it was weld filler material control.

MR. KERR: Well, what you are talking about sounds to me like important problems, but if somebody sent me something that said a concern has been expressed that FSAR Amendment No. 53 lessened the experience requirement for the

plant manager, why include something like that in a list of what may be important problems?

MS. ADENSAM: We just elected to send them all.

MR. KERR: Okay.

MR. ETHERINGTON: Do you know how many contributors there were to these allegations, how many different parties?

MS. ADENSAM: I truly don't know, Mr. Etherington. I do not know whether it was one or a dozen.

MR. ETHERINGTON: Was it a large number of --

MS. ADENSAM: I don't know.

MR. REED: I have noted from this list that you have now up on the screen, and having read the INPO exit interview evaluation report, a strange paralellism between the items. Now, do you have any explanation for that?

MS. ADENSAM: Absolutely none. I noticed the same parallelism.

MR. REED: You have to deduce either there is a lot to it or it is the same people who are being interviewed by INPO and writing the allegations.

MS. ADENSAM: Clearly we can speculate that --

MR. REED: I can go from one extreme, there is a conspiracy, or there is a lot of meat here. Did you come to that conclusion?

MS. ADENSAM: I can speculate that that might be the case, yes, sir.

MR. REED: Thank you.

MS. ADENSAM: Mr. Ebersole?

MR. EBERSOLE: Yes. Ms. Adensam, in connection with all these things, the plant should be somewhat forgiving of small departures from rigid adherence to the tenth decimal point in regulatory procedures. I tried to make a case in point of the hangers. We can't have a plant if it has a hanger failure -- and this is, of course, true of Diablo Canyon. We have serious results if one hanger fails.

When you are looking at this, do you examine the delicacy of seriousness of each of these topics in this context?

MS. ADENSAM: Yes, sir. I was going to go in my next slide to a little bit of how we are approaching these. Actually, the type of information we have is -- the INPO field notes that you have available to you go into a great deal more detail on these than what we have in terms of the "allegations."

To some of them, we have not much more than what you see on this slide, particularly on the Black & Veatch ones, and there is just not that much information.

[Slide]

So the approach that we have taken is that we have assigned responsibility for these. We have tried to -- Enclosure 3, for instance. We have tried to group. It's a

great deal similar type of concerns. We tried to put them in categories ourselves and group them as an allegation, and we assigned responsible organizations within NRC. You know, some of them are going to the Regions, some to NRR and some to I&E, and we are in the process of screening them for safety significance and the impact on licensing.

We are going to review the responses to the letters that we sent TVA, and the May 30th letter, which you do not have, was a set of questions with regard to Black & Veatch requesting certain documents from TVA. We are hoping to use that to help us resolve the Black & Veatch concerns.

We review and evaluate these issues, and we are preparing either an SER or an inspection report. There are some which we feel we do have to address prior to licensing, and I think the Black & Veatch IDVP and the allegations associated with it are one of them.

But this is generally the approach that we have taken, and I agree with you that these -- you know, Mr. Kerr, these are very general in a lot of cases, and if the licensee says, gee, I don't know what to do with them, they are sharing our concerns. We would love to have had a lot more specificity. It helps us considerably in tracking these down because if someone does have a concern and they give us a general allegation that says I've got a problem with the hangers -- and there is a real one out there -- it makes it

extremely difficult for us or the licensee to resrond to that, which hanger. You know, which type of hanger.

So that is the approach that we are taking.

Where we are right now is we are trying, frankly, to establish a schedule, and we have had some difficulty in that regard because TVA has not yet given us their schedule.

As far as the INPO field notes are concerned, I will apologize. I was not really prepared to discuss those. We have recognized the similarity that Mr. Reed says he recognizes. We are considering how we should be evaluating them. We have not as yet released the notes because of sensitivity of INPO documents, as you well know.

I have prepared a letter to the Licensee advising them that we were going to release them and giving them an opportunity to comment prior to doing so. If TVA has released this document already, there is no need for me to send that letter, I would assume.

MR. REED: Well, we are almost out of time. You are welcome to say anything that you want further if you have any, but I am trying to fish for what do we do now. It seems to me what I am hearing from you is that the Staff is well on top of this, as our 1982 letter asked, and it bothers me that there is committee after committee. You know, the more committees you have got, nobody is responsible for anything sometimes.

Maybe we ought to clean house and put just one man

in charge who is the qualified person and shape the whole thing up. But there is committee after committee that has been formed here, and still we have got allegations and the INPO report and murky waters, and I just don't know what to think.

I guess what you are doing, it seems to me, is to go down there with your point of view and your objectivity and try to sort this thing out as the way to go, and all we have got to do is -- we have listened, and we can make no judgments at this point in time, I think.

How do other people feel?

MR. REMICK: I agree in general with what you said, Glenn. I don't think we contribute to the process at this point. We have asked to be kept informed. We have been informed. It appears that the Staff is doing what we expect the Staff to do, and certainly we are not the ones to investigate allegations and so forth.

So I think that our further action is to continue to watch it as it develops.

MR. EBERSOLE: I agree with that. I think this extreme ventilation that the system is being exposed to is a good thing, and I can't believe that at the culmination of this, we will not have found a good job.

MR. KERR: What does the Staff need from us at this point, if anything?

MS. ADENSAM: I don't know, to be honest with you. My understanding is that this whole meeting with the Committee was precipitated by the Committee.

MR. WARD: Yes. You are not looking for anything from us.

MR. KERR: Oh, you don't need a note saying you have satisfied the requirements in our letter as far as we are concerned?

MR. SIESS: How about just an admission that we have been kept informed?

MR. WARD: We have certainly been kept informed. It seems to me the next step would be to review your conclusions after you have completed your review. What sort of schedule are you on for that?

MS. ADENSAM: We are trying to progress at a reasonable pace, but as I said, we are having difficulty with schedule because TVA has not given us their schedule for a fuel load date. That has been in a state of flux in the last few weeks, and as a consequence, like I said, the review group is going down Monday of next week to try and close out all their open concerns related to the review effort they have been putting in of reviewing the TVA documents on the IDVP.

MR. SIESS: There will be a supplement to the SER?

MS. ADENSAM: Oh, yes, sir, because we have not yet addressed in our SER that point that was in the Committee's

letter. In Supplement 1, we discussed that they were going to do the Independent Design Verification, but we did not present our evaluation of the results, and that would have been scheduled -- you know, allegations aside, we would have been issuing a supplement for that, and we have not yet issued that supplement. So clearly, there would be one.

MR. WARD: Okay. Well, it seems to me, then, that the next step as far as the Committee is concerned is to review the SER when there is a final draft or at some appropriate time related to that, and whether we want to do that with another subcommittee meeting first or just with the Full Committee, we can wait and determine at a later date.

MR. WYLIE: Could I ask just a question? From what you presented, you are going to concentrate on reviewing the Black & Veatch reports and the committee reports that TVA had in regard to that review, and then go through all of that and then prepare your SER. Are you going to attempt to dig in to the answers of each of these allegations?

MS. ADENSAM: Those that apply to the Black & Veatch report, yes.

MR. WYLIE: Well, they are much broader than that.

MS. ADENSAM: Yes. In what sense, Mr. Wylie? I'm not sure I understand.

MR. WYLIE: Well, I went through that list just a moment ago. They seem to be shooting from the hip, so to

speak.

MR. WARD: You mean the allegations extend beyond the Black & Veatch report?

MS. ADENSAM: Oh, no, sir. Those are other allegations. The allegations that I am referring to --

MR. WYLIE: You are not going to dig into these at all?

MS. ADENSAM: Oh, yes, sir, but not as part of the Black & Veatch review.

MR. WYLIE: Well, that is my question. This committee that is going down is not going to do that; is that correct?

MS. ADENSAM: There are allegations which we see as being part of the Black & Veatch review and which we are asking the review group to look at, and those are the ones here. The concerns on Black & Veatch. These were allegations. The allegation was that they closed out 500 items, there was only one left, and NRR hadn't looked at it.

MR. WYLIE: Yes, but this was limited to vertical slice.

MS. ADENSAM: Well, it went a little beyond that, i a sense, because there was a concern about how much depth Black & Veatch had gone into in the construction --

MR. WYLIE: Well, what I'm fishing for is how does the Staff prepare to review the answers to all these

allegations? Are you going to look at all of them? Okay. He says yes. Would that be part of the report that we would get back?

MS. ADENSAM: It might be a part of the same Safety Evaluation Report. It might be another supplement. Some of them may be closed out in the inspection reports.

MR. WYLIE: Well, I would be interested in how these things are answered.

MR. SIESS: I think it took four supplements to cover all the allegations on Diablo Canyon. At least four. And one of them was that thick (indicating). But I think you can count on a supplement SER that will cover every allegation the Staff has received.

MS. ADENSAM: Or an inspection report. Some of them may be closed out in inspection reports.

MR. SIESS: Is this being handled any differently than, say, Diablo or Comanche Peak?

MS. ADENSAM: I don't really know how Diablo and Comanche Peak are being handled. It is being handled differently to some extent because of the Black & Veatch involvement and the fact that we had not yet closed out our review of that report.

MR. SIESS: Is there some group in the Staff now that handles allegations on some uniform basis, or is it just ad hoc?

MS. ADENSAM: Well, there is an Action Office delegated, depending on what the allegation is, and that Action Office has responsibility for resolving the allegation and reporting back to the alleger if that individual can be identified.

MR. SIESS: Well, Action Office. You mean Office of Nuclear Regulatory Research or Office of something like that?

MS. ADENSAM: Yes, sir. And as a matter of fact --

MR. SIESS: Well, now I am really confused because for Diablo Canyon there was an allegation review team. Some formal thing was set up. There was something done for Waterford where there were large numbers. There is something being done for Comanche Peak. Are all of these different? Does NRC still have no systematic way of looking at allegations? Is it just left up to you and the individual reviewers?

MS. ADENSAM: I would say those three cases were not the norm, probably because of the large number of allegations.

MR. SIESS: They were not the norm based on history, but I don't have too much difficulty saying they might be the norm for the future. I mean we are up to four now. I don't know.

MS. ADENSAM: I would venture to say that if we wind up with, you know, 1600 or 2000 allegations on Watts Bar, that there may very well be a team to look at Watts Bar allegations

also.

MR. SIESS: So you think it is a function of the number?

MS. ADENSAM: I think that had a lot to do with it.

MR. REED: Or is it a function of origin? I think the allegations here at Watts Bar are going to be mostly employee-type allegations rather than public, aren't they?

MR. SIESS: No, they were employee allegations at Diablo Canyon.

MR. REED: But here there are mostly employee.

MS. ADENSAM: I have seen many that evidently were views expressed by people within TVA. Whether we got them from an employee or not, I just don't know.

MR. REED: I was thinking of that parallelism that seems to be showing up.

MS. ADENSAM: I agree it points that way. I can't argue with you.

MR. REED: Well, are we through, essentially?

MR. EBERSOLE: Ms. Adensam, this system that TVA ha got about a third party handling the allegations, is this unique in the business? I thought it was going to be certainly very troublesome but a very sweeping, overall control of the allegation problem. That is one way to put it behind them.

MS. ADENSAM: I do not know that there is anyone

else who has hired an independent interviewer. There have been other utilities which have had brought in parties to do assessments of allegations, but I don't know of anybody who has proposed to interview everybody on site as TVA is doing.

MR. EBERSOLE: Well, it sort of struck me that this should lay the allegation problem at rest because it is an open invitation, independent, and apparently carries no jeopardy to the individual who carries an allegation.

MR. SIESS: I think you are an optimist. This company that is doing it for TVA, they haven't done it for any other nuclear plant?

MS. ADENSAM: I don't know that. Maybe TVA can answer.

MR. PIERCE: I can answer that. To my knowledge, they have done Waterford and Wolf Creek, and there may be others. I know they have done these two.

MR. SIESS: Thank you.

MR. EBERSOLE: So there is nothing new about this.

MR. WARD: I think, Mr. Reed, we can conclude, then?

MR. REED: Well, thanks very much, Elinor. I know on June 26 you didn't have a lot of input, and I think the Subcommittee felt the waters were fairly clear at that time, and things have certainly happened in the last two weeks to change things, and it looks like you have got a big task ahead of you and we thank you, and we think that you will get back

to us with pretty straight information.

MS. ADENSAM: You might be interested, Mr. Reed. There was an employee response team evaluation. There is an inspection effort going on next week at the site. We will be talking to both the independent reviewer and the Nuclear Safety Review Staff.

MR. EBERSOLE: One final thing. We have had some noise from on the Hill, you know, Henry Meyers. Are you-all taking his observations into account and intend to respond to them?

MS. ADENSAM: Yes, sir, we are.

MR. EBERSOLE: Thank you.

MR. WARD: Thank you very much.

[The portion of the meeting from 9:35 a.m. to 11:00 a.m. was not reported.]

[11:00 a.m.]

MR. WARD: We are now at Agenda Item XI, the GESSAR II. Dr. Okrent?

MR. OKRENT: Well, you will recall that we had a meeting last month on GESSAR II that got into parts of the subject of containment capability and parts of the subject of hydrogen, but we ran out of time before we got into sufficient depth. And so, I suggested that for this meeting we try to provide greater depth for the committee's information in these areas. And I hope that that is the way the NRC Staff and GE will go, but time will tell.

Anyway, I think we should just proceed with the agenda and stay to the two hours.

MR. VILLA: Before we begin, Mr. Ward, I would like to just make two points regarding the review. We are very interested in wrapping up this review, or at least seeing a closure to this review, and so we are encouraging our people to try to do everything they can to cover each one of the agenda items with brief responses and, of course, hopefully, they will be as complete as the committee would like them to be.

So we will begin with a comparison of the contrast between GESSAR MARK III containment and other existing MARK III containments. Mr. Vij will begin.

[Slide.]

MR VIJ: Good morning, ladies and gentlemen. My name is R. Vij with General Electric. I think when I read the transcripts of the last meetings you were interested in different parameters of MARK III containments, and the first slide addresses that issue.

The first column is plant names, then containment type, design pressures, ultimate pressure capability calculated by the different AE's, dome configuration, containment diameter and containment fill at the base, and the fill is the concrete which is poured between the freestanding steel containment and the shield building.

I will just go through one row over here and then the numbers are before you. GESSAR II, which has a freestanding steel containment backed by a concrete shield building and with a five-foot annular space between that, which is filled up to the water line, four or five feet above the water line for GESSAR II.

The design pressure is 15 psig. The ultimate pressure capability which was calculated and presented to you last time is 85 psig. Dome configuration is torispherical. Containment diameter is 120 feet. And there is a fill of five feet thick between the freestanding steel containment and the shield building.

Grand Gulf is not a double barrier; it's a lined, reinforced concrete containment. Design pressure, ultimate

pressure capability calculated by Bechtel, hemispherical, 123 foot diameter. This gives you, by the way, the sizes of the plants -- 238, 251, 238 and these are smaller units, 218 and 218.

Now, one thing which I want to point out is that GE developed the methodology that a given certain energy dump after a break inside the drywell, then how do you calculate the pressure and temperature transients, the design pressures and temperatures, on the drywell and the containment.

Beyond that, we have suggested the various dimensions of the containment for different sizes in a drawing we call reference containment definition, and we give it to the customers and architect engineers. But the containment -- the actual size of the containment is governed by the way the architect engineer lays his equipment inside the containment, and then once you envelope the configuration, then you can calculate your design pressures and temperatures and design the containment and the drywell to meet the goals and the NRC requirements.

So after the contract, each project, each AE and the customer are individual, independent entities, and they develop their own design. GE does not give them the detailed structural design of every containment to be followed. And that will account for some differences in diameters; that will account for some calculated -- difference in calculated values

over here [indicating].

MR. OKRENT: Well, with GESSAR II, would there be any freedom given to the licensee or the architect engineer with regard to anything that relates to the containment building?

MR. VIJ: Well, for GESSAR II we will not give any freedom over here, we will not give any freedom over here [indicating], and the detailed structural design is also given. Actually, this also should be the same. I don't see any reason for an architect engineer to change the GESSAR II detail design if it's already done.

MR. OKRENT: Now, how about inside the building? It's my recollection that there are differences between Grand Gulf and River Bend with regard to the actual containment layout below the vessel; whether there's a concrete or a cavity or so forth.

MR. VIJ: Below the vessel is the pedestal, and the dimensions from the top of the basemat and the underside of the vessel is dictated by General Electric from the viewpoint of controlling the under vessel controller drive housings. So they will have to keep that space as minimum.

MR. OKRENT: As a minimum. But they could change it.

MR. VIJ: They can make it more; we don't mind that. But then they have to do their own structural design

for that configuration. But the minimum space, they have to maintain. I think it's about 19 feet.

MR. OKRENT: But then -- but there may be variations. That's an unspecified part of the containment, you're saying, except for the minimum --

MR. VIJ: Yes. One minute. Let me put this sketch up here.

MR. OKRENT: I'm trying to find out is it really a standard containment and --

MR. VIJ: GESSAR II, yes, it is a standard containment. But every AE, every customer has some flexibility in their own design. Previously done designs. In future, if some customer buys my GESSAR II, I will not give his AE the flexibility to change.

MR. OKRENT: Well, what flexibility -- I thought you just said there will be some flexibility in GESSAR II.

MR. VIJ: No. I said there was flexibility in the different AEs' designs in the past. But tomorrow, if somebody buys a BWR 6 MARK III nuclear steam supply system, I will again give the AEs flexibilities, so long as they meet my minimum requirements. But if some customer buys GESSAR II design from me, then I will not give him the flexibility because I have the design developed with it.

MR. OKRENT: Does he have any flexibility in the containment design if he buys GESSAR II?

MR. VIJ: If I am giving him the detailed design why should he have any flexibility?

MR. OKRENT: You can just say no.

MR. VIJ: I will say no, then.

MR. EBERSOLE: Can I ask you a question? In consideration of the severe accident policy there's a fundamental go/no-go issue that comes up, and that has to do with -- and I don't know how this is going to come out -- but it seems to be that we get to a point where we have to admit that the core has melted, and that the vessel in turn will be penetrated.

And this gets to the next and controversial stage. When the primary vessel is molten, or the bottom drops out -- in your case it's the control rod drive tubes -- what does it encounter as it proceeds downward? Does it encounter water, inevitably, because that's the way you design it, or does it hit dry concrete and start doing other things? And what's your rationale and what have you done here?

MR. VILLA: Excuse me, could we defer that question until the agenda item comes up?

MR. EBERSOLE: Sure, if it's coming up anyway. I just want to know how you got to this root sort of question.

I think I would have to say now we don't yet know whether we're going to consider that you must always argue there's going to be water there somewhere, or there will never

be water after the melt. And this gets into that area.

MR. VIJ: The last sheet is on the core meltdown. We will come to that.

MR. MARK: Could you say easily what accounts for the difference in pressure capability of Perry and GESSAR II? One has thinner steel or thinner concrete, or what?

MR. VIJ: Perry also uses a freestanding steel containment, and this is also freestanding steel containment. So the question of concrete will not come in. It will be the detailed configuration of the thickness of the shape of the dome, differences between Perry and GESSAR II. Or it could be even difference in the calculational methods.

MR. MARK: Well, I was wondering if it was calculational method or design. Were they done by different people, the calculations? Then you'd get different answers.

MR. VIJ: If you look at the differences, I think they're in the same ballbark. When Dr. Bagal presented this number to you last time, he said that he is using some allowable stresses -- not allowable -- some stresses between yield and the ultimate. Somebody might have used a somewhat different one. Or the containment configuration might be slightly different.

MR. MARK: Well, it's the same -- same diameter, same configuration?

MR. VIJ: But diameter and height doesn't determine

the stresses; it is what thicknesses of steel I am putting in there. And then when I go to the head, whether it's two-center torispherical, three-center hemispherical --

MR. WARD: Well, he's asking you what are the differences. I mean, they are both designed. What are the differences? Is it the shape, the configuration, or is it the thickness?

MR. VIJ: Personally, I have not studied the detailed structural design of every bend and twist in the Perry containment.

MR. WARD: All right. He doesn't know, Carson.

MR. VIJ: You can pick up this number from the FSAR.

[Slide.]

MR. VIJ: The next issue to be addressed is fabrication flaws in the containment. What we did was, our fracture mechanics group performed this analysis. They took the maximum stress in the containment for the ultimate pressure capability and it is somewhere in the dome. And they used the lower bound fracture toughness properties of the material and when they picked this value, they looked at only three of them; the plate material of the containment dome, the weldment material and the heat-affected zone around the welds. And they picked out the most conservative value which will give you the smallest flaw. And it came out to be plate material.

The fracture analysis results showed that a potential crack which is three inches long and a half inch deep can be tolerated without unstable propagation to failure. As you can see, this half inch depth is about more than 25 percent of the wall thickness. And with all the controls on the welding procedures; pre-heating, multi pass to improve the impact properties of the weldings, we don't expect that the procedures will leave flaws more than about 10 percent of the wall thickness.

But then all the 100 percent, all the welds are 100 percent radiographed outside where a flaw up to a 2 percent depth of the wall can be detected with an instrument. So any flaw which is angular, which is more important from the containment filler viewpoint -- laminar flaws are not that important for the pressure. More than 2 percent should be detected radiographically. But unstable flaws is much, much more than that.

So at least on this basis, we think that undetected flaws should not compromise the calculated ultimate capability of the containment which was presented to you last time, which is about 85 psig.

MR. EBERSOLE: May I ask a question? This is just considering the containment is a pure membrane, but unfortunately, there are beaucoup penetrations in it of all sorts which are, you know, discontinuities and other types of

vulnerability to pressure and temperature. Are you going to develop your argument considering not merely the membrane but also the penetrations?

MR. VIJ: I asked this question of Dr. Sam who is our expert on the fracture mechanics and he said that when you come to the fracture mechanics it is the tension in the membrane which is most important; compression is not, bending is not. And I took his answer. I am not a fracture mechanics expert.

You see, there's the tension. Even when we picked up the stresses in the dome, the 85 psig is close to the knuckle, and there is the combined stress. But he picked up the maximum tensile stress anywhere in the membrane to perform this fracture analysis.

MR. EBERSOLE: Well, you're still talking about the metal of the containment. Let's talk about the material interior to a given containment penetration, which is sometimes a variety of insulating materials or other -- a great variety of materials going into the penetration proper.

Is it in your domain to talk about the viability of these under high temperatures and pressures? These are the penetrations within the penetration frame itself. This is the guts of the penetration, to be blunt.

MR. VIJ: I think the penetrations question is being discussed by Deborah later, but that's from the viewpoint of

temperature effects on it. We have not considered those portions in the fracture analysis.

MR. EBERSOLE: Do you control in high detail the character of those penetration guts?

MR. VIJ: Again, I will not be able to answer that question in detail. But when I come to the containment filler mode, I will try to put things in perspective that hey, what it really means in real life.

MR. EBERSOLE: Okay.

MR. SHEWMON: Do you know what the yield strength of this material is that you're quoting an analysis for?

MR. VIJ: Yield strength I think is 46 psi.

MR. SHEWMON: You will, around the penetrations from the welding operation, have stresses at the yield. So I suspect you can go up to the yield, which is not very high in this stuff, and say that it won't crack if it's less than that flaw. So I don't think the penetrations are going to bother you much. He is already at yield from the welding --

MR. EBERSOLE: You're talking about the external frame around the penetration proper, and I'm getting to the internal aspect of the penetration.

MR. SHEWMON: Yes. On the other one, the external one, you were one question back. I'm too slow.

MR. EBERSOLE: By the way, this includes now the electrical penetrations in particular, such ones as the large

main coolant pump penetration, which is a very large penetration, which is subject to interesting phenomena if the main leads go into a short circuit mode. These penetrations tend to be the fuse of the electrical circuit, and so they must be protected with safety grade protection.

MR. OKRENT: Before you go on, is there an experience record that tells us what the chances of missing flaws of that size is?

MR. VIJ: All the metallurgists I have talked to and the ISI people in GE, everyone felt that in carbon steel the welds are stronger than the base material. And there's no way that flaws of this size will be missed either in the inspections of base metals which are hard-rolled metal plates, or during welding where you have to have at least 10 or 12 passes in 1 3/4 inch thick plate; 10 or 12 weld passes, and then 100 percent radiography which will pick up 2 percent angular flaws.

MR. OKRENT: Well let's see. You say 2 percent. It's 2 percent of an inch and a half or something like this, right?

MR. VIJ: Right.

MR. OKRENT: So I'm wondering if you have a tight crack why you're so confident radiography will be that good.

MR. VIJ: Radiography, as I understand it, if the crack is vertical to the surface, angular to the surface, then

the penetrometer which is about 2 percent thickness -- and if they take that hole in the film below that, they should be able to pick up any flaw which is more than 2 percent of the thickness, in that direction.

MR. OKRENT: But again, if you have a tight crack -- in fact, in either direction -- I think radiography is sort of helpless, isn't it?

MR. SHEWMON: Do you weld this from both sides or one side?

MR. VIJ: Personally, I don't know. Maybe it's both sides. I do not know.

MR. SHEWMON: If you weld from one side it's just a matter of did you forget to get it in someplace. If you weld from both sides, then you pass over the outsides and leave your tight crack in the middle.

MR. VIJ: I'm sorry. I thought you were asking whether they radiograph from one side. The welding is from two sides. It's an inverted V groove from both sides.

MR. SHEWMON: So the original contact is very slight.

MR. VIJ: Right.

MR. SHEWMON: So it's very difficult to get a very tight crack there. It's either you fill in or you don't fill it in.

MR. OKRENT: Assuming the welding procedure is not

somehow --

MR. SHEWMON: Well, if you muck it in, then you don't fill it in, and that's the kind of a void that radiography is good for.

MR. OKRENT: Well, let me put my question a different way. In the radiation -- I'm sorry. In the inspection using radiography of the several freestanding containments that have been built like this, have they never found flaws larger than 10 percent of wall thickness? Is this what you're telling me? And corrected it?

MR. VIJ: Yes, right.

MR. OKRENT: They have never found something larger than 10 percent?

MR. VIJ: To my knowledge, that is true, because they will pick up anything more than 2 percent, and then the allowables are as per the ASME Code.

MR. OKRENT: But I'm just trying to get really a yes or no answer. In the buildings that have been built, radiography has never detected a flaw larger than 10 percent of wall thickness? They just haven't existed?

MR. VIJ: When you say never, again, I would have to be overall knowledge of everything that has gone on in the field, which I am not.

MR. OKRENT: Well, then, it seems to me it's relevant to know what the experience is, to know in the first

place whether -- if a welding procedure in fact has left flaws that large which were detected by radiography.

MR. VIJ: That's what my metallurgists have told me; that more than this they will be detected and they will be corrected also if they are not acceptable.

MR. OKRENT: Does the staff know whether in radiography they have found flaws larger than 10 percent of wall thickness, to be corrected? Or that this is just a non-experience?

MR. CHENG: I have not been following the experience of the containment welding so I don't have an answer to that.

MR. KNIGHT: I would have to presume that we don't have here -- and I doubt if anyone really has -- the experience base that you're looking for. One could only assume that it's quite possible that the welding procedures could, in fact, give you -- you know, you could have initially had flaws larger than that to be found, ground out and corrected. I guess I couldn't personally accept that that couldn't be true.

MR. OKRENT: Yes, well I'm sure that other structures of similar thickness, flaws much deeper and longer than shown in the first bullet there have, in fact, not only existed but escaped inspection in the past.

MR. KNIGHT: That may be true. And certainly, the premise that I think all of this discussion operates on and

which I believe is the point you're getting to is that you have adequate procedures in place, you have adequate inspection processes in place, and your system makes them work.

MR. OKRENT: Yes, but we are trying, in effect, in all of this to get a feeling for what the likelihood of a flaw that might run being pressed -- I mean, when you did your PTS you didn't assume there were no flaws, even though the pressure vessel was certainly built with a care I assume that at least matches this one. At least.

MR. EBERSOLE: Dave, is your point that if you found lots of these then you would realize this is the first and last pass that you go through to find them, and your jeopardy would be a lot higher than if you found very few indeed?

MR. OKRENT: I'm just trying to understand whether there is some risk, significant risk, of the vessel failing due to the presence of flaws larger than the one shown and perhaps at a lower pressure -- the larger the flaw, presumably the lower stress, to some extent, --

MR. EBERSOLE: So is what you're saying that this method of X-raying and measuring should be a confirmatory process rather than an initial survey, or make an initial finding of defects?

MR. VIJ: It happens to be confirmatory because there are many agencies looking at that. There are the

insurance people, then the AE's people who all of them are supposed to look at 100 percent of the test results of the radiography and then spot checks by the NRC.

MR. EBERSOLE: But they all look at the same test results.

MR. VIJ: That is true, of course.

MR. EBERSOLE: So this is the common denominator where if you miss, -- if you have lots of faults which you fixed, you may have missed one or two.

MR. OKRENT: Well, let's let the subject go for now.

MR. ETHERINGTON: I think in machinery if you do radiography for inspection, the frequent purpose is to reject equipment that's not sound. But in structures I think where you have radiography you expect occasionally to have to make repairs. I'm not speaking of these containments; I'm speaking more in general.

MR. OKRENT: Okay. Let's go on.

[Slide.]

MR. VIJ: Next, we get to dominant containment failure modes, and even at the cost of --

MR. VILLA: Excuse me. At this point I guess the staff may want to give their comments, or we will go on to the next point. It's up to you.

MR. OKRENT: Well, let me pose a question to the staff. We have heard a presentation of an estimated

calculated failure pressure, and there has been a discussion which says that it applies a zero chance that this failure pressure would be degraded due to the existence of flaws.

Can the staff comment on that? Do they buy that?

MR. ROSENTHAL: This is Jack Rosenthal, NRR. We have an extensive presentation and it's simply a matter now of do you wish the staff and GE to alternate, or shall we let GE proceed with their presentation, and then have the staff and its consultants.

MR. SCALETTI: I think we have some constraints here with personnel. If we could go on with the staff's presentation with regard to design flaws and get that out of the way.

MR. KLECKER: I am Ray Klecker from Division of Engineering, NRR. We have made similar calculations to those that were referred to by General Electric, and Brookhaven National Lab has also made similar calculations which we will let them speak to themselves later.

We used a different analytical model, presumably, than what they did. However, we come up with answers that are very close to what they have. Quite frankly, we would say you could tolerate roughly a two-inch long throughwall crack or its equivalent crack area if you allow it to go a little longer.

So actually, the number you saw up there was a

little more conservative than what we did. We used a little higher toughness I think probably than they did, in our assumptions.

However, I think the bottom line is that any crack that may pre-exist, like as big as we're calculating here, we would conclude should be found during an inspection; that you do have enough leeway there, even if we put a margin of 2 or 3 on the crack size for uncertainties.

MR. OKRENT: My problem is with the word "should." I agree it should, but I'm trying to know whether we have some semi-quantitative way of estimating the chance that such a flaw, or a flaw that can run, exists. It is not through or you would get a leak when you do your first original containment --

MR. EBERSOLE: Let me ask about that question. Don't they, in fact, paint these steel vessels with such liberal coats of paint that you could have a crack and you'd never see the leak, with a leak test?

MR. KLECKER: Well normally, the leak test is done prior to the finished painting.

MR. EBERSOLE: I'm talking about over the long life of the plant.

MR. KLECKER: Well, if a crack this big existed from day one, I'm sure it would be found.

MR. EBERSOLE: Oh, sure. But if it were to come in

later.

MR. KLECKER: We don't really know of any mechanism that would cause a crack to grow later because the containment is supporting its own weight essentially.

MR. EBERSOLE: It's not a dynamic --

MR. KLECKER: That's correct.

MR. WARD: Well, I guess back to Dr. Okrent's question, Ray, when you said should be found, do you mean that there's a high probability it would be found in normal inspections?

MR. KLECKER: I would say a pretty high probability because looking at the requirements of the ASME Code and so forth, the allowable crack sizes are considerably smaller than what we are all calculating to be close to the critical crack size. By critical I mean you're reaching a level of toughness where even though you may have stable tearing for a while, it would tear and eventually reach instability.

So I think there is a margin there sufficient. Unfortunately, none of us inspect these vessels, so I can't speak personally as to the chances of missing something.

MR. WARD: Well, wait a minute. I know you don't personally inspect them, but do you expect that the Code methods for inspection that are required can, with high probability, be expected to detect flaws like this?

MR. KLECKER: I would say yes as long as all of the

welds are given an inspection and the plates are given at least a partial inspection. But the chances of having these cracks in the plate, which are rolled plates, are quite minimal. But they should be detected long before you get to the field.

MR. WARD: Well, you seem to be unusually cautious about your expectation that the Code will be followed. I mean, we depend on that in a lot of places; not just here. We don't depend on people from the NRC checking every detail of a plant.

MR. CHENG: Perhaps I can amplify on the Code requirement. A gentleman from GE mentioned that when you do the radiography you've got to have it in 2 percent, 2T sensitivity. That's what he's talking about. You have 2 percent of the wall thickness, you have to put the penetrometer over there to see how clearly you can see that X-ray when you shoot through. That's why he was talking about 2 percent penetrometer in the sensitivity.

However, if you go to the Section 3, the metal containment, when you do the radiography, the acceptance standards is in terms of the length, the length of the indications. And based on the containment wall thickness we're talking about here, say a range from an inch and a half to an inch and three-quarters, the allowable deviant indications length based on the Section 3 is about half an

inch, or six-tenths of an inch long.

Now, we're talking about a critical crack size or throughwall that has a length of two to three inches; if we are talking about a partial throughwall, the length if it's 50 percent, say a 50 percent throughwall, that length becomes about twice, four or five inches long instead of 2 or 3 inches we're talking about.

If you say go further, 25 percent throughwall crack, that critical length becomes roughly double than the original one, you're talking about 8 to 10 inches. Now, we know the Section 3 is published in the standards, it's a standard, and when they require people to find something back in the half an inch, or the quarter inch length, you would certain expect that if you have a six-inch or 10-inch long crack, you should be able to detect that one. That's the way we feel when we say you shouldn't expect them to pick up if they have a throughwall crack two or three inches long.

MR. VIJ: Well, I think one is answering the question literally, but the second is they calculated the ultimate capability at 85 psig, and suppose in real life it is 70 or 75; then what happens? And I think we will answer those questions in the next few minutes.

MR. ETHERINGTON: Why is the ultimate so high? It doesn't need to be that high, according to the code, does it?

MR. VIJ: No. The code actually is silent on the

ultimate capability. According to the code, you are designing it for a 15 psig level A and B, and then the CPM rule has 45 psig, and you stay at the low level, C.

Now you design to those ones, and then you can calculate as to what happens to be the ultimate capability. You are not designing it for ultimate capability.

MR. ETHERINGTON: Okay. Skip it.

MR. VIJ: This is the MARK III containment. Again, when I read the transcripts -- and excuse me for saying it --

MR. FOREMAN: Excuse me a minute. Dino wants to go ahead, I think, with some more of the NRC presentation.

MR. VIJ: Do you want me to stop here at this time?

MR. FOREMAN: Yes. Let's interrupt at this time.

MR. ROSENTHAL: Dr. Reich from Brookhaven will present his structural analysis and address the issue of flaws. But while he's walking up to the podium and strapping on his microphone, I would like to give some risk perspective.

We calculate late containment failure, assuming no flaws, of the order of 11 to 28 hours into the event. Presuming that flaws did exist, and hence the containment would fail earlier in time, once can look at the difference in consequences between an early containment failure and a late containment failure. We call that -- we would label those sequences 1-TL-3 and 1-TE-3.

We calculate about a factor of 3 in person-rem

between the early and late failures. The reason for the small difference is that in these pressurization sequences, slow overpressurization sequences, we have primary system retention, and we bubble the fission products through a pool.

The issue, then, in my mind, becomes a question of what fails first. If we fail the drywell or the pool before the wetwell -- and especially if you fail the pool -- then you will end up with a very different sequence in terms of consequences.

So what we'd like to do is to explore the structural strength of containment, make some comments about the strength of the drywell, consider flaws, and make some judgments about whether it will be the wetwell that fails first, which is relatively benign, or if we would end up with failures in the drywell or the pool, which denies us the fission product scrubbing.

Given that we're just worried about containment failure itself, the difference between the E-3 and the L-3 release -- it's only a factor of 3 in person-rem -- both sequences, we have zero early fatalities.

MR. EBERSOLE: Are you leading into the notion that one should design coordinated failure?

MR. ROSENTHAL: This containment will have venting procedures for wetwell venting, and you then have to judge with the operator, in fact vent the wetwell in a time period

of 11 to 28 hours, which seems ample time to take action. And if there was a real event, would we really go ahead and do that?

MR. EBERSOLE: Well, I was thinking about not deliberate venting, but even if you just tried to hold together and eventually did fail, you would have coordinated failure.

MR. ROSENTHAL: Well, what we want to assure is that the drywell and the pool stay intact, that the wetwell fails, hence relieving the pressure, which might endanger the pool or the drywell.

MR. REICH: BNL was asked by NRC to verify some of these structural failure analysis results, which were presented in Appendix G of the GESSAR report, and we were also asked to carry out some independent analyses for these structural members.

Specifically, our studies were concentrated on the three structure that we have listed on the slide. One is a torispherical steel containment, a drywell steel head, and the concrete roof slab. These three studies were basically a bunch of studies that were deterministic. And finally, we went back to the torispherical steel containment, and we did a reliability evaluation.

I am going to talk about all of these in more detail, and at the very end, I will address this question

about the fracture.

I guess most of you are familiar with what I am talking about.

[Slide.]

The steel vessel is right here (indicating). The steel head right here, and the roof slab right over here (indicating).

The roof slab is a very complex structure, which is stiffened by various walls on top, which are sort of -- there is a pool of water on there, and these walls provide a substantial amount of stiffness, as we will soon see.

Let me discuss a few moments some geometrical details pertaining to the torispherical steel containment. When we looked at this design -- and that is about a year and a half ago -- this was a two-center design, and I understand that recently there were changes made, and now this is a three-center design.

Now by two-center design, it's essentially that there are two radii describing the upper portion here. The spherical portion here is described by this large radius, which is an 80-foot radius, and the knuckle region here has this 11-foot. This R-2 is equal to 11 feet.

Also note that the wall itself is comprised of various thicknesses. The thickness varies from 1.25 inches to 1.75 inches, as you can see. I have listed that all along

here. In addition, the cylindrical section of the containment has various ring stiffeners along here. These go all around the containment, and on the top portion, you also have the crane girder which goes all the way around. Now these provide a substantial amount of stiffness.

Furthermore, on the very bottom portion, these 25 or so feet and some inches is reinforced with about nine feet of reinforced steel, and that's part of the reinforced biological shield wall. This portion down here (indicating) is substantially stiffer, does not move much.

Going now to the analysis that we performed, as I mentioned, some of these were checking what GE did, and some of these were independent analyses.

One of the first things we did was to look at the plastic limit analysis which GE performed, and essentially what they did was, they used the shield Drucker formulation, and this formulation essentially gives you plasticity at three hinges, which are A, B, and C on this figure.

Once that happens, this equation also gives you the associated pressure, the failure pressure.

Now this formulation, though, has some limitations. It does not give you displacements. It does not account for ring stiffness or the crane girder stiffening effect. It does not account for changes in thickness. And it was for that reason that we did our own evaluation, elastic/plastic finite

element analysis, to see really what happens in terms of plastic failure.

And just to give you a quick rundown on what we did there -- this figure did not come out the way I wanted it to -- but essentially we used an eight-node isoparametric element. We used 236 of these elements, and there were a total of 884 nodes. And in doing this, we really took particular care in modeling the knuckle region.

It did not come out clearly over here, but we took -- in the thickness region alone, we took three layers of elements, and then in this direction (indicating), we took very fine elements. These are finely spaced elements, so that we could capture all the local effects, including -- and we included in this model the ring stiffness, the crane girder, the bottom reinforcement, and the methodology that we used is a computer method called ENFAT, which we have checked out, which we know is a working code that computes direct results.

[Slide.]

This sort of gives you a very good idea of how plasticity really grows at certain pressures in the knuckle region. As we can see over here, at 34 psi, we have only a few elements which are plastic. At 38, we have the darker lines, so the plasticity has spread. Now at 42, plasticity has spread over the whole knuckle region.

A better understanding of what happens can be gotten

from a typical radial displacement versus pressure plot, and that is right in the knuckle region.

[Slide.]

And what you see here is essentially at about 34 psi. The slope starts decreasing gradually as plasticity really spreads into the region. At 42.7, the slope is virtually flat. This indicates a complete loss of structural stiffness. There is imminent plastic collapse.

However, in looking at this, you will note -- and I don't know if you can see it very clearly -- the displacement is quite large. You have about 12 inches of displacement. Now what this means is that in the formulation that we used previously, it is small displacement theory, and the same is true for the Drucker equation. That also uses small displacement theory.

These two actually -- the finite elements and the GE/Drucker solution -- come out quite close. The mean value for the GE was 38, and the upper value is probably closer to 42, so they're very close in number. However, the formulation does not fit the actual condition that's going on. You have a very large displacement.

If you look at the radii, which was 11 feet, R-2, you will notice that you have 10 inches of displacement for that. It is the wrong formulation to apply. So therefore, we decided to look at this with a large displacement formulation

[Slide.]

We used the same finite element grid that we used previously. However, this time the strain/displacement relationship is based on large deformation theory. Also the pressure loadings are deformation-dependent.

Now as you can see over here (indicating), this is the same node that we previously plotted for the small deformation theory. There is a stiffening effect in the knuckle region. Although the whole region is fully plastic, the collapse is really not imminent. It's not predicted even at 100 psi, and we have 12 inches of displacement at about 100 psi.

Now at 100 psi, we stopped the analysis. First of all, this type of analysis is not that cheap. Also the main reason, though, that we stopped was that in talking to our load analysis group on the Trevor-Pratt, it comes out that to reach 100 psi would take an awful long time, and basically the consequences of such --

MR. WARD: Can you tell us what that means?

MR. REICH: It would take about 14, or maybe longer, hours to reach 100 psi internal pressure.

Now the consequences for such a long event are probably very small. This is what we were told. So we stopped that.

[Slide.]

Now we did one more thing. We looked at buckling evaluation. We had very large shear stresses, deformations, in the knuckle region. They're high, and indeed a buckling mode is possible. However, this type of evaluation is very complex. The shell mode can undergo non-axisymmetric plastic deformations. So what you really need for the buckling, you need a 3-D analysis model, and we weren't going to go into that.

Instead of that, we checked it very roughly with an equation. And GE did actually the same thing here. They used an expression by Gillette. Now this expression gives a buckling failure pressure of about 61.3, which is lower than the 100 psi.

There are, however, some reservations that we have regarding this buckling failure pressure. These equations, if you look at them, they are based on analytical results and small tests, small torispherical head tests -- small tests.

Now we just saw that if you have large deformation, you get different results, and they probably will affect the buckling, too, so we don't know how accurate this 61.3 psi is, which is based on small deformation theory.

The other question that we have, even if buckling does occur, how certain are we that buckling will result in loss of internal pressure -- which we don't know. It may

buckle, and yet it may not crack.

Those are the four analyses that we did on the containment.

I now go to the drywell head.

[Slide.]

Let me talk about that a little bit. This consists of basically three sections, as you can see. There is a spherical top section, an elliptical, and then an ellipsoidal section, and then there's a cylindrical section right here (indicating).

There are two different thicknesses there, as you can also see. This cylindrical section and ellipsoidal section are the same. The spherical section has a bigger thickness.

We used again large deformation, non-linear finite element analysis.

This material is somewhat different than the other material. This is a 3-L-4 stainless steel, and it has a yield of about 30 ksi.

Again, if we look at how the plasticity spreads -

[Slide.]

-- at about 170, there is a plasticity formed at these inner elements. At about 190, we have a pretty good amount of plasticity formed throughout the region, the knuckle.

In looking at the displacements --

[Slide.]

-- I must say, this scale here is a little bit in the wrong direction. It is too large. It does not give us true meaning. It is really -- the displacements are not unbounded. It's not really flat, the curve, if you plot this correctly. And that indicates to us that the capacity, the failure capacity, is really higher than 190.

So looking at this, you would get failure probably first in the containment, rather than in this head (indicating), the drywell head.

What we noticed, what GE did, they looked over here at various levels. The had Level A capacity as 116. The buckling capacity for Level A was 75. This is all psig. Level C capacity was 160. There was no buckling done for Level C. It was not a licensing requirement. However, some elastic buckling of the spherical cap, an elastic buckling of the spherical cap was performed, and they used an expression from the literature. It can be found there. And this yielded a buckling pressure of 196 psig.

Now this is higher than the 160 capacity, which is defined by ASME. However, since the finite element shows that plastic collapse will occur above 190, buckling failure, I think, cannot be ruled out. The value of the 196 is for elastic buckling, basically of the spherical portion. It

could be lower if there was elastic/plastic behavior, which we do have, so I really don't know how correct that value is.

As mentioned, if you want to get it, really you need a three dimensional analysis. However, it is not recommended for this, because it's felt really that the vessel would fail before this.

[Slide.]

The next item we looked at is basically the roof slab, and as I mentioned to you, the roof slab is reinforced on top, the drywell roof, and the finite element model here was really very detailed.

The steel rebar reinforcement --

[Slide.]

-- and these were the Laird models. The rebar reinforcements were put into the correct position exactly as the drawings indicated. These drawings were given to us by GE. We used the same concrete properties, the seal properties, as supplied to us.

This was a concrete element with a shell element with nine nodes per element. In this total analysis, we used about 452 nodes.

There is, by the way, one point -- this Point A, which is on top of here (indicating) -- and at the point on the lower portions of the slab, I am going to show you the displacement of this relative to pressure.

[Slide.]

This slab is 48 inches thick, by the way. At about 40 psi, gauge cracking does happen to occur in the upper and lower portions, the upper nine inches and lower nine inches. Cracking is initiated there, and the forces are then transferred to the rebar. But as you can see, even at 125, it is not plastic that much yet. There is not that much plasticity in it. In fact, what we have is substantial cracking in the upper nine inches and the lower nine inches, but the middle 30 inches are virtually intact. They have very small cracks, and the concrete is -- I mean, and the steel is completely elastic.

So we feel, even at 125 g, this thing will stand up quite a lot more than that.

So to summarize these few analyses --

[Slide.]

-- torispherical steel containment, the Shield-Drucker plastic limit approach, gives you 38 psi mean value; small deformation elastic is pretty close to this. These two are virtually the same; however, as we mentioned, the theory is incorrect for large deformation. Two percent strain at 100; this is tensile strain at 100 psig. The slope is not yet flat, as I mentioned. Buckling failure by an equation which is based on small deformation gives you this (indicating).

The drywell head, large deformation, elastic/plastic finite element, section fully plastic at 190, slope not flat. And these are the analyses you get out of the code. GE did these. And if you consider only buckling of the spherical portion, elastic buckling, you get this type of a number (indicating).

[Slide.]

For the roof slab, even at 125, the top and bottom nine inches are cracked, the middle section virtually intact. The rebars are still elastic. GE did a very simplified complication at that time, and then just plugged those numbers in to check against the service and the factored allowables, and these are the numbers they showed for that (indicating).

MR. ROSENTHAL: At this time, I think in the interest of all the presentations, we would like to ask the committee's interest and get some feedback from you.

Dr. Reich also has some discussion of probablistic analysis or a sensitivity analysis to the input variables used in this calculation.

MR. REICH: That was the last item on this.

MR. OKRENT: Could I ask one question? If you get this extensive plastic deformation, does one know what will be the behavior of flaws that were subcritical before the plastic deformation? Are they changed in size as something that accompanies the plastic deformation, or do you expect them to

hold their original size? Does anyone know?

MR. KLECKER: This is Ray Klecker again from the Staff.

As you approach plasticity in what we would refer to as generally tough materials here, the small cracks will tend to blunt first before they begin to run. And what we have done in our analysis is to take it just to the point where they blunt but not extend by crack extension -- you know, unstable tearing.

So in this case, if you have a small crack, it would tend to open up, but the sharpness of the crack itself would tend to decrease.

MR. SIESS: Are the large plastic deformations confined to the knuckle region in the torispherical shell?

MR. REICH: Yes.

MR. SIESS: And there are no penetrations in that region?

MR. REICH: No, there aren't any.

MR. SIESS: So you have looked at any way in which the plastic deformations might affect the integrity of the penetrations?

MR. REICH: No. What we looked at -- and this is almost what we were going to talk about before -- we looked at a very rough calculation of what would happen if you had yielding. This is using an Irvin's type of equation. And we

looked at it two ways, and we used some conservative numbers. We used a yield, as you can see, of 36 ksi, and we used a J critical of 600. And this gives you, if you have sigma yield, if you're at yield, this assumes an elastic, perfectly plastic type of behavior.

Now this is crack depth, in other words, .123, and this would be crack length before you get critical crack growth (indicating), and these are the type of numbers you would get under that condition.

Furthermore, if you had a lower stress -- supposing I wasn't yet completely plastic, supposing I was up to 24 ksi and plastic is 36, this is the type that you would get (indicating). In other words, you could be very large -- if you are less than .3, you could have a very large crack and nothing would grow critically. And even coming down to .8, you would have a crack of about 11.8, and you would not as yet have critical crack growth.

Now I agree, these are very rough numbers and quick calculations. We could get these numbers more refined, and we might get them smaller. Even if we had an error here of, say, 50 or 60 percent, these numbers are very large.

Now I do have some experience, for instance, from bridge inspection, where people do inspections for cracks on bridges, and the inspectors do usually -- they can miss a crack of a quarter of an inch there, because they're much

rougher things to do over there -- a quarter inch depth. But those are not done as carefully as any nuclear type of inspection. Those are visual types of inspections.

MR. SIESS: I missed the early part of this. Is there any question about the capacity of the large opening in the containment building, the equipment hatch or personnel hatches? Are those all higher capacity than the shell?

MR. REICH: We did not check that out. The assumption is that they are reinforced. They may have a higher capacity.

MR. SIESS: Well, equipment hatches sometimes have a domed head on them, and if we are talking about two percent strain, there is ovaling. There is work being done elsewhere in the Staff on those things. I assume the Staff knows about it.

MR. ROSENTHAL: Let me just point out that that work is critical in some of our -- especially ice condenser plans where we've done -- sponsored some fine work.

The point that we're trying to make here, and in general, we think of the hatches as not structurally stronger than the containment, and what we are worried about is thermal effects on seals, which we can address later. You know, GE has a couple of comments, and so do we.

But I think the point we are trying to make is that times to containment failure are very long. If the wetwell

fails, it is a relatively benign release. We have a lot of margin between the wetwell failure and the drywell and the head of that drywell, should a crack exist.

So now if there's no flaw, then containment may never fail. If there is a flaw, containment may fail early. But if it fails early and that relieves the pressure, then you don't fail the drywell or the pool.

So we're rather confident that you will maintain your drywell integrity and your pool integrity.

MR. KERR: Is there going to be some discussion about what conclusion one can draw about containment failure from calculations of this sort?

If containment failure occurs in an actual containment, is it likely to occur as predicted by this analysis, or is it likely to occur at penetrations or some other spot?

You must have reached some conclusions. Have we already heard about this, or are we going to hear about it?

MR. ROSENTHAL: Well, we are far more concerned about deflagrations causing failures of penetration seals due to the thermal environment, and, in fact, we've modeled those.

MR. KERR: So you are convinced that there will not be distortions or pressures or forces on penetrations due to distortions in the structure that results in -- only deflagration is likely to cause penetration failures?

MR. ROSENTHAL: I think the argument is that that is so greater a dominant failure mode that we just haven't expended the resources to look at the structural strength.

MR. KERR: But you have expended enough resources to --

MR. ROSENTHAL: To convince us that that is the problem. The problem is the thermal environment on organic seal material.

MR. KERR: Okay. Suppose you have both the thermal environment and high pressure. Then what do you do?

You just assume that it occurred? I'm not sure what sort of thermal -- I mean, is this a flash kind of thing or a high temperature, because I haven't heard anything --

MR. ROSENTHAL: Sustained temperature. We have a presentation on it.

MR. KERR: Okay. I will wait.

MR. SIESS: You are familiar, then, with the work at Argonne, EG&G, and Sandia on containment leakage?

MR. ROSENTHAL: Yes, sir.

MR. SIESS: The pressure without temperature is worse for the drywell.

MR. ROSENTHAL: The work on the head is at Ames Laboratories, on the equipment hatch.

MR. OKRENT: Are there any other questions on this now.

[No response.]

MR. OKRENT: Let me make a request of the Staff that I am reminded of because Mr. Reich is here. There was a report published, NUREG-CR-4149, by Brookhaven -- Sherma, Wang, and Reich -- on ultimate pressure capacity, reinforced prestressed concrete containments -- in which they made some estimates for Zion and Indian Point. And at some point -- not today -- I would like to understand the import of -- if any, of the Brookhaven estimates on what the Staff thinks of the -- what Staff thinks the risk is at Indian Point and/or Zion.

They did estimate some lower numbers than other estimates and the different failure mechanisms.

MR. ROSENTHAL: The remainder of Dr. Reich's presentation is contained in the handout, and with your permission, we would like to move on, given the shortness of time.

MR. OKRENT: Yes, we should move on to the next.

MR. VILLA: We will begin with Item 4, which is the dominant containment failure modes.

[Slide.]

MR. VIJ: I will spend a couple of minutes to reinforce what Jack was saying all along here, that we are discussing an issue where the containment capability and the hydrogen generation are concerning what is important and what is not important.

I am sure all of you are familiar with this configuration. The reactor pressure vessel surrounded by the biological shield wall resting on top of the pedestal here. And outside this is the drywell, and the 1120 vents below the water surface contained within the drywell wall and the rear wall, and then the next boundary is the containment.

Okay. Let's make sure that we understand that this drywell, even though the word is "drywell," they are significantly different from MARK Is and MARK IIs. This drywell is a pressure boundary, a large pressure boundary for larger pressures only, only when you have a very, very sharp transient, because this volume and this volume (indicating) are in direct communication through these vents, and the only resistance you have to their free communication is about 5 psi or so.

So the drywell wall and the penetrations and the equipment hatch and the personnel lock, and the head, et cetera -- actually the head will be to a lesser extent, because it has the participation from the water, the hydrostatic pressure from the outside also -- the whole drywell boundary is not challenged at all in this low-pressure phenomena.

Am I making this point very, very clear? Then during any slow pressure buildup, either within the drywell or within the containment, that they drywell head, the drywell

ceiling, the drywell walls, the penetrations are not challenged at all beyond 5 or 6 psi.

And let's stop being plain analysts. Let us be practical engineers. A design which is designed for 30 psig for code-allowable stresses, subjected to zero pressure throughout its normal life and then later on subjected to 5 or 6 psig differential pressure, the failure of that boundary doesn't excite me at all, at least.

MR. EBERSOLE: What you're saying is that the drywell has essentially no duty at all to speak of for the slow depressurizations. It's in the large depressurization that you have some duty.

MR. VIJ: Yes. We discussed that one. You know, the only one is when you have a steamline break over here, and in a very small time, you get to about, I think, 30 psi. We calculated values much lower than that.

But when you have the hydrogen detonation of that and combustion --

MR. EBERSOLE: In the drywell head case, you've got water on top of the drywell head, haven't you?

MR. VIJ: Yes, right.

MR. EBERSOLE: Does the differential pressure include consideration of that hydrostatic head, or do you just ignore it in calculating the loads?

MR. VIJ: When you calculate the loads on the

drywell head, this is one of the loading conditions.

MR. EBERSOLE: Well, I'm talking about, you have an external pressure nominally, which is the hydrostatic head, right?

MR. VIJ: But that's only -- that's enclosed by these walls, right.

MR. EBERSOLE: I understand. But there is water pressure on top of that drywell head, isn't there?

MR. VIJ: That is true, sir.

MR. EBERSOLE: Do you neglect that when you calculate the internal pressure capability?

MR. VIJ: Do we neglect that? No. But in both cases, you consider it with water and without water. The head has the 30 psig capability without water.

MR. EBERSOLE: Without water. You treat it as if there's no water there.

MR. VIJ: That is right.

MR. EBERSOLE: Okay. Thank you. So that is another conservatism?

MR. VIJ: That's another conservatism, so far as the head is concerned with respect to these walls.

And now let's go to the containment, and here is a containment where, in this region, the thickness of the wall is 1.75 inches thick, and above that, it is less than that. When you go to the dome, it is again 1.75 inch. And this is

backed by eight-feet-thick concrete over which there is a 2.5 or 3-foot reinforced concrete design.

If you assume failure in this whole region, just look at the shape of this. It is a freestanding steel plate shell, and it is shell backed by eight-feet-thick concrete up to the water line, and above the water line, where will you expect any failure, if it has to happen? It has got to be above this place (indicating).

You can perform as many as finite element analyses as you want. It is a guaranteed fact of life that here it is stronger. The plate thickness is more and backed by eight feet of concrete.

MR. EBERSOLE: Could you characterize what you would call the worst containment bypass potential? I'm talking about the suppression pool bypass.

MR. VIJ: The suppression pool bypass?

MR. EBERSOLE: I know you've buried the pipes. That's dead and gone. You don't have to worry about downcomers. You do have reverse flow valves, don't you?

MR. VIJ: We have the vacuum breakers.

MR. EBERSOLE: That's what I mean.

MR. VIJ: Yes. Which are set at 2 psig.

MR. EBERSOLE: Is that the detail that presents what you would call the highest potential for suppression bypass?

MR. VIJ: No, I don't think so. They are redundant

valves in series as well as in parallel. Parallel, of course, doesn't help, but in series and then open inside. I think there's a checkvalve and global motor-operated valves, many, as I looked at those ones.

No, I don't see that as a potential bypass at all. And again, subjected to -- you see, they are also designed for 30 psig, okay.

MR. EBERSOLE: Yes.

MR. VIJ: And the phenomenon that we are interested in at this moment is a differential of 5 or 6 psig.

MR. EBERSOLE: What sort of pressures do you design for for the maximum pipe break?

MR. VIJ: Design pressure?

MR. EBERSOLE: Of the drywell.

MR. VIJ: It's 30 psig.

MR. EBERSOLE: And what is the maximum calculated pressure you will ever get on this?

MR. VIJ: I think it is on the order of 15 or 16, but I'm not sure.

MR. EBERSOLE: About half that?

MR. VIJ: It's a lot less than 30 psig.

[Slide.]

MR. VIJ: Now with that as a background, I think the discussion of other things -- well, let us see what phenomenon we are interested in. These is the failure modes discussion

again, loading and failure mode, hydrogen detonation in containment.

The loads are shock wave, internal pressure on the containment and external pressure on the drywell. In this case, if you have a local detonation -- and the people have presented these results before to you -- the containment failure is assumed above the water line, and I can repeat that below the water line, we have the 1.75-inch-thick plate backed by eight feet of concrete.

If it's a global one, again the containment failure above the water line, and the results have been presented to you where drywell failure was also postulated.

So in this case, both of the barriers are postulated to fail.

In the second scenario, when you have hydrogen combustion, you generate internal pressure on the containment and small external pressure on the drywell. Again, I'm assuming a somewhat smaller rate of hydrogen generation here, about approximately 5 psig. Containment failure above the water line, same reasoning, and no drywell failure, since there are no significant loads, about 5 or 6 psig differential.

Then you go to the slow burning, in slow burning, again internal pressure on containment, small external pressure on the drywell, no containment pressure since

pressure doesn't reach the failure capability of the containment, and the drywell again only 5 or 6 psig differential.

The last one in this case is steam and/or non-combustible gas overpressurization. Now even though it's happening inside the drywell, slow pressure generation again, small internal pressure on the drywell because of the free communication between the drywell and the containment, only differential pressure to that extent, and internal pressure on the containment. Again, we are assuming containment failure above water line at containment ultimate pressure capability, and no drywell failure since no significant loads.

Now over here is when we are talking about the containment ultimate capability, Dr. Okrent, and this is where they mention a number of 85 psig. And if it's 80, 75, 70 -- actually Deborah has the number -- even if it fails at 17.25 psig at which the containment was tested, the increase in this case is negligible.

Okay. With that kind of a background, you know, we shouldn't be too much interested in the personnel lock of the drywell head, because they are not being subjected -- they are not being challenged.

MR. REMICK: A question: Maybe you are going to go beyond this, but Item No. 4 that you are talking about, the steam overpressurization, you say small internal pressure on

drywell, approximately 5 psig, is that the maximum pressure you would reach in a rapid steam release?

MR. VIJ: No. In the rapid, I think, as I mentioned, I can confirm the numbers, I might have about 15 or 16 psig.

MR. REMICK: Okay. So this is the slow?

MR. VIJ: Yes, this is the slow.

MR. MARK: In Item 3, you say -- or no. In Item 4, you say you fail at ultimate pressure capability. In the hydrogen, Item 2, I think, you say containment failure above water line. Again, at the ultimate pressure capability?

MR. VIJ: Yes. But the pressures are more than the ultimate capability.

MR. MARK: Well, they are. It depends on how much hydrogen you are burning.

MR. VIJ: The calculated pressures, am I right?

MR. MARK: It depends on what the fraction of hydrogen is. If you only have 8 percent, you don't approach that capability. If you have about 20 percent, you might.

MS. HANKINS: Deborah Hankins, General Electric.

Recall that in all of these cases, we are assuming no igniters. So we are allowing the hydrogen in most cases to build up to a fairly high concentration before we combust it, and that is why we get such high pressures.

MR. MARK: Well, I am understanding that you need to

get at least 20 percent before you will get above these ultimate pressures.

MS. HANKINS: No. We find if you combust hydrogen at even, say, 12 percent very rapidly -- in other words, combust it in a very short period of time -- we get pressures of 120 psi.

MR. MARK: [Nodding negatively.]

MR. KERR: Are you both using the same percent? When you talk about percent, what are you talking about?

MR. MARK: Well, we think we burned 8 percent at TMI and got 28 psi.

MR. KERR: No. Eight percent of that that was available.

MR. MARK: No, no. Eight percent hydrogen in a hydrogen and air mixture.

MR. KERR: Well, are you talking about the same percent?

MS. HANKINS: Yes -- no, not the same percent. We are both talking hydrogen by volume. I'm saying about 12 to 15 percent, we get very high pressures, because it's instantaneously combusted, no heat transfer.

MR. MARK: I guess I'm a little surprised.

MR. MOELLER: Are there vacuum breakers between the wetwell and the drywell?

MR. VIJ: Yes. That's what we were talking about

previously.

MR. MOELLER: Okay. Do you have data on the failure rates or projected failure rates for those?

MR. VIJ: No, I don't have. But Debbie is saying -- she is nodding in the positive.

MS. HANKINS: Yes. We submitted a study to the Staff on suppression pool bypass, which included the failure of the mechanical and the motor-operated valves.

MR. MOELLER: So you have analyzed for it?

MS. HANKINS: Yes, we have. It's a very low probability for the failure of both of those.

[Slide.]

MR. OKRENT: By the way, is there any way the water in the pool can drain out? Is there any failure aside from a hole in the containment -- I'm not talking about the containment structure itself -- that could lead to draining or loss of a substantial part of the water?

MR. VIJ: Yes, if you assume a break in the holey pipe, so to say -- that is, the ECCS suction lines have one valve on the outboard, and if you assume a break between the valve and the containment boundary, which is equivalent to assuming a hole in the containment, yes, under those circumstances, you would.

MR. OKRENT: There is just one of these pipes?

MR. VIJ: No. There are at least five

MR. OKRENT: Okay. We will come to that another time. Thanks.

MR. VIJ: Looking for some validation methods, I think our discussion covered that. Even if you look at these sizes and loads, failure modes, what we discussed were based on analyses. We considered the load types and application. You consider the structural configuration of the structures with the loads that are imposed. You calculate the stresses, and the highest stressed points are assumed to fail first, and no failures are assumed where loads are significantly less than the design loads.

Based on that, I think we can very comfortably conclude in the present discussion that the drywell structure, head, personnel lock are not challenged, and suppression pool bypass due to drywell boundary failure should not be a concern.

With that, in your presentation, I do have the sketches for the drywell head and the personnel lock. You can look at them, but I don't think they are challenged.

In the interest of time, I think I will go to the last question, which was on the molten core and the effect on the basemat and drywell wall.

[Slide.]

MR. OKRENT: By the way, what is just below the vessel? Is it concrete, or is there what I will call a cavity

that could be filled with water before you see concrete? In other words --

MR. VIJ: During normal operation -- let me get the containment slide for a minute --

[Slide.]

During normal operation, the water is between the rear wall, communicating through these vents with the suppression pool --

MR. OKRENT: Well, I can't see inside.

MR. WARD: David, look at the Brookhaven, second page. If that's accurate, I think it shows it.

MR. OKRENT: Yes. Okay. So there is that space below for receiving the control rods, drives, et cetera.

MR. VIJ: There's the vessel here and the vessel skirt and the pedestal and the pedestal over here (indicating). This is a cavity. There is no water in this cavity during normal operation.

MR. WARD: Is that clear enough?

MR. OKRENT: Yes, that helps.

MR. WARD: And the answer is that that is not going to be full of water, or that is full of water?

MR. VIJ: Now if you get a break over here where you are spilling water on the floor, then there are opening in the pedestal which can take water into the cavity.

MR. ETHERINGTON: And you would lose enough water so

the suppression pool became ineffective?

MR. VIJ. No. In the design, you take care of -- you include in your calculations the drawdown volume, we call it, and compensate for that, and that's why in this GESSAR-II design, you have a pool dump as a design feature.

[Slide.]

A very simplified analysis was performed where we assumed that the molten core burns about a six feet deep hole in the basemat, and it is encircled by the pedestal's inside diameter, as I showed you on the last slide. And the temperature of the molten core was assumed to be 4000 degrees Fahrenheit.

First, it's a two-part analysis. First, you perform the heat conduction analysis and a linear elastic stress analysis performed with those pressure radiations and the corresponding displacements, and then these are the results.

The temperature in the drywell wall at the basemat -- and let me show you again over here (indicating) -- this is the drywell connection to the basemat, and this is what I'm talking about. At that point, the pressure is on the order of 150 degrees -- actually on the outside wall, it's about 50 degrees; on the inside wall, it is slightly higher than that, and I just picked a number between 50 and 200, 150 degrees Fahrenheit.

The flexion of the drywell wall was approximately

half an inch at that point, and then we have a composite section there. It is two concentric cylinders tied together by shear ties and filled with concrete. And we calculated the stresses in that composite section. And in concrete, we had about 3500 psi. The maximum compressive strength of concrete is about 4000 psi. And in steel, we calculated the stresses close to 40 ksi, as compared to a yield strength of 46 ksi.

So we don't expect any danger to the drywell wall's overall stability because of these assumptions of the molten core.

MR. OKRENT: How much hotter would the molten core have to have been for you to be concerned? Can you make a guess?

MR. VIJ: I don't know whether I can assume the linear elastic behavior or not. Suppose I put this one as the limit (indicating), and then, of course, there is not much margin. But at the same time, at this moment, you have a few bulges here and there. You know, you don't want to start the plant all over again after the molten core. So what we are worried about is the overall stability, and I think there is quite a bit of margin. You can probably crush the concrete locally. You can go beyond yield in steel. There should be a lot of margin.

And then as you go above the wall, these values are reduced significantly.

MR. SIESS: Are you thinking about the compression load on the wall there, when you're talking about the concrete? Are you thinking about the compression load on the wall, the axial compression on the wall?

MR. VIJ: Yes. What happens is, on the inside, you have a compression. You see, the slab is trying to expand. The wall is going to stay there. So this is the kind of deformation that you have at the joint, where the inside will be in compression and the outside in tension.

MR. SIESS: So you have a bending.

MR. VIJ: Yes, a bending.

MR. SIESS: And you are worried about the concrete strength in bending?

MR. VIJ: Well, when you have bending, then you can work them into equal compression on the outermost fiber of concrete and tension on the outermost fiber of steel.

MR. SIESS: I suspect gravel in there would be about as good as the concrete, so I wouldn't worry too much about the deterioration of concrete strength. I think we looked at that once at Fort St. Vrain.

How thick is the steel?

MR. VIJ: The steels is 1.25 inch on the inner cylinder and 1.25 on the outer cylinder.

MR. SIESS: The thickness of the wall?

MR. VIJ: The thickness of the wall is five feet.

And both the cylinders are tied together by shear ties.

MR. SIESS: The same thickness inside and out? The steel is the same thickness on the inside and outside?

MR. VIJ: Yes. Both of them are 1.25 inch thick cylinders,

If you have no more questions, I think I have covered my area.

MR. OKRENT: Are there Staff comments in this area?

MR. ROSENTHAL: At your choice, I can either speak to concrete thermal stresses, things we've included or excluded.

MR. OKRENT: Why don't you offer the comments that you feel are significant that are on the same subject just covered by GE, so that we have them together. And then let's see where we stand on time.

MR. ROSENTHAL: My name is Jack Rosenthal. I am in the Reactor Systems Branch, Division of Systems Integration.

[Slide.]

We have a containment here with a dry cavity, and we will focus on the containment cavity right over here (indicating).

Now let me give you a little bit better picture.

[Slide.]

The drywell walls are two feet thick. The pedestal here is about six feet thick, and we have 25 feet from the

cavity to the wetwell. And we are concerned about the integrity right over here (indicating).

The reactor deadweight is, in part, supported by a skirt that sits on the shelf, and in part, the deadload is supported by piping hangers up in the upper head. And the concern that we had here was ablation of this concrete.

Now the corium would be 1.6 meters as some sort of froth height, and it's going to dig its way down in the pedestal. There are three feet to a liner and then another twelve feet to the bottom of the basemat, so it has a ways to go down. And we just let the corium just sit up here as a hand calculation, or at least the APS keeps telling us it's more reliable than our codes.

And what we did is, we just said that the heat to the walls is rho C sub p, delta T; and ablation, heat of ablation, and then some ablation rate. I have values of the heat to the walls. Let me flip back to the slide here. I'm going to call this the lower cavity. It is radially eroding out, downward eroding out, and I'm going to call this the upper cavity (indicating), and we're talking about radiative heat transfer and convective heat transfer up. You might see -- quarter of the surface area might be the walls up here (indicating), and it really doesn't matter if these walls come down or not in this analysis.

So we had some heat fluxes as a function of time out

of Corcon, various types of concrete, different Corcon mods. Let me just give you a range of values here: 10 to 20 watts per centimeter squared; lower down, 12 to 3 watts per centimeter squared; going up --

MR. OKRENT: Could you possibly find a pointer and stand off to one side? I'm not sure I will understand it any better, but I could at least see it.

MR. ROSENTHAL: Let me point out, you expect an aerosol in here which acts as a radiation shield, and we have a boundary condition, and that is, there's no way that steel is going to melt. The upper head itself would melt at some temperature, and the dripping of steel into the corium actually reduces the temperature of the corium.

[Slide.]

We calculate ablation rates at 10 to 20 centimeters per hour in the lower cavity, 2 to 3 centimeters per hour in what I term the surroundings. That agrees reasonably well with the beta tests, which showed early on about a foot an hour, roughly 20 centimeters per hour, as an initial ablation rate, but of course that's going to slow down as the stuff cools off.

And we looked at the total ablation at 10 hours, and I get 120 centimeters and 140 centimeters, 120 centimeters axial erosion, 140 centimeters radial erosion of the pedestal.

[Slide.]

And I conclude that pedestal integrity would be doubtful. And what you have done is, you have eroded out the concrete right over here (indicating), and that, in turn, takes the weight of the shelf above it. So I am not at all convinced that the reactor vessel would not move at 10 hours.

Now let's get a little bit of perspective in on this. By the time the 10 hours has elapsed -- and let me say that I consider this is a band calculation -- we held the concrete up or the corium up instead of let it chew its way down. I think we are using conservative heat fluxes, but we still have the concern.

[Slide.]

Early on in the event, you took the volatile fission product release and you put it into the pool. At about 150 minutes into the event, we would say that the vessel fails, and you begin core concrete interactions.

Now those interactions will continue on for some period of time. By 12 to 15 hours into the event, you have frozen the corium that is sitting on the floor. Again, that's what we think is a reasonably conservative calculation. So what we believe is that should this event, in fact, occur, you have put the volatiles into the pool. The majority of the vaporization release has already occurred, and those fission products are somewhere else, hopefully in a retained state at that time.

I'm not sure that the reactor won't, in some way, move, and that has the potential for ripping out major piping that penetrates the drywell and wetwell. So I am not at all confident that one wouldn't have a geometry other than the one you are looking at, but I think that it is at a time when the risk viewpoint says that that is not an intolerable failure

[Slide.]

We were asked to look at --

MR. OKRENT: Excuse me. You are saying, at that point there is at least a fair chance that if the reactor were no longer well-supported, that you would lose drywell integrity; is that what you're saying?

MR. ROSENTHAL: Drywell and conceivably wetwell integrity, because those pipes penetrate both boundaries. They are pretty big, rigid pipes, although they may be very hot themselves by that time.

MR. OKRENT: Okay.

MR. ETHERINGTON: What percentage of the decay heat is in the corium?

MR. ROSENTHAL: The overall decay heat is about 1.5 percent at two hours after trip, and decreasing pretty slowly; about 70 percent of the decay heat is in the corium at two hours or so, and the rest being in the iodine and cesium.

Now as you go out in time, of course, because the halflives are different, a greater proportion of what is left

of the decay heat is in the corium.

[Slide.]

You had asked about thermal gradients, and we just took a one-dimensional, time-dependent heat transfer and took the surface of the pedestal as 1700 K. Just assume it is chewing away. And we just said, what kind of thermal gradients are we going to see, and I sketched in what I hope is an exponential, and what we see is that an hour into the event, penetration is about 25 centimeters, and at 20 hours, about 60 centimeters.

Now you should compare that to 7.7 meters to go from the cavity out to the pool, and then you have to ask yourself -- let's say that you do crack through from the cavity into the pool. Well, under the temperature gradients that you're talking about, the water ought to still be running away from that heat front. But if you flip the problem around and say that the water does somehow invade the cavity, then so much the better.

Okay. I would now like to --

MR. WARD: Could I interrupt just a minute? Dave, we have ten more minutes. Does this fit your schedule okay?

MR. OKRENT: Well, I think it is better for them to treat a certain number of matters in sufficient detail and let the committee ask its questions.

General Electric came up to me before the meeting

and said, "It looks like a very crowded agenda," and, you know, we are going to meet with them again next month.

MR. ROSENTHAL: Let me just point out --

[Slide.]

-- we assumed when these analyses were done at 72 psi, a containment failure -- and we are talking eleven hours with intercalculations -- Corcon Mod II, for what it's worth, is 28 hours to 72 psia, and maybe never to the kinds of pressures that were discussed earlier today.

This assumes only passive heat sinks. If you go to 45 psi and say, let's assume no more than 45 psi as some sort of failure pressure, then with Corcon, again 11 hours to containment failure.

[Slide.]

Now for the more meaty matter of this particular meeting, and that is, what did we include and exclude, I think, is the heart of this meeting.

We did consider overpressure failure due to non-condensible gas generation. We considered deflagration and detonations. We looked at structure, seals, piping penetrations and their fragility to pressure and temperature.

We did not consider steam explosions. We did not consider direct heating. The direct heating we view as a high-pressure problem primarily. The dominant sequences on this plant are low-pressure, because of the perceived

reliability of the ADS.

Mechanical failures, we did not consider RPV failure. From a probablistic standpoint, there is an argument that double MSIV failure is unlikely, but it is not included in our consequence analysis. And of course that is, it is a leak to a fluid system, to a contained system, but of course it bypassed the drywell and wet well.

We didn't look at a large-break LOCA with stuck-open vacuum breakers. One of the features of the newer ASPO work is credit for primary system retention of some amount, and the primary system retention would be largest in something like a station blackout, which is portrayed as the dominant event sequence. We would have lesser retention in a small-break LOCA, which goes faster, and we would have the least retention in a large-break LOCA or an RPV failure, where there just isn't time for the aerosols to be retained in the primary system, and hence primary system retention would be minimum.

Let's fold that in with the vacuum breakers, our checkvalve, and an MOV in series. There is some chance that they both failed, and there is some chance that you combine that with a wetwell failure.

The vacuum breaker -- and that's based on a question of the perceived probability, now -- we're talking about a large-break LOCA, a failure of ECCS, a failure of the vacuum breakers. You're really pushing it down. The vacuum breakers

failed closed. There is an interesting wrinkle that Sandia came up with. Let me go back to a slide here.

[Slide.]

Remember we were telling you, or several people have said, that the delta P between the wetwell and the drywell isn't very large, because it can't be greater than the static head of the water to the horizontal vents.

Well, let's assume the vacuum breakers fail. And they way they might fail is you just might have a station blackout sequence, and the MOV on the vacuum breaker line fails the strobe prior to loss of electric power. Then the pressure in the wetwell is greater than the drywell, and you'd slosh some water back over into the drywell and into the cavity.

It's an interesting wrinkle because it just makes things better rather than worse, because it would quench the vaporization release.

Now we did look at a range of containment failure times, and the first one I focused on was a 1-T-E3 and 1-T-L3. The late failure is at eleven hours here, and the early failure is at 150 minutes or so. We get a difference in consequences for the difference between early and late failure of about 3. And that would assume --

MR. EBERSOLE: Before you remove that, let me ask you a question.

With its new design, GE has eliminated the high-pressure steamlines that used to feed the old HPCI. I think they retained the turbine-driven RCIC, and they still retain the line that goes outboard into the equipment area, which is the reactor water cleanup system. Those two lines -- I think that's all there is -- represent potential through-line failures resulting from the original failure at the pipe outboard of the valves or whatever, and the not-zero potential of valve failure facing a tremendous mass flow and all, you know, the loadings that go with it.

Do you look at these things?

MR. ROSENTHAL: We did, and let me tell you the status. Yes, we're concerned about the RBCW line, the RHR line, and the quintessential issue in this plant is maintaining pool integrity, so we are worried about the RHR line.

MR. EBERSOLE: By the way, the main steamlines I didn't include, because I always think they're going to go into the universe and not kill the equipment which is critical to shutdown; is that correct?

But these other lines have a regressive effect, in that they discharge into the very areas which you need to maintain.

MR. ROSENTHAL: Yes. And we have discussed this. Dr. Michelson was particularly concern on Limerick and

Shoreham.

We have done a systems analysis of those line breaks. You will find it written up in the advance copy of SER-4, which is in your possession, and I am not prepared to discuss it at this time, but you will find it written up.

We do not have conditional consequences calculated for those events at this time. In order to do it with some fairness, one has to use the ASPO code suite in a mode which models revaporization of fission products and posits it on long lengths of pipe, and no one has done a calculation of that sort yet.

What you have to do is say, what's the probability of the event, and what is your judgment of the relative -- of the fission product behavior for that event, as distinct from the 1-SB-E1 event, which is also in your advance copy of the SER, which is a bypass due to hydrogen detonations. And we have made the judgment that, although these line breaks are of concern and they are the equivalent of the boiler Event V, that the consequences would not be significantly worse than what is labeled the 1-SB-E1, which is about an order of magnitude worse in consequences than that 1-T-L3 event, and we'll confirm that.

MR. EBERSOLE: When you looked at the consequences or when you used to look at them, all you looked at was just the direct emanation of fission products and, you know,

getting out and going in the wrong places. You didn't look at equipment degradation.

MR. ROSENTHAL: In the systems analysis we did -- and you will find it written up -- we asked, where does the water go, when does it go, and what effects does it have.

MR. EBERSOLE: Well, is this design constrained to preclude discharge of these pipes into any area where one could not theoretically just continue to discharge while you cool the core down?

In short, if this equipment, RCIC, and the reactor water cleanup system in cellular design areas where the end result of this discharge, continued discharge, is merely to run on out to atmosphere without degrading shutdown equipment, you have escaped the bulk of the problem.

MR. OKRENT: Jesse, I would suggest that we not pursue this now. There is going to be some subcommittee meeting where we take up the remaining Michelson/Ebersole et cetera questions, and this will be --

MR. EBERSOLE: I'm just suggesting that this be a design constraints.

MR. OKRENT: I understand Mr. Scaletti or Mr. Thompson wants a couple of minutes, so you had better finish up in one minute. I had certainly asked for more than two hours, but that's all we could get at this meeting. It's a crowded meeting.

Go ahead.

MR. ROSENTHAL: There is a concern over excessive drywell/wetwell leakage. We do have the E1 and E2 sequences which model scrubbing of the valval release, but bypass, total bypass of the pool of the vaporization release, and the consequences typically differ from the late ones by a factor of 3 or 4 in terms of person-rem, and you can find that in the SER.

You are also concerned about drywell head failure, and you can see again in the SER that there are conditional consequences for the I2/I2Q sequence, and I think there was again a factor of 3 or 4 difference.

The reason for that is, should that pool, in fact, fail at the head -- right over here (indicating) -- then you quench the core debris. It's a fortuitous failure. One should not take credit for miracles in PRAs. But in any case, if it does fail here, you quench the vaporization release. If it does not fail here but fails some other place, you dump the water here and here (indicating), and then you don't fail. But in terms of ultimate consequences, we are only seeing a factor of 3 or 4 difference.

And the reason for that and the reason for all of these is, as I said -- the issue here is, do you have suppression pool? And even on a pessimistic basis of effective pool scrubbing -- that is, an effective filtered

vent -- the thing to worry about is, do you bypass that pool or fail the pool?

MR. OKRENT: Well, I will just note one thing for the committee to think on that arises out of what I heard, and that is this ablation discussion that you gave, which raises the possibility that there may be -- and I will use the word "may" -- a significant probability of a loss of integrity late, but nevertheless a loss of integrity of containment, and therefore, a release of whatever may be available late, as contrasted to the claims for some containments that they stay relatively tight for a large family of accidents.

It is something to reflect on.

MR. ETHERINGTON: I think ablation is a very soothing word for something that might be rather disruptive. I can imagine chunks of the concrete breaking away, rather than just washing away.

MR. OKRENT: You mean it might go faster?

MR. ETHERINGTON: Yes.

MR. OKRENT: Then it would be more likely -- but in any event, I think we had best -- we're two minutes overtime already, so why don't we let the Staff have two minutes?

MR. THOMAS: I am Cecil Thomas, Division of Licensing. I have two brief subjects I'd like to say something about.

As the committee is probably aware, on June 27th the

Commission approved the Severe Accident Policy Statement with one minor modification. That modification does not affect the GESSAR II review.

On July 3rd, the Commission issued a Staff Requirements Memo asking the Staff to modify the Severe Accident Policy Statement as noted and to forward it to the Office of the Secretary for publication in the Federal Register.

We expect the Severe Accident Policy Statement to be published in the Federal Register in the next couple of weeks.

As I have earlier briefed the committee and the subcommittee, in accordance with the provisions of the Severe Accident Policy Statement, we are preparing to amend the GESSAR II FDA to permit it to be referenced in new CP and OL applications until the severe accident review is completed. And I emphasize, the amendment would allow GESSAR II to be referenced, but we would not issue a new CP or OL for an application that referenced GESSAR until the review was completed.

Upon the successful completion of the severe accident review, we will further amend the GESSAR II FDA to allow it to be referenced and CPs and OLs to be issued for a fixed period of time.

The second subject has to do with some problems the Staff is experiencing in the review. As I think both the

committee appreciates and the Staff appreciates, this is the first standard design to undergo a severe accident review under the provisions of the Severe Accident Policy Statement. It certainly has taken a lot more time and resources than we envisioned at the outset, and I am sure the committee feels the same way.

I would point out to the committee that the Staff, particularly NRR, is feeling an extremely strained impact on the resources required to support this review. We are really strapped on both in terms of Staff, technical assistance dollars, and travel dollars, and it's beginning to adversely impact some of our other reviews.

For that reason, the Staff would urge that the subcommittee and the committee complete the reviews as soon as they reasonably can. Specifically, we would like to be able to wrap this up on a schedule that would result in an ACRS letter in September.

MR. OKRENT: Well, we have heard you, but we will have to see what pace the committee can hear, digest, and assess the information, and we only just got the last SSER -- I got it, I guess, last -- sometime during the last seven days, I don't remember when -- and the point that I find of non-negligible interest concerning ablation, for example, was only first, to my knowledge, brought out at this meeting. I don't think I heard it before.

So maybe we are still learning. And there are some other things on the table that we haven't finished yet -- for example, whether or not something needs to be done concerning security or sabotage. So we are trying to cooperate. We are scheduling as much time as the rest of the committee business will permit.

We will have a subcommittee meeting the day before the full committee meeting next month, and then a meeting with the full committee. I don't know yet what we will try to cover or where we got today, and then we'll certainly need at least one more subcommittee meeting. Mr. Michelson, as you may know, is away in July and August. We will do our best to cooperate.

MR. KERR: Does the subcommittee have any feeling for what the schedule might be?

MR. OKRENT: I would guess it will be difficult for the committee to make it by September, but we will try to make it as close to that.

MR. SIESS: Have we got a list of open items, or do you have a list of open items?

MR. OKRENT: The term "open items" is a complex one in this review. The Staff may have some list, but the committee may be looking at the term "open items" in a different context.

MR. KERR: Did you understand from Mr. Thomas'

comment how the Staff intends to deal with the severe accident issue differently than before the Commission approved the policy statement? It really was not clear to me. It doesn't have to be, but --

MR. OKRENT: If you have a question on what Mr. Thomas said, I would prefer you address it to him.

MR. KERR: Okay. I will wait.

MR. THOMAS: One thing I forgot to mention that does have a bearing on the need for September, as you are all aware, NRR has a pending reorganization. We feel that will further exacerbate our resource problem. That's why September --

MR. SIESS: Is that the intent of the reorganization?

[Laughter.]

MR. SIESS: Or is it intended to improve things?

MR. THOMAS: I will let that pass.

MR. OKRENT: Anyway, thank you all. I think, in fact, we did, indeed, have a more detailed discussion this time than last time, and that is to the point. I think we can dispose of certain matters.

Okay. Let's break for lunch and return at 2:10.

[Whereupon, at 1:10 o'clock, p.m., the general meeting was adjourned, to reconvene at 2:10 o'clock, p.m., in executive session.]

CERTIFICATE OF OFFICIAL REPORTER

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Name of Proceeding: 303rd General Meeting

Docket No.:

Place: Washington, D. C.

Date: Friday, July 12, 1985

were held as herein appears and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission.

(Signature) Suzanne B. Young

(Typed Name of Reporter) Suzanne B. Young

Ann Riley & Associates, Ltd.

NRR STAFF PRESENTATION TO THE

ACRS

SUBJECT: WATTS BAR IDVP AND ALLEGATIONS

DATE: JULY 12, 1985

PRESENTER: ms. E. G. ADENSAM

PRESENTER'S TITLE/BRANCH/DIV:
CHIEF, LICENSING BRANCH NO. 4, DIVISION OF LICENSING

PRESENTER'S NRC TEL. NO.: 301-492-7831

WATTS BAR IDVP AND ALLEGATIONS

* GENERAL REQUIREMENTS OF AN IDVP

* WATTS BAR IDVP

* STAFF REVIEW

* WATTS BAR ALLEGATIONS

GENERAL REQUIREMENTS OF AN *IDVP*

REVIEW ORGANIZATION

* COMPETENT

* INDEPENDENT

SCOPE OF REVIEW

* TAILORED TO CONCERNS

WATTS BAR IDVP

* BLACK & VEATCH

  - AUDIT THE AFW SYSTEM OF WATTS BAR UNIT 1 TO ENSURE THAT THE SYSTEM HAS BEEN DESIGNED AND CONSTRUCTED IN ACCORDANCE WITH THE LICENSE APPLICATION AND LICENSE COMMITMENTS

* TVA

  - EVALUATE THE BLACK & VEATCH FINDINGS TO DETERMINE THEIR APPLICABILITY TO OTHER WATTS BAR SYSTEMS

STAFF REVIEW

BLACK & VEATCH EFFORT *(AFW SYSTEM)*

* B&V BOTH COMPETENT AND INDEPENDENT
* 97 FINDINGS REVIEWED OUT OF 428
* ONE FINDING STILL UNDER REVIEW
  - RESPONSE TO IE BULLETIN 79-02

TVA GENERIC EFFORT

* INSPECTION REVIEW
* DEDICATED REVIEW GROUP

OBJECTIVES OF REVIEW GROUP

1. TO DETERMINE IF THE TVA PROGRAM TO ADDRESS THE FINDINGS OF THE B&V REPORT WAS ADEQUATE WITH RESPECT TO EVALUATION OF GENERIC APPLICABILITY OF THE FINDINGS AND CORRECTIONS MADE TO THE PLANT DESIGN AND CONSTRUCTION

2. TO ADDRESS RECENT ALLEGATIONS REGARDING IDVP

DEDICATED REVIEW GROUP
REVIEW OF WATTS BAR IDVP

* B&V REPORTS

* TVA POLICY COMMITTEE REPORT AND RELATED DOCUMENTS

* NSRS REPORTS AND RELATED DOCUMENTS

* SITE VISIT

* PREPARE SER

TYPES OF CONCERNS RAISED

* GENERAL
* MOST PREVIOUSLY IDENTIFIED IN NSRS REPORTS, NCR's, CDR's, OTHER TVA CORRESPONDENCE, AND INSPECTION REPORTS
* CONCERNS REGARDING B&V
  - CLOSEOUT OF 500 ITEMS
  - ONLY ONE CONSTRUCTION SPECIFICATION LOOKED AT BY B&V
  - B&V DID NOT KNOW HOW THE PLANT WAS BUILT
  - B&V COMPARISON MADE WITH REGARD TO DESIGN CRITERIA NOT REGULATORY CRITERIA
* OTHER TYPES OF ISSUES:
  - MATERIAL TRACEABILITY
  - RECORDS OF TOTAL LOADS
  - STRUCTURAL STEEL WELD REQUIREMENTS
  - CABLE PROBLEMS
  - PROCUREMENT PROBLEMS
  - VOLTAGE REGULATION FOR BUSES
  - UNISTRUT USE
  - INADEQUATE D/G LOAD MARGINS

REVIEW PLAN

* ASSIGN RESPONSIBLE ORGANIZATION

* SCREEN FOR SAFETY SIGNIFICANCE AND IMPACT ON LICENSING

* REVIEW RESPONSES TO MAY 16, AND MAY 30, 1985 LETTERS

* REVIEW AND EVALUATE ISSUE

* PREPARE SER OR INSPECTION REPORT

Rosenthal #10

GESSAR - SEVERE ACCIDENT THREAT

TO CONTAINMENT

PRESENTED TO ACRS, JULY 12, 1985

JACK ROSENTHAL - ONRR/DSI/RSB

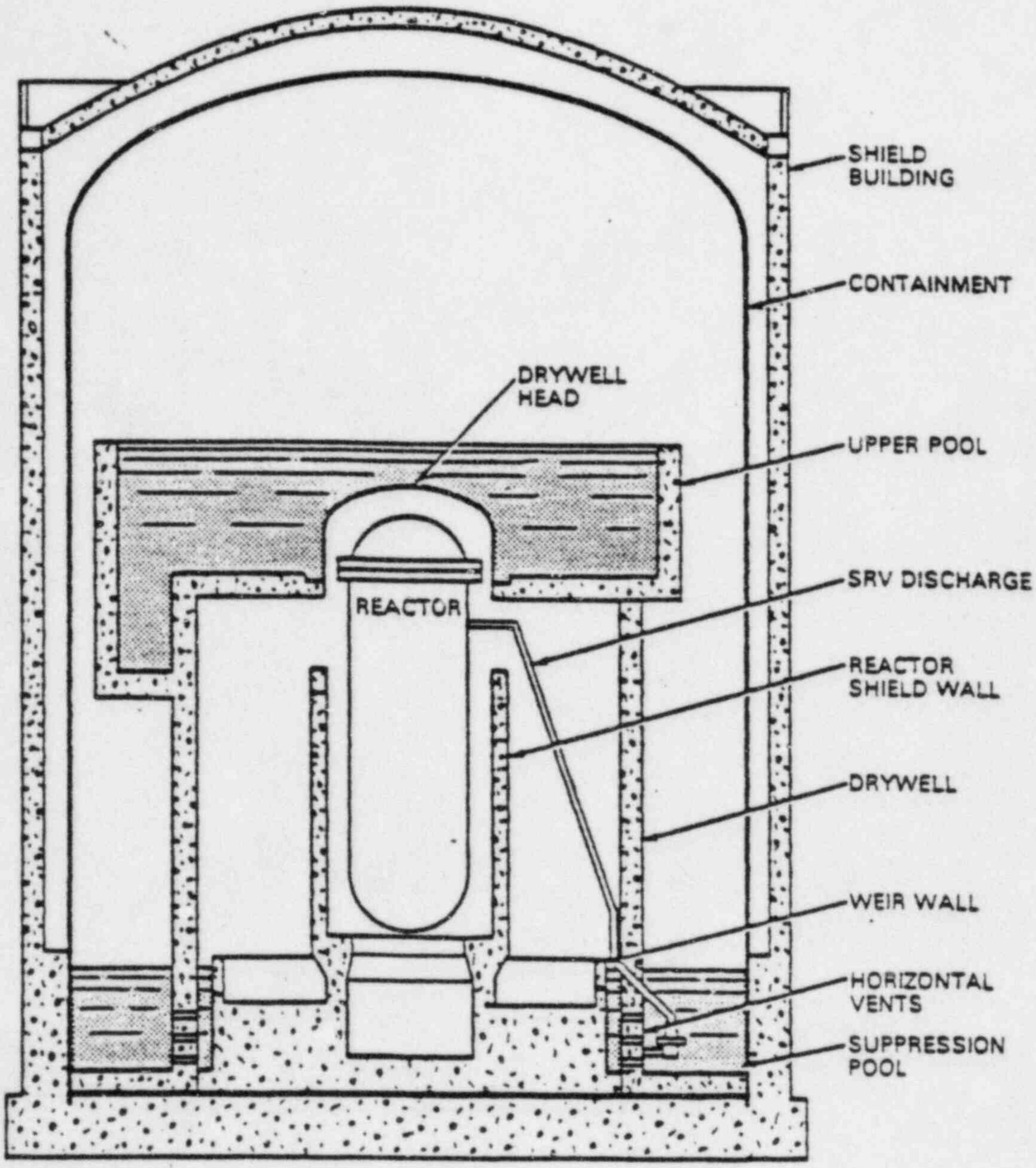


Figure 15.1 Principal features of MARK III containment

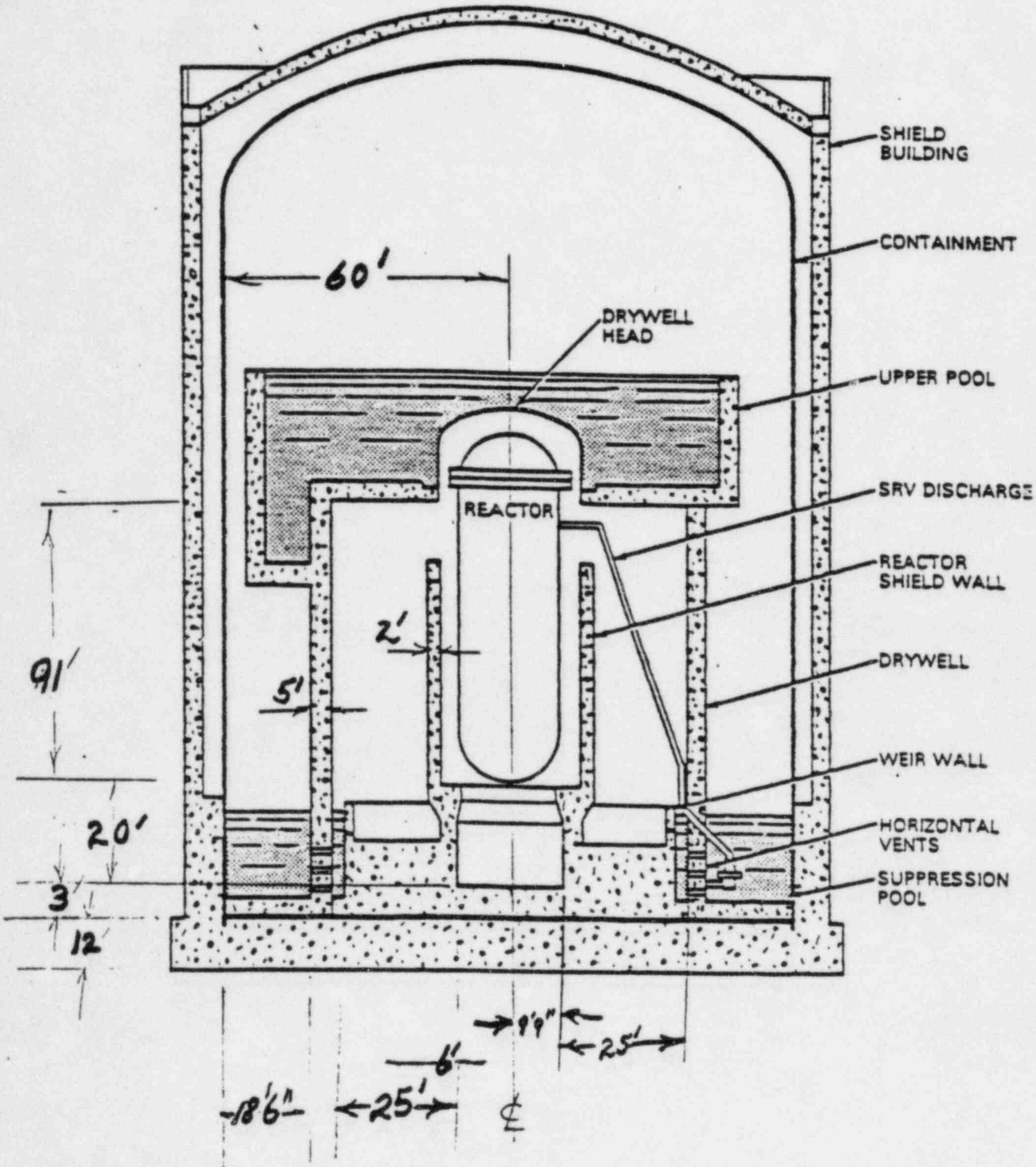
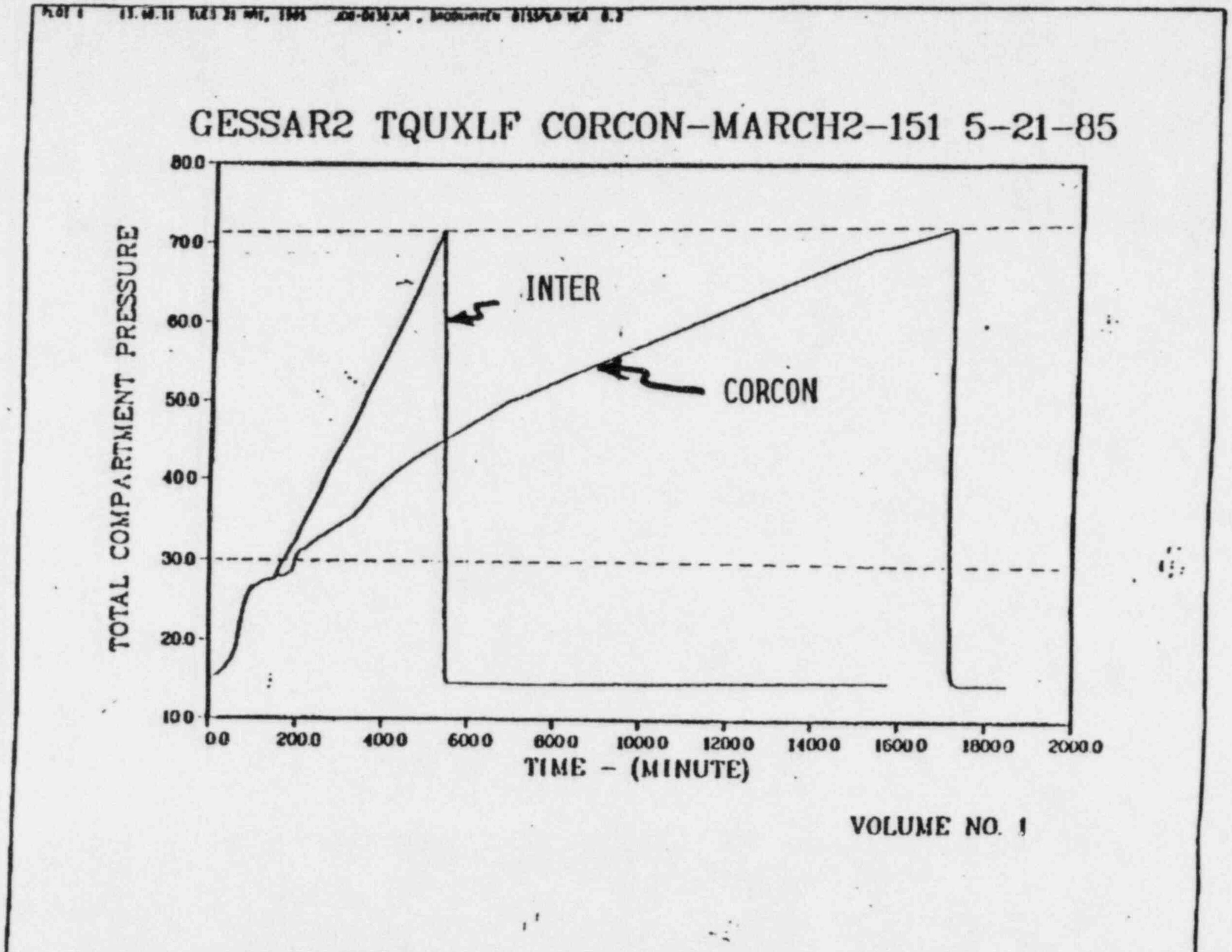
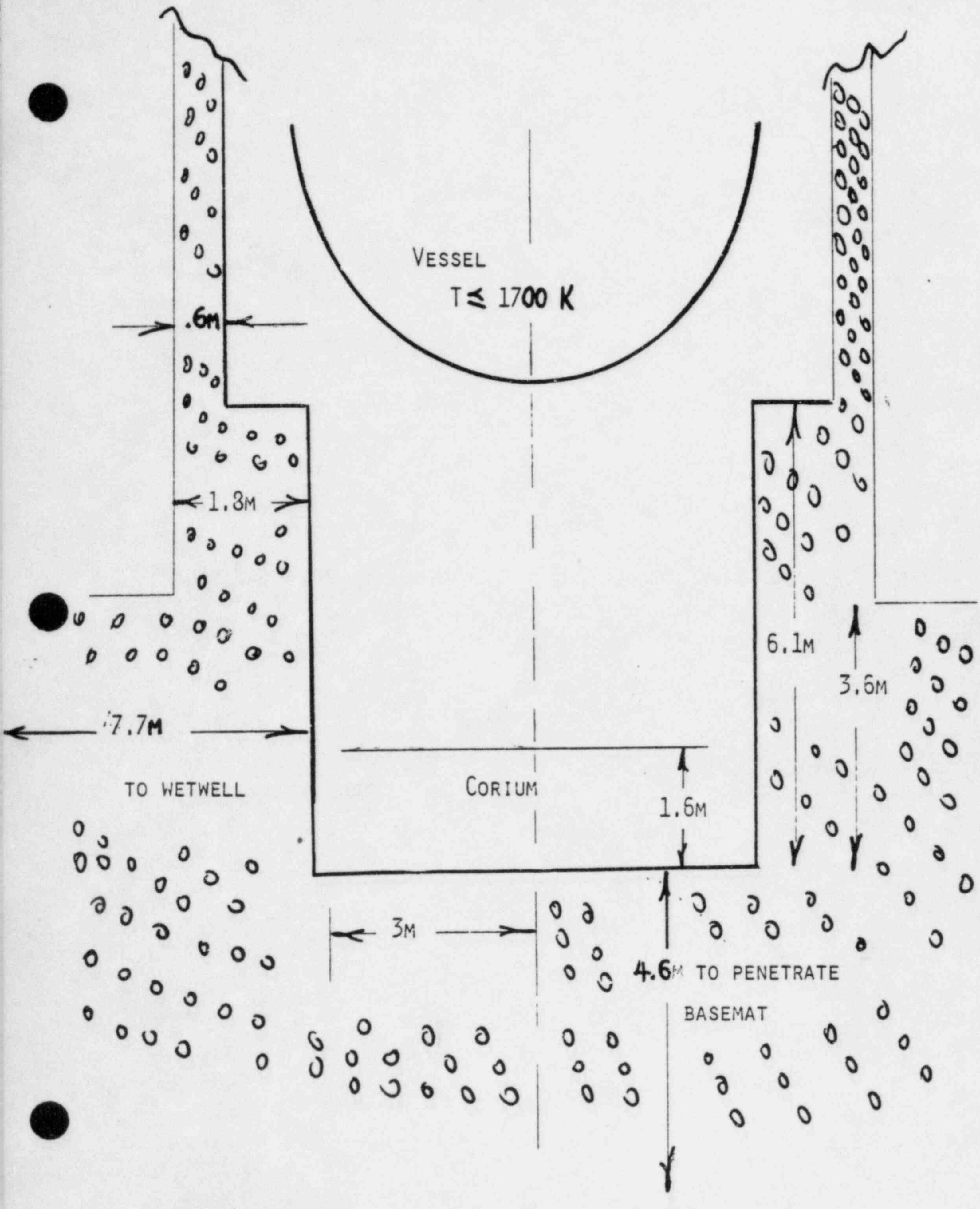


Figure 15.1 Principal features of MARK III containment



GESSAR2 TQUXLF CORCON-MARCH2-151 5-21-85
TOTAL COMPARTMENT PRESSURE
80.0
70.0
60.0
50.0
40.0
30.0
20.0
10.0
0.0
200.0
400.0
600.0
800.0
1000.0
1200.0
1400.0
1600.0
1800.0
2000.0
TIME - (MINUTE)
INTER
CORCON
VOLUME NO. 1


VESSEL
T ≲ 1700 K
.6M
1.8M
7.7M
TO WETWELL
CORIUM
1.6M
6.1M
3.6M
3M
4.6M TO PENETRATE
BASEMAT

- ABLATION RATES

$$\dot{q} = \rho\left[C_p(\text{CONCRETE})(T\ \text{ABLATION} - T\ \text{INITIAL}) + \lambda\ \text{ABLATION}\right]\dot{x}$$

$\dot{q}$ = 10 - 20 W/CM^2 (LOWER CAVITY)

$\dot{q}$ = 12 - 3 W/CM^2 (SURROUNDING)

ρ = 2.5 GM/CM^3, C = 1 J/GM/K, λ = 240 J/GM

$\dot{x}$ = ABLATION RATE INITIAL = 10-20 CM/HR (LOWER CAVITY)

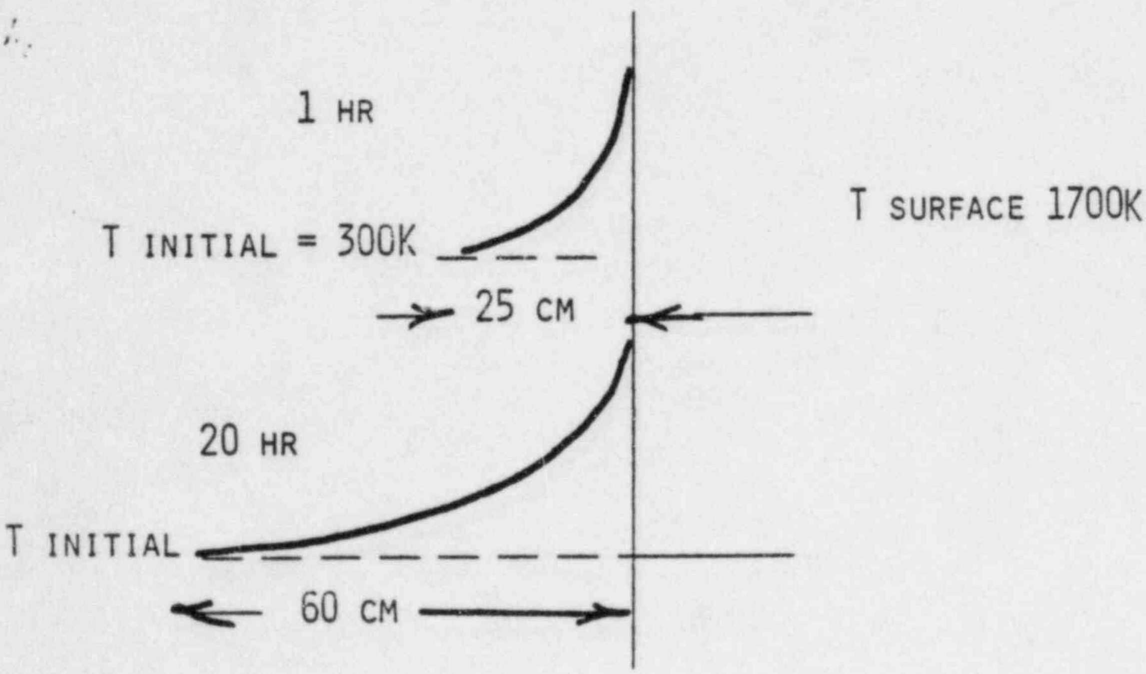
= 12-3 CM/HR (SURROUNDING)

$x_{10\ \text{HRS}}$ ≈ 120 CM AXIAL

≈ 140 CM RADIAL, PEDESTAL INTEGRITY DOUBTFUL

- THERMAL GRADIENT

$$\nabla^2 T = \alpha\, \partial T/\partial t$$



-

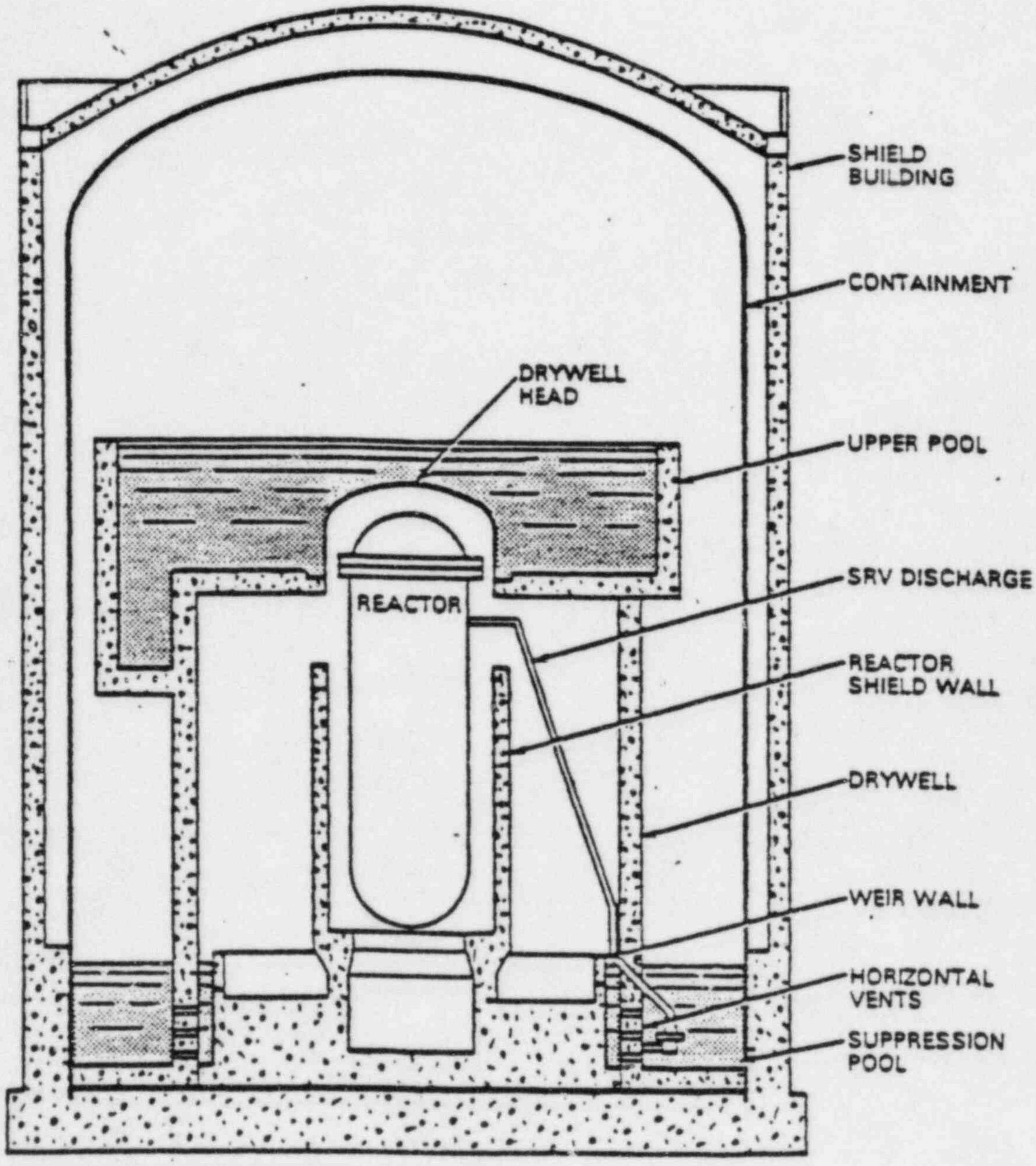


Figure 15.1 Principal features of MARK III containment

- Dominant Containment Failure Modes

  Overpressure due to non-condensible gas generation

  Hydrogen deflagration and detonations

  failing structure, seals and piping penetrations by pressure and temperature

  Phenomenological issues not considered:

  steam explosions

  direct heating

  Mechanical failures not considered:

  RPV failure

  Double MSIV failure

  LBLOCA with Stuck Open Vacuum Breakers

  Vacuum Breakers Fail Closed

  Mechanical failures considered

  1-T-E3, 1-T-L3 releases span range of early and late containment wetwell failure due to potential fabrication flaw

  Excessive Drywell-Wetwell Leakage Considered

  E1, E2, sequences model total bypass of pool by vaporization release

  Drywell Head Failure

  I2, I2Q span range

Table 15.1 Conditional consequences predicted by the staff for internally initiated events and probability of occurrence with and without UPPS, per reactor year

Release category*	Early fatality	Early injury	Latent fatality	Person-rems	Probability w/o UPPS	Probability w/UPPS
1-T-L3	0	0	40	7 x E5**	3 x E-6	9 x E-7
1-T-E3	0	0.0005	200	3 x E6	8 x E-6	1 x E-6
1-T-I2Q	0	3	200	3 x E6	1 x E-5	1 x E-6
2-T-B3	0	0	300	5 x E6	4 x E-6	4 x E-7
ATWS	0	1	400	6 x E6	3 x E-6	3 x E-6
1-T-I2	0	6	500	8 x E6	3 x E-6	3 x E-7
1-SB-E1	0.006	10	600	9 x E6	1 x E-9	1 x E-9

*See definitions in Table 15.15.
**7 x E5 = 7 x 10^5.

Notes:

(1) All conditional mean consequences were calculated using the upper range BNL source term values described in SSER 2.

(2) The calculations assumed the Shippingport site, with public evacuation within 10 miles and relocation 12 hours after plume passage.

(3) Mean consequences were computed over 91 different weather conditions.

CONSIDERATION OF HYDROGEN ISSUES

CURRENT REQUIREMENT

- QUANTITY :100% ZR - H_2O CLAD EQUIVALENT

78000 LB. ZR

3400 LB. H_2

- RATE :ACCEPTABLE TO STAFF

GESSAR

Containment Volume (wetwell + drywell)	= 1.4E6 cu. ft.
Mass Air (@ 80F initial)	= 1.05E5 lb.
Mass O_2	= 2.2E4 lb.
Max Mass H_2 that could burn (based on Mass O_2 and assuming complete combustion)	= 2.7E3 lb.
Corresponding Mass Zr oxidized	= 6.2E4 lb.
Fraction clad oxidized to produce sufficient H_2 to burn all O_2 in containment	= 79%
Fraction clad oxidized to produce sufficient H_2 to burn O_2 to 4% lower limit (2.2E3 lb H_2)	= 65%

GESSAR - H_2 RATES

UNMITIGATED SCENARIOS, I.E. TOTAL LOSS OF INJECTION

IN-VESSEL PRIOR TO SLUMP

FOLLOWING EPG WITH SUCCESSFUL ADS

25 LB./MIN. PEAK

500 LB. TOTAL

FAILURE OF ADS WITH 2SRV'S OPEN

85 LB./MIN. PEAK

1400-1700 LB. TOTAL

IN-VESSEL AT CORE SLUMP

~250 LB. H_2 AT ~400 LB./MIN.

EX-VESSEL CCI

2.5 TO 6 LB. H_2/MIN.

40 TO 150 LB. CO/MIN.

CORCON CALCULATIONS

HEAD FAILURE = 150 MIN.

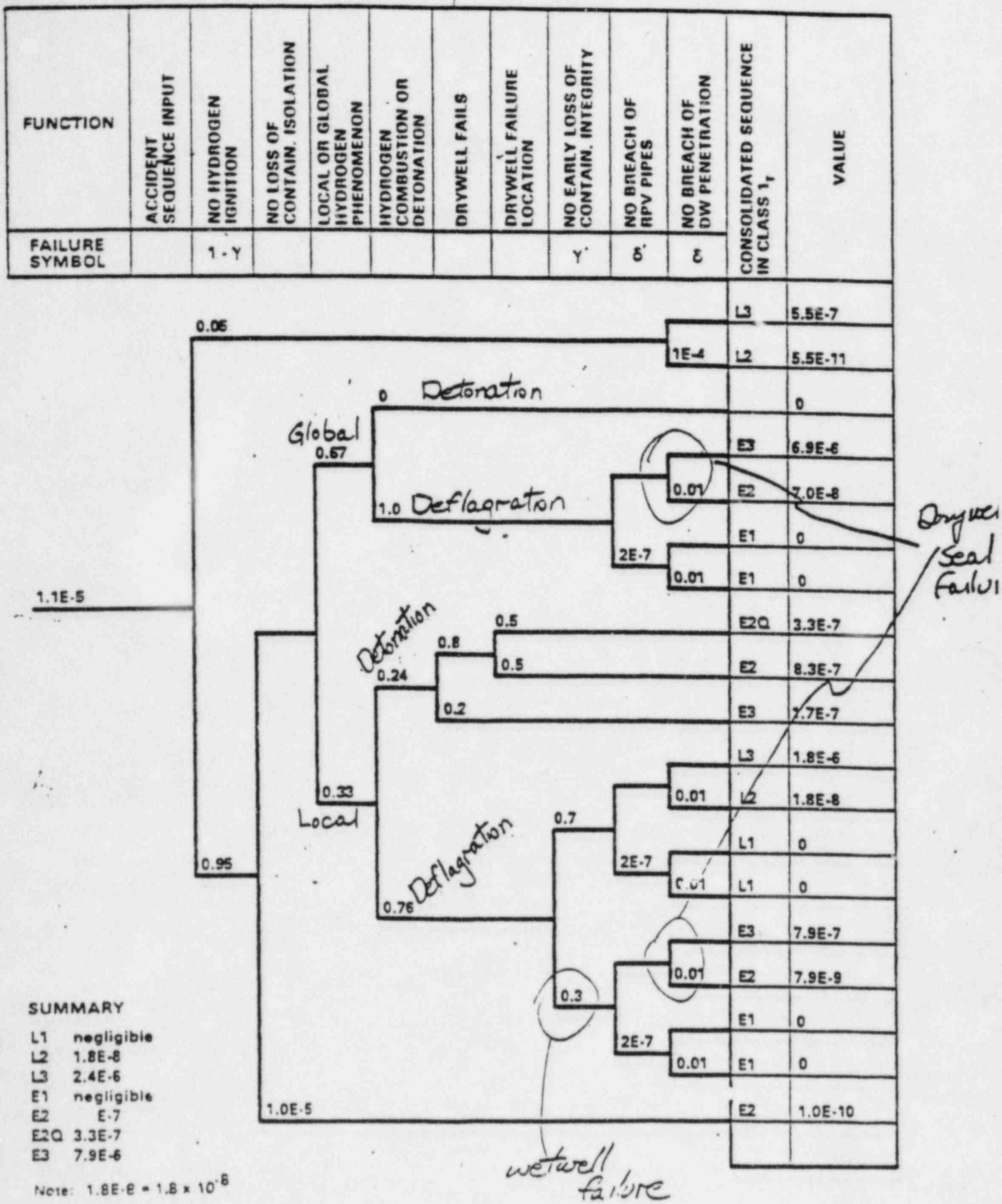
CONT. FAILURE = 1700 MIN.

AT ASSUMED 72 PSIA

4200 LB. H_2

58000 LB. CO

99000 LB. CO_2



SUMMARY

L1	negligible
L2	1.8E-8
L3	2.4E-6
E1	negligible
E2	E-7
E2Q	3.3E-7
E3	7.9E-6

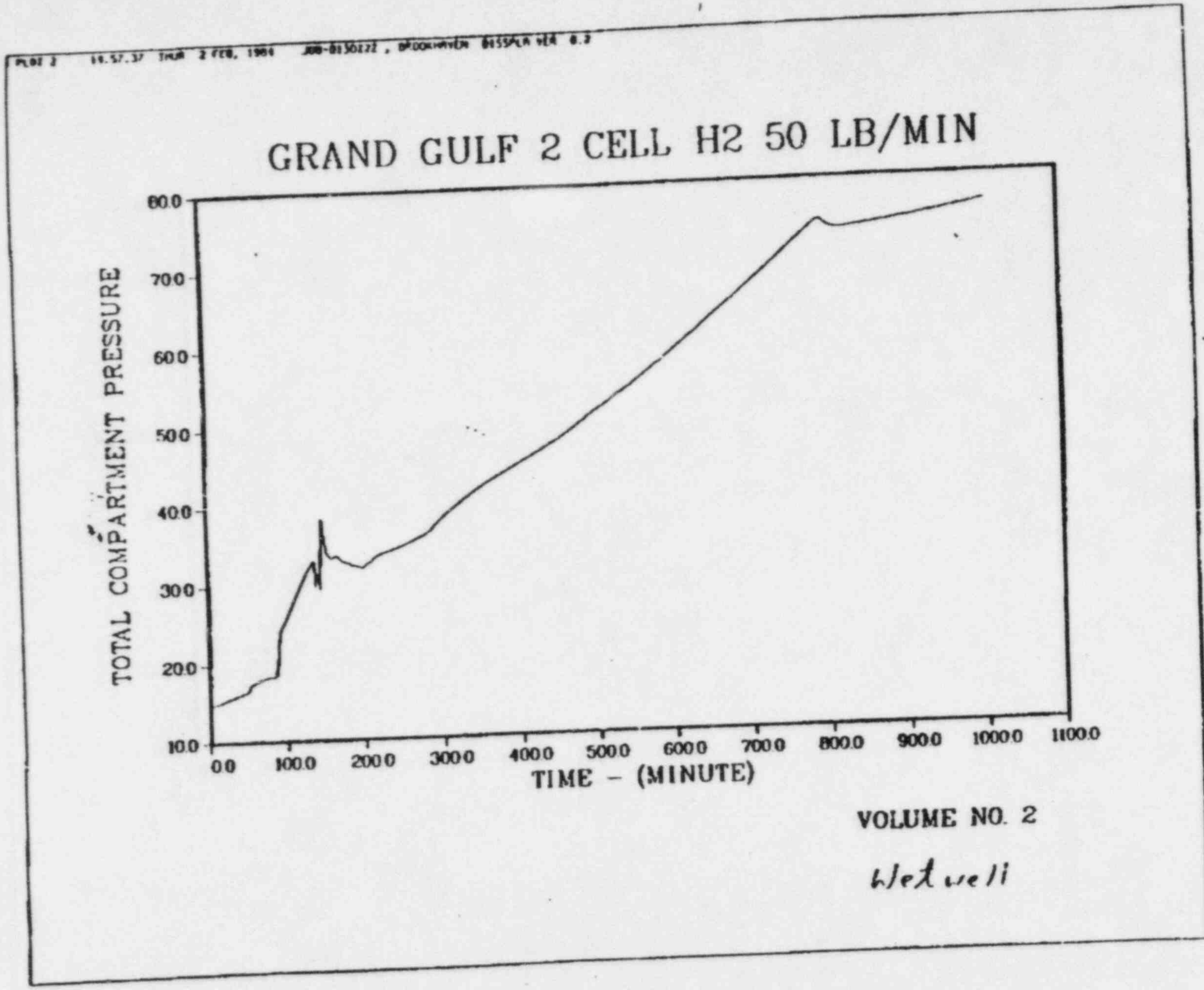
Note: 1.8E-8 = 1.8×10^{-8}

Figure 15.2 CT1-P_a best estimate containment event tree

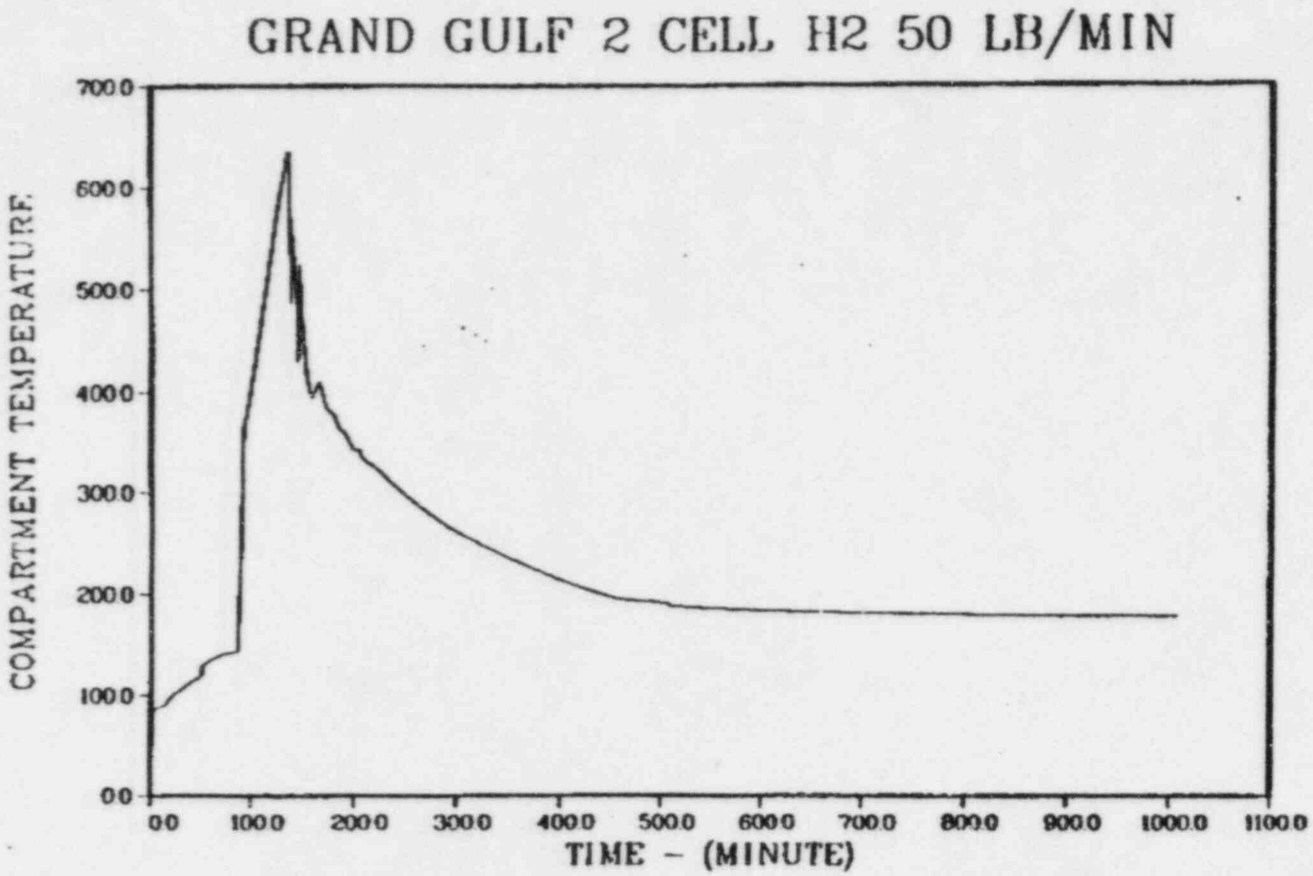
GESSAR - H_2 THREATS TO CONTAINMENT

ADIABATIC BURN OF ALL O_2 CAN OVERPRESSURIZE
CONTAINMENT (~10 ATM)
1-T-E3 TYPE RELEASE

CONTINUOUS BURN 3000 LB. BURN OVER 1 HR. AT GRAND GULF
~40 PSIA, ~600 F


PLOT 2 19.57.37 THUR 2 FEB, 1984 JOB-0130222 , BROOKHAVEN DISSPLA VER 8.2
GRAND GULF 2 CELL H2 50 LB/MIN
TOTAL COMPARTMENT PRESSURE
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
0.0
100.0
200.0
300.0
400.0
500.0
600.0
700.0
800.0
900.0
1000.0
1100.0
TIME - (MINUTE)
VOLUME NO. 2
Wetwell

PLOT 6 14.57.30 THUR 2 FEB, 1984 JOB-0130222 , BROOKHAVEN



VOLUME NO. 2

wetwell

DEFLAGRATION

GLOBAL - NO IGNITORS

- FAILURE OF WETWELL SEAL ASSUMED UNIT PROBABILITY
- SMALL PROBABILITY OF RPV PIPE BREACH OR DRYWELL SEAL FAILURE LEADING TO E1 OR E2 RELEASE, OTHERWISE E3 RELEASE
- ABOUT FACTOR OF 3 IN PERSON-REM CONSEQUENCES

LOCAL - WETWELL SEAL MAY FAIL

- SMALL PROBABILITY OF RPV PIPE BREACH OR DRYWELL SEAL FAILURE LEADING TO E2 OR E3 RELEASE OTHERWISE L3 RELEASE
- ABOUT FACTOR OF 4 IN PERSON-REM CONSEQUENCES

LOCAL DETONATIONS

1-SB-E1 PORTRAYS DRYWELL AND WETWELL EARLY FAILURE

CREDIT FOR PRIMARY SYSTEM RETENTION AND POOL SCRUBBING OF VOLATILES

PERSON-REM CONSEQUENCES ABOUT AN ORDER OF MAGNITUDE GREATER THAN 1-T-L3

1 - T - I2, 1 - T - I2Q PORTRAYS DRYWELL HEAD FAILURE DUE TO DETONATION SHOCK LOAD

ABOUT FACTOR OF 3 IN PERSON-REM CONSEQUENCES DEPENDING ON FAILURE LOCATION

Hydrogen Considerations:

Optimum Ignition Sources

HCOG test program to confirm adequacy of glow plug

Power Source

Diverse Power Source recommended for ignitors

Limitations of Ignition Source

Status of HCOG considerations

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 16, 1985

Docket Nos: 50-390, 50-391
and 50-438, 50-439

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

Subject: Concerns Regarding TVA Construction Sites

Enclosure 1 lists eleven concerns about your Watts Bar facility that have been communicated to the NRC. We ask that you review these concerns and take appropriate steps to assure that your programs and implementation of those programs in these areas are adequate to meet applicable requirements and to support safe operation of the facility. Furthermore, we ask that you address any generic implications of these issues. We recognize that some of these concerns are not very specific. However, that lack of specificity should not lead you to assume there is no basis for concern. Your review of these matters should be broad enough for you to certify the safety significance of these concerns. Pursuant to Section 182 of the Atomic Energy Act of 1954, as amended, we ask that you provide the results of your review as soon as possible to assist us in our evaluation of these concerns. Enclosure 2 lists a number of questions that we have regarding these concerns. Please provide us with your response as soon as possible.

We also ask that you identify any outstanding cases currently under review by TVA's Office of the General Counsel regarding employee harassment, reprisals or intimidation.

We recently received some additional concerns (see Enclosure 3) regarding both your Watts Bar and Bellefonte facilities. You should review these to determine that no new issues related to safe plant operation have been identified. We regret some omissions occur, but this is how they were received by us.

I suggest we meet as soon as you are prepared to discuss your schedule for responding to this letter. Should you have any questions on this matter, please refer them to E. Adensam of my staff on FTS 492-7831.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten resondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Hugh L. Thompson, Jr.

Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

WATTS BAR

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

cc: Herbert S. Sanger, Jr., Esq.
General Counsel
Tennessee Valley Authority
400 West Summit Hill Drive, E 11B 33
Knoxville, Tennessee 37902

Mr. D. Checcet
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. Ralph Shell
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. Donald L. Williams, Jr.
Tennessee Valley Authority
400 West Summit Hill Drive, W10B85
Knoxville, Tennessee 37902

Resident Inspector/Watts Bar NPS
c/o U.S. Nuclear Regulatory
Commission
Rt. 2 - Box 300
Spring City, Tennessee 37381

Regional Administrator
U.S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Mr. David Ellis
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. Mark J. Burzynski
Tennessee Valley Authority
Watts Bar NP
P.O. Box 800
Spring City, Tennessee 37381

BELLEFONTE

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

cc: Herbert S. Sanger, Jr., Esq.
General Counsel
Tennessee Valley Authority
400 West Summit Hill Dr., E11B33
Knoxville, Tennessee 37902

Mr. H. N. Culver
Tennessee Valley Authority
400 West Summit Hill Dr., 249A HBB
Knoxville, Tennessee 37902

Mr. William T. Watters
Licensing Engineer
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. R. A. Wallin
Babcock & Wilcox Company
P.O. Box 1260
Lynchburg, Virginia 24505

Mr. Robert B. Borsum
Babcock & Wilcox Company
Suite 220
7910 Woodmont Avenue
Bethesda, Maryland 20814

Mr. Donald L. Williams, Jr.
Tennessee Valley Authority
400 West Summit Hill Drive, W10B85
Knoxville, Tennessee 37902

Resident Inspector, Bellefonte NPS
c/o U.S. Nuclear Regulatory
Commission
P.O. Box 477
Hollywood, Alabama 35752

J. Nelson Grace
U.S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30323

ENCLOSURE 1

CONCERNS RELATED TO WATTS BAR

1. A concern has been expressed that there is no material control of ASME bolts smaller than 1"; and, therefore, the bolts < 1" are mixed up and no one knows where the good ones are.

2. A concern has been expressed that electrical hangers have been modified after their initial inspection and not reinspected.

3. A concern has been expressed that field modifications have been implemented on components, piping, supports, structures, and embedments that resulted in no accurate records of total loads on these elements.

4. A concern has been expressed that the cumulative effect of tolerances has not been factored into the design and drawings, especially with respect to hanger location.

5. Several concerns have been raised regarding the Independent Design Verification Program conducted by Black & Veatch. These are:

   a) a concern regarding the close out of about 500 items,

   b) a concern that only one construction specification was looked at by Black & Veatch in their review,

   c) a concern that Black & Veatch did not know how the plant was actually built, and

   d) a concern that Black & Veatch only compared the system's design and construction to its design criteria, not to the underlying regulatory criteria.

6. A concern was expressed with respect to the use of three Q lists that all differ.

7. A concern was expressed that the method of identifying NCR's and the use of the Inspection Rejection Notice (IRN) system effectively negated the NCR process. It was submitted this was so because an NCR was only generated when 1) the equipment/component/system/etc. had been previously inspected and accepted; 2) the records for that inspection were in the vault, and 3) there was a subsequent discovery that something was wrong; however, if there were a problem identified in an initial inspection an IRN is generated.

8. A concern has been expressed with respect to structural steel welding requirements in that TVA is using a different code than the code normally used by the industry for structural steel welding.

9. A concern has been expressed that in FSAR amendment #53 TVA lessened the experience requirements for the plant manager.

10. A concern has been expressed regarding weld filler material control, especially in the area of storage and issuance of materials.

11. A concern has been made that the Quality Assurance (QA) organization at construction sites lacks the independence required by NRC regulations. Also the statement was made that inadequate QA organization independence problems were identified in a Management Analysis Company (MAC) report, "Assessment of Organizational Change in the Tennessee Valley Authority Power Program and the Nuclear Quality Assurance Program."

ENCLOSURE 2

QUESTIONS ON WATTS BAR CONCERNS

1. With respect to concern 1 our initial inspection during the week of April 29, 1985, identified instances where unmarked bolts were installed in the facility on ASME components and supports. In addition, the staff also learned that two NCRs (1979 and 1981) have been issued regarding the purchasing and installation of bolts without required markings. Describe the reasons for the apparent QC breakdown in bolt control, the reason for the repeated NCR in 1981 and your evaluation of whether this occurred in other QC areas. In view of the above, what is your basis for determining compliance with criterion VIII of Appendix B to 10 CFR 50? If documentation does not exist which demonstrates compliance with this regulation, describe the process and provide sample documentation which leads you to conclude that you comply with this regulation. Demonstrate that bolts less than 1", which have been installed, comply with all applicable ASME Code requirements related to identification and control.

2. With respect to concern 2, please review your records to determine if such modifications have occurred and, if they did, what assurance you have that the modifications have been properly reinspected. This review should include applicable work requests. Verify that documentation exists to demonstrate that modified electrical hangers were designed and constructed pursuant to applicable FSAR commitments. To the extent such documentation does not exist, what is the basis for concluding that electrical hangers, as currently installed, comply with 10 CFR 50, Appendix B, Criterion X?

3. With respect to concerns 3 and 4, 50.55 Interim Report No. 1 on NRC WBNCEB8419 states that TVA's drawing series 47A050 includes several tolerances (e.g., location of concrete anchorages, movement of attachments, and modification of baseplates) such that the cumulative effect of these tolerances may result in significant increases in baseplate stresses and anchor bolt loads. Interim Report No. 1 also states that there is no evidence that these potential increases due to cumulative effects were considered in the design of various supports and that the cumulative effect of these tolerances could increase baseplate stress by 150% and anchor bolt load by 50%. Provide the TVA engineering specifications establishing acceptable dimensional installation tolerances for supports, baseplates, and anchorages. Provide the analytical bases for establishing the above procedures which demonstrate that the effect the tolerances have on the stresses on loads in interfacing components and structures will cause these values to exceed their allowable limits. Provide the process used when field modifications are made and confirm that each modification exceeding TVA engineering specification installation tolerances has been analyzed to demonstrate continued compliance with design allowable parameters. Provide sample documentation which demonstrates how this process has been used.

4. With respect to concern 5, please review this concern to determine no new issues are raised which would impact your assurance that the Black & Veatch review was properly designed and conducted and that TVA's close out of identified open items was consistent with your licensing commitments and safe operation of the plant.

   In addition, please address the following questions:

   The NRC staff can identify only one General Construction Specification (GCS-G-32) in the documents reviewed by B&V. How many other General Construction Specifications are applicable to the auxiliary feedwater system? If you identify other applicable General Construction Specifications that were not reviewed by B&V, how could TVA use B&V to support a conclusion that construction complied with the FSAR commitment? What corrective actions (i.e., design, hardware, procedural modifications) have been taken as a result of the B&V review? Does B&V agree that these actions resolve the concerns expressed in the B&V findings? What specific actions have been taken in systems other than the Auxiliary Feedwater System to determine the extent to which deviations found by B&V in the AFW system existed in other systems? How have such actions been documented?

5. With respect to concern 6, identify the documents that demonstrate your compliance with Criterion II, Appendix B, 10CFR Part 50, for maintaining a Q list from the date of the construction permit (CP).

6. With respect to concern 7, is this a proper description of the NCR process? Please verify and certify that reporting of deficiencies meets your licensing commitments and the regulations and that IRN's and NCR's are properly controlled. Is there a master file of IRN's and their resolutions? You may wish to consider having your Quality Technology Company Employee Response Team solicit employee views regarding improper use of the IRN process in lieu of the NCR process.

7. With respect to concern 8 please verify your code use for this welding to assure regulations and licensing commitments have been met. Verify your implementation of other types of welding conformed to the accepted standards. Please provide memoranda or other documents indicating problems with TVA's AWS welding program not previously provided to NRC.

8. With respect to concern 10, how does your program assure the ASME Code requirements of 10 CFR 50.55a are met and 10 CFR Part 50, Appendix B Criterion VIII traceability requirements are met. To what version of the ASME Code was TVA committed in the CP? Did this version of the ASME code require traceability of filler material to welds by heat and lot numbers? What internal or external approvals for your program were required and received?

What version of the specification GCS-G29M, Process Specification 1.M.3.2(RO) & 1.M.3.1(R7) was used prior to 1/12/83 and 1/13/83? There appears to be an inconsistency between these two process specifications in that 1.M.3.2(RO) for power boilers is more stringent than 1.M.3.1 (R7) for nuclear plants. Describe how TVA is implementing these process specifications in the current version of GCS-G29M in the field?

9. With respect to concern 11, describe the adequacy of the independence of the QA organization as it applies to Watts Bar. In addition, describe actions taken by TVA to resolve problems identified in the MAC report and actions TVA is taking with respect to the report's recommendations. Provide any analysis which has been conducted by TVA to determine the extent to which Watts Bar design and construction quality may have been compromised as a consequence of deficiencies enumerated by the MAC report. If no analysis has been conducted, do you intend to conduct such an analysis? If not, why not?

Enclosure 3

Electrical, I&C and Diesel Generators

- Electrical and I&C Regulations (Reg. Guides, NUREGs, Bulletins and Notices) have been ignored and violated to a very large degree at all plants.
  - Caused by a lack of knowledge by personnel
  - Caused by a poor attitude toward safety and regulations by personnel
  - Caused by a lack of knowledge of industry positions on regulations
- 5% voltage drop at each plant causes problems
  - Cycles diesel generators unnecessarily, degrading reliability
  - Too many plant shutdowns
  - TVA compensates by operating buses at higher than normal voltage ratings, anticipating voltage reductions, stressing equipment and components unnecessarily and reducing their lives and reliabilities
  - Inadequate voltage regulation for buses
- Diesel Generator margins inadequate
  - TVA has added DGs to BF, Sequoyah and Watts Bar
  - Each time a question is raised, TVA must conduct another study
  - TVA adds [illegible] without upgrading licensing documentation
- Diesel generator reliability problems
  - Requires reliability upgrading program
  - Requires reduction in number of starts
  - Requires much attention given to testing program
  - Requires preventative maintenance upgrading program
  - Requires more interaction with INPO and other utilities, as well as vendors, to establish resolutions to problems

- Electrical separation and physical separation of redundant wiring and cabling and for equipment and components are all inadequate at all plants
  - Detailed reviews need to be made (They are so extensive that a consultant probably should be used, providing independence from TVA)
- Environmental Qualification of electrical and I&C equipment and components is inadequate at all plants
  - Qualification was often not done
  - If done, records do not exist in many cases, resulting in requalification or replacement of items
  - Current upgrade programs needs scrutiny
- WBN - (maybe other plants) Class 1E and Non-Class 1E Batteries are unacceptably supported (no battery tie-downs)
  - Unistrut supports unacceptably used
- Human Factors engineering and/or reviews have not been implemented for control panels and stations at WBN (possibly other plants also) - Violation of intent of NUREG-0700
  - Too many poor engineering practices in this area
- Out of service tags for valves, electrical equipment, etc., at Bellefonte have been violated everywhere
  - Extremely serious personnel safety problem
- Thermal overload bypass and indication problems at WBN - probably have similar problems meeting Reg. Guide 1.97 at other plants
- There are cable ampacity problems at WBN where derating was not properly considered
  - Probably problems at other plants
- Inadequate management, control and <u>status listing</u> of a.c. and d.c. electrical loads, including diesel generator loads

- Inadequate control of or preparation of calculations for loads
- Inadequate management and control of load margins, including electrical loads and mechanical loads (heat, BHP, etc) that translate into electrical loads

° Cable tray fill criteria of 60% for I&C cables is inadequate

- National Electrical Code allows 40% and 50% on exception basis. TVA violates code
- Industry practice is 40%
- The situation is even worse with the addition of spray-on fire retardent materials which take up space in trays

° Cable pull tension monitoring is lax

° Cable bending radii problems

° Computer cable routing program inadequate and its status system is inadequate

° Cable trays are too heavily filled; cables [illegible]

° Cable megger readings are not stored as QA records, losing traceability

° Construction Test and Installation Specs (Called General Construction Specs with G-   numbers) are often incomplete and inadequate

° Electrical testing and planning inadequate

- Engineering either does not address testing or does so inadequately
- Acceptance criteria is inadequate to nonexistent

° Electrical Standards and Guides are treated as guides and are not adequately incorporated in design criteria as requirements

° Electrical design criteria, where it exists, is not complete, is vague, and in general is inadequate

° Cabling is routed outside trays, coiled on tray supports or floors, tied on sides of trays and supports, tied on bottoms of trays, etc. All this and more exists at WBN, where extremely bad cable practices exist such as the above and 90° wire bends [illegible]

° Between 400 and 500 breakers were unacceptably set at WBN. EN DES practices and attitudes concerning these were poor. The National Electrical Code and good engineering practices were violated.

° Many cable trays at WBN are full, some exceeding 100% tray capacities, and they are not identified at site or in computer status as full

° Wall penetrations of cable trays are not identified by name and/or number at WBN

° Lighting fixtures at WBN are not properly restrained and caged to prevent them from becoming missiles or swinging missiles during seismic events

° WBN - (and Possibly other plants) - Unistrut material is used to support instruments, pipes, conduit, control stations and panels, fluid piping on skids, instrument lines, CO_2 fire protection piping, fire protection water piping, lighting, etc.

- All unacceptable use for Seismic Category 1 support
- Items supported as such may either fail or become missiles to cause other [Illegible]

° TVA commitments in FSAR, SER, and NRC Question Responses are treated lightly and are not being met in a wide number of areas

- Personnel do not follow regulations and commitments, and do not think they even need to report deviations or change commitments and obtain NRC acceptance

° TVA safety and licensing evaluations by EN DES (Including NEB) are inadequate and appear too much in cover up mode

° TVA personnel have attitude problems in meeting regulatory commitments

- Too many crafts and others on site at WBN

- Gross lack of knowledge of regulations and their seriousness by TVA personnel at all levels

- Lack of frequent visits to sites by Designers

- Communications problems among designers, constructers and operation personnel

- Procurement specs, drawings and vendor supplied documents not per as-built and/or as delivered configurations

  - TVA inadequately reviews vendor work

  - TVA receipt and inspection of equipment are inadequate (Example: TVA in many cases does not inspect until ready to install - not when received)

- Construction process does not always follow EN DES requirements documents or vendor requirments/instructions

  - These do not always get included on as-built documents

  - Too much after-the-fact approval

  - QC inspection is often inadequate - (It only takes a walk thru a plant such as WBN to see examples everywhere)

- Engineering (EN DES) inadequately addresses and considers operation, maintenance, testing and construction requirements and general industry practices, in the design process

  - There are no forced interactions with other utilities

  - There is no formal system to track and assign commitments for problems identified to INPO

  - There is poor tracking of NRC experience information

- Improper reporting of events at operating plants or in design/construction

  - TVA personnel are inadequately trained and not knowledgeable in what is reportable

- Lack of adequate (or any) configuration control (management) in EN DES or at sites
  - Poor interface control between systems
- Lack of traceability of design requirements
  - Standard answer is "Its TVA Practice"
- Design/installation drawings do not always represent or include design requirements
  - Design guides or standards are utilized only when designer wants to use them
  - Design guides/standards inadequate in many areas
  - These are misused - applicable parts are [Illegible]
- Material control is poor
  - Traceability of requirements, paperwork, and materials are inadequate
  - Paperwork for quality records is poor
  - Storage requirements implementation is poor
  - Handling of equipment in storage and during and after construction is poor. WBN equipment in many cases is in poor condition and filthy dirty inside and outside
  - Equipment receipt and inspection is inadequate (identified previously)
  - These problems exist at Bellefonte and WBN (probably elsewhere)
- Lack of adequate tracking for EN DES commitments and design changes
- Lack of good status system (punch lists) for completion of commitments and completion of NRC actions, and completion of work at sites. Plant construction, pre-op, etc. status is poor
- Project Engineering inadequate (or nonexistent) to incorporate TVA and industry operating [Illegible]

° Calculation Problems

- Some are not ever prepared
- Some are inadequate in scope and quality
- Some are not stored as quality records, but are destroyed
- Traceability of design requirements is impacted due to above problems
- There is inadequate interface control and control of calculations

° TVA has set up design criteria (WBN) and, after the fact, have inactivated a large percentage of criteria

° As-Built drawings and documents are nonexistent or in poor condition in many cases

° TVA does not adequately (or at all) independently verify vendor calculations or designs.

- There are no design reviews of vendor design

° TVA does not conduct independent design reviews of its work

° QA has not effectively audited the design and construction process

° Lack of coordination of effects of upcoming (near or long term) design changes with all disciplines and site construction

- inadequate evaluation of impacts (not under configuration control)

° Lack of accountability of TVA personnel and management for not following procedures, regulations, etc. and for not doing adequate and acceptable job

° Too much blame on QA for quality problems versus emphasizing and demanding an ethic to do it right the first time. Put quality into design and construction

- Commitment (action) system in TVA nonexistent
  - No action party and schedule
- Lack of effective communications and interface control among organizations with EN DES - Branches, Projects, Procurement, etc.
- Protective and defensive attitudes of NEB and various Branch/Project groups concerning problems rather than an attitude to admit [Illegible]
- Lack of proper environments and fire protection in equipment storage areas
- Lack of knowledge (on site and in EN DES) as to status of QCIRs and IRNs
- Untimely closeout of ECNs
  - Lack of knowledge of status of ECNs or designs affected

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 16, 1985

Docket Nos: 50-390, 50-391
and 50-438, 50-439

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

Subject: Letter on TVA Construction Sites Dated May 16, 1985

In Mr. H. Thompson's letter to you dated May 16, 1985, "Concerns Regarding TVA Construction Sites", there was an error of omission in Question 3 of Enclosure 2. The second request which reads "Provide the analytical bases for establishing the above procedures which demonstrate that the effect the tolerances have on the stresses on loads in interfacing components and structures will cause these values to exceed their allowable limits." should read as follows:

> "Provide the analytical bases for establishing the above procedures which demonstrate that the effect the tolerances have on the stresses or loads in interfacing components and structures will not cause these values to exceed their allowable limits."

We apologize for any inconvenience and ask that you respond to the restated question above.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

cc: See next page

Vij TP5 9

GESSAR II CONTAINMENT CAPABILITY

A PRESENTATION TO THE ADVISORY

COMMITTEE ON REACTOR SAFEGUARDS

WASHINGTON, D.C.

GENERAL ELECTRIC COMPANY

JULY 12, 1985

MK III CONTAINMENT DESIGN

COMPARISON

PLANT	CONT. TYPE	DES. PRESS (PSIG)	ULT. PRESS CAPAB. (PSIG)	DOME CONFIG.	CONT. DIA (FT)	CONC. FILL @ BASE
GESSAR II (238)	STEEL + CONCRETE SHIELD BLDG.	15	85	TORI-SPHERICAL	120	YES
GRAND GULF (251)	LINED REINF. CONCRETE	15	67	HEMI-SPHERICAL	123	N/A
PERRY (238)	STEEL + CONCRETE SHIELD BLDG.	15	94	TORI-SPHERICAL	120	YES
RIVER BEND (218)	STEEL + CONCRETE SHIELD BLDG.	15	90	TORI-SPHERICAL	120	YES
CLINTON (218)	LINED REINF. CONCRETE	15	95	HEMI-SPHERICAL	124	N/A

POTENTIAL FOR UNSTABLE PROPAGATION OF AN UNDETECTED FLAW

- MAXIMUM STRESS IN THE CONTAINMENT FOR ULTIMATE PRESSURE CAPABILITY (85 PSIG) LOADINGS CONSIDERED.

- LOWER BOUND FRACTURE TOUGHNESS PROPERTIES CONSIDERED.

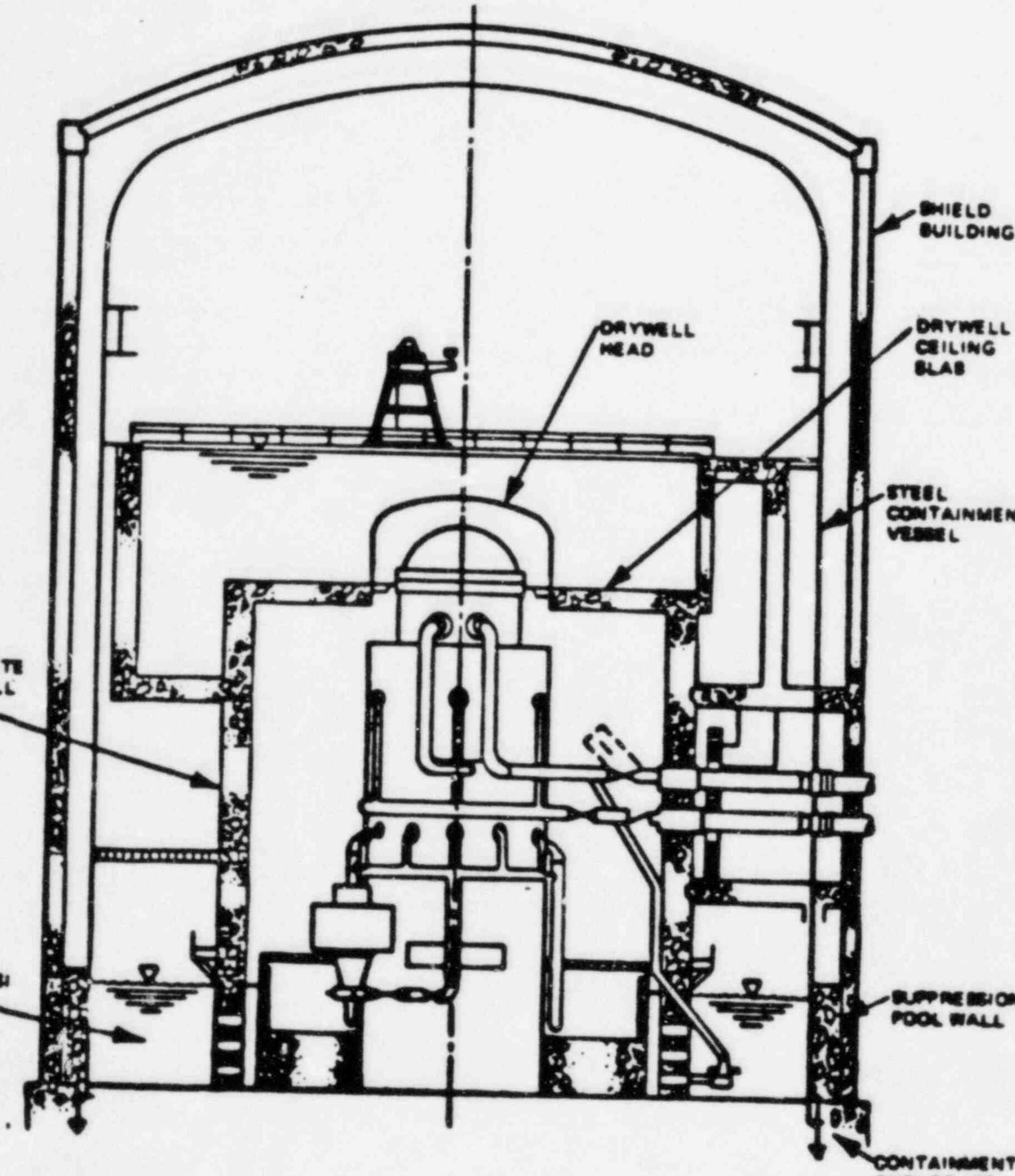
  - PLATE MATERIAL
  - WELDMENT
  - HEAT AFFECTED ZONE

- BASED ON A CONSERVATIVE FRACTURE ANALYSIS, A POTENTIAL CRACK OF UP TO 1/2 INCH DEEP (>25% OF WALL THICKNESS) AND 3 INCHES LONG CAN BE TOLERATED WITHOUT PROPAGATION TO FAILURE.

- WELDING PROCEDURES LIMIT FLAWS TO ~10% OF WALL THICKNESS.

- UNLIKELY THAT ANY WELD DEFECT SIGNIFICANTLY OVER 2% OF WALL THICKNESS AT WELD JOINTS WILL ESCAPE FULL RT OF WELDS.

UNDETECTED FLAWS WILL NOT AFFECT CALCULATED ULTIMATE PRESSURE CAPABILITY OF THE CONTAINMENT



Mark III Containment Buildings of Standard Plant

DOMINANT CONTAINMENT FAILURE MODES

o LOADING AND FAILURE MODE

1. H2 DETONATION IN CONTAINMENT o SHOCK WAVE o INTERNAL PRESSURE ON CONTAINMENT o EXTERNAL PRESSURE ON DRYWELL	1. LOCAL: CONTAINMENT FAILURE ABOVE WATER LINE GLOBAL: A) CONTAINMENT FAILURE ABOVE WATER LINE B) DRYWELL CEILING FAILURE
2. H2 COMBUSTION IN CONTAINMENT o INTERNAL PRESSURE ON CONTAINMENT o SMALL EXTERNAL PRESSURE ON DRYWELL (~5 PSIG)	2. A) CONTAINMENT FAILURE ABOVE WATER LINE B) NO DRYWELL FAILURE SINCE NO SIGNIFICANT LOADS
3. H2 SLOW BURNING o INTERNAL PRESSURE ON CONTAINMENT o SMALL EXTERNAL PRESSURE ON DRYWELL (~5 PSIG)	3. A) NO CONTAINMENT FAILURE SINCE PRESSURE IS LOW B) NO DRYWELL PRESSURE SINCE NO SIGNIFICANT LOADS
4. STEAM AND/OR NON-COMBUSTIBLE GAS OVERPRESSURIZATION o SMALL INTERNAL PRESSURE ON DRYWELL ~5 PSIG o INTERNAL PRESSURE ON CONTAINMENT	4. A) CONTAINMENT FAILURE ABOVE WATER LINE AT CONTAINMENT ULTIMATE PRESSURE CAPABILITY B) NO DRYWELL FAILURE SINCE NO SIGNIFICANT LOADS

- VALIDATION OF FAILURE MODES
  - BY ANALYSIS
    - LOAD TYPES AND APPLICATION
    - CONTAINMENT AND DRYWELL STRUCTURAL CONFIGURATION
    - CALCULATED STRESSES
    - HIGHEST STRESSED POINTS ASSUMED TO FAIL FIRST
    - NO FAILURES ASSUMED WHERE LOADS ARE SIGNIFICANTLY LESS THAN THE DESIGN LOADS.

> - DRYWELL STRUCTURE, HEAD, PERSONNEL LOCK NOT CHALLENGED.
> - SUPPRESSION POOL BYPASS DUE TO DRYWELL BOUNDARY FAILURE SHOULD NOT BE A CONCERN.

FAILURES WITHIN DRYWELL

- FAILURES IN DRYWELL IGNORED ON THE BASIS OF LOW PROBABILITY
  - REACTOR PRESSURE VESSEL ONLY.
- RPV INSPECTION
  - AS PART OF FABRICATION
    - 100% RADIOGRAPHY
    - 100% UT OF ACCESSIBLE PORTIONS (ALL WELDS ACCESSIBLE IN BWR/6)
  - ISI
    - 25% UT WITHIN 10 YEARS
    - 100% UT IN PLANT LIFE
- CRD HOUSINGS
  - AS PART OF FABRICATION
    - 100% PENETRANT TESTING OF CRD HOUSING TO VESSEL WELDS
    - 100% UT OF CRD HOUSING TO VESSEL WELDS
  - ISI
    - EXCLUDED
    - WELD IN COMPRESSION
    - FAILURE OF ONE HOUSING PENETRATION POSTULATED

DRYWELL HEAD AND CONNECTION
DRYWELL PERSONNEL LOCK

- o DESIGNED TO ASME, SECTION III, NA-3352.
- o MATERIAL AS ALLOWED BY ASME, SECTION III, NE 2000.
- o FABRICATION AND INSTALLATION TO ASME, SECTION III, NE 4000.
- o TEST PRESSURE
  - o 30 PSIG FOR HIGH PRESSURE TEST.
  - o 3 PSIG FOR LOW PRESSURE TEST.
- o INSPECTION AND TESTING
  - o INSPECTION TO ASME SECTION III, NE 5000.
  - o REPEAT SHOP TESTS AT 30 PSIG PRESSURE FOR LEAKS.
  - o FIELD TESTS
    - o PREOPERATIONAL STRUCTURAL PROOF TEST AT 30 PSIG.
    - o INTEGRATED HIGH PRESSURE LEAK RATE TEST.
    - o INTEGRATED LOW PRESSURE LEAK RATE TEST.
    - o PERIODIC LEAK RATE TEST AT LOW PRESSURE.
    - o AFTER EACH CLOSING OF DRYWELL HEAD AND PERSONNEL LOCK, CONNECTION TESTED FOR LEAKS AT 30 PSIG.

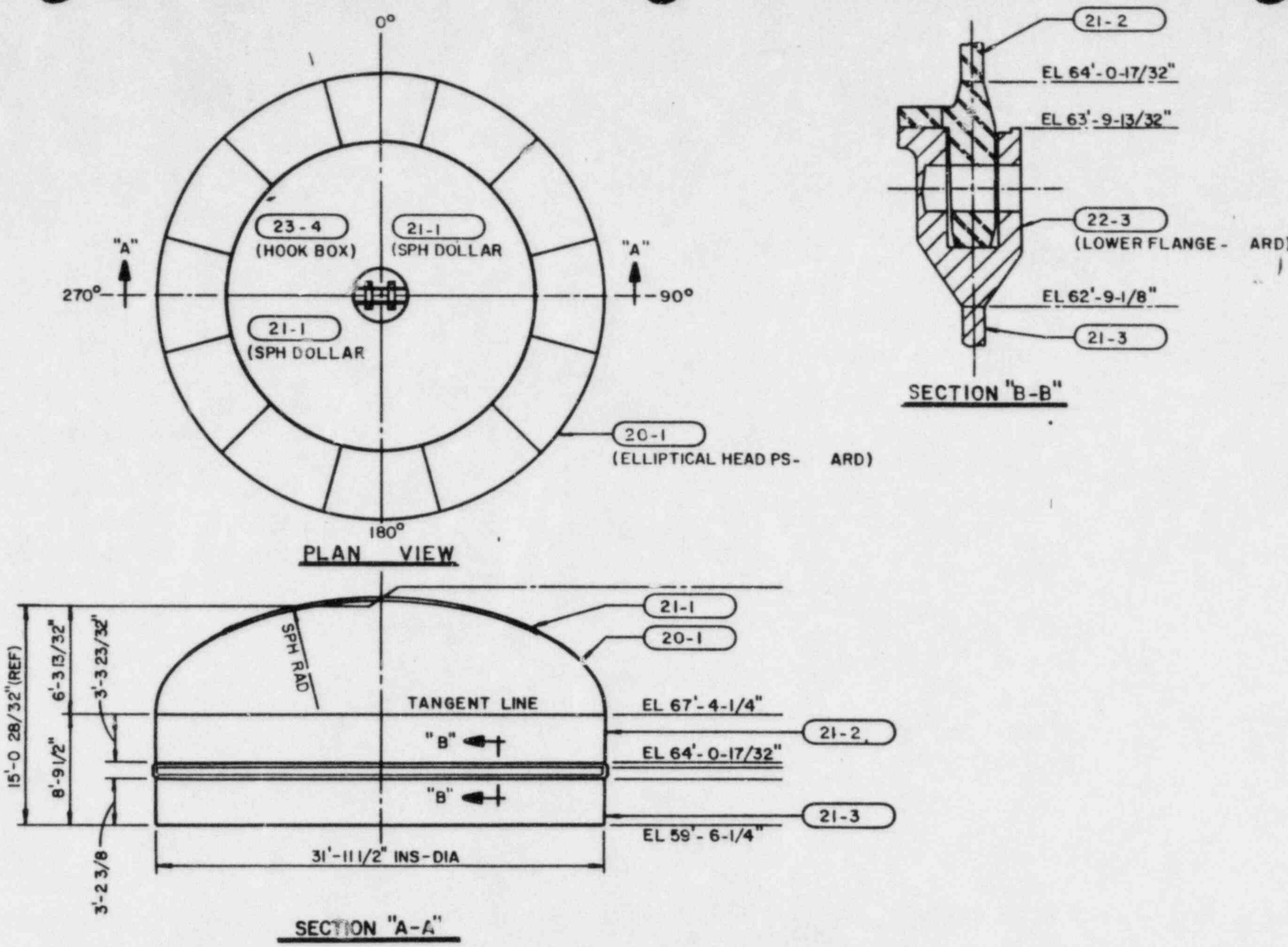


Figure 3.8-11. Drywell Head

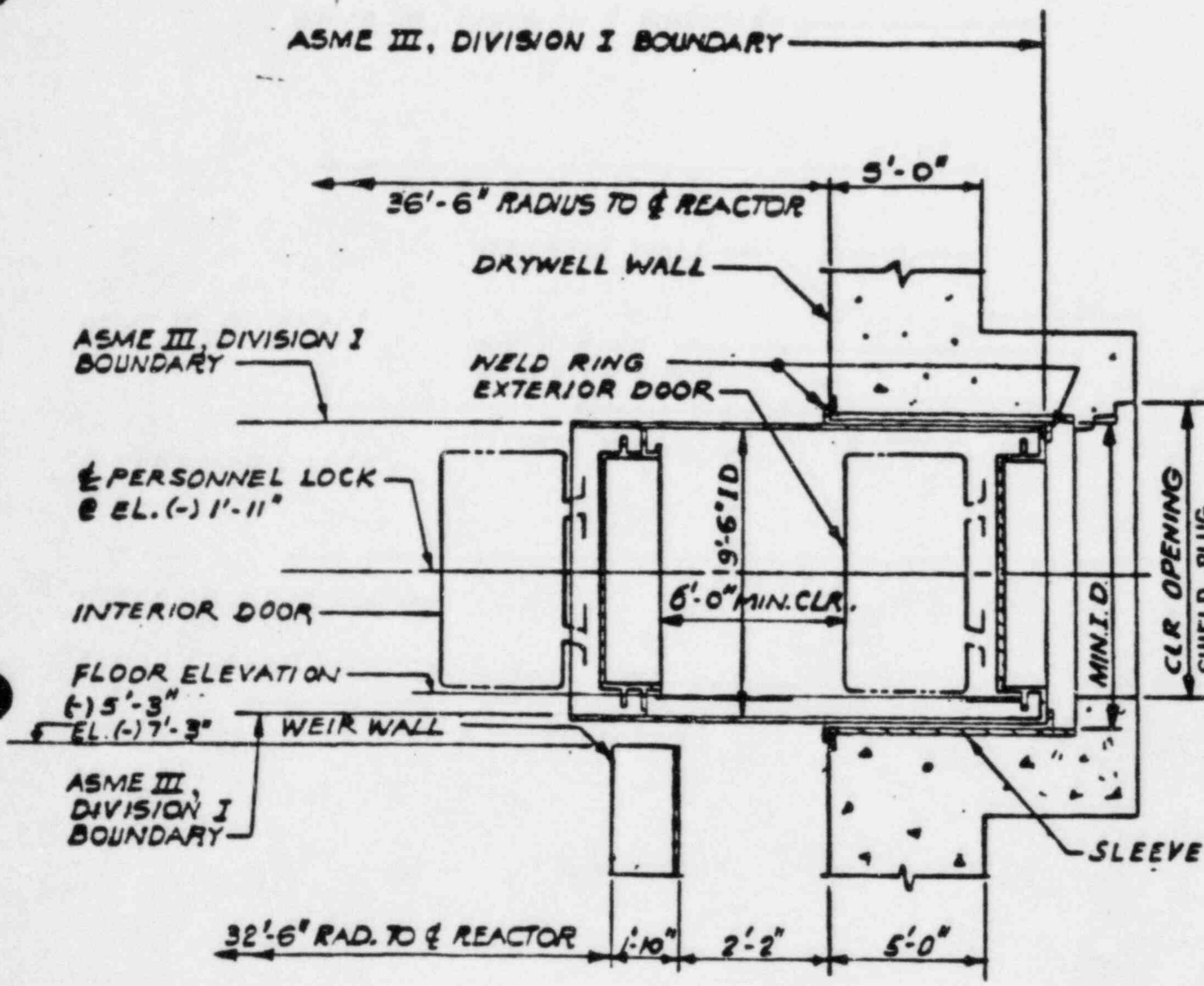
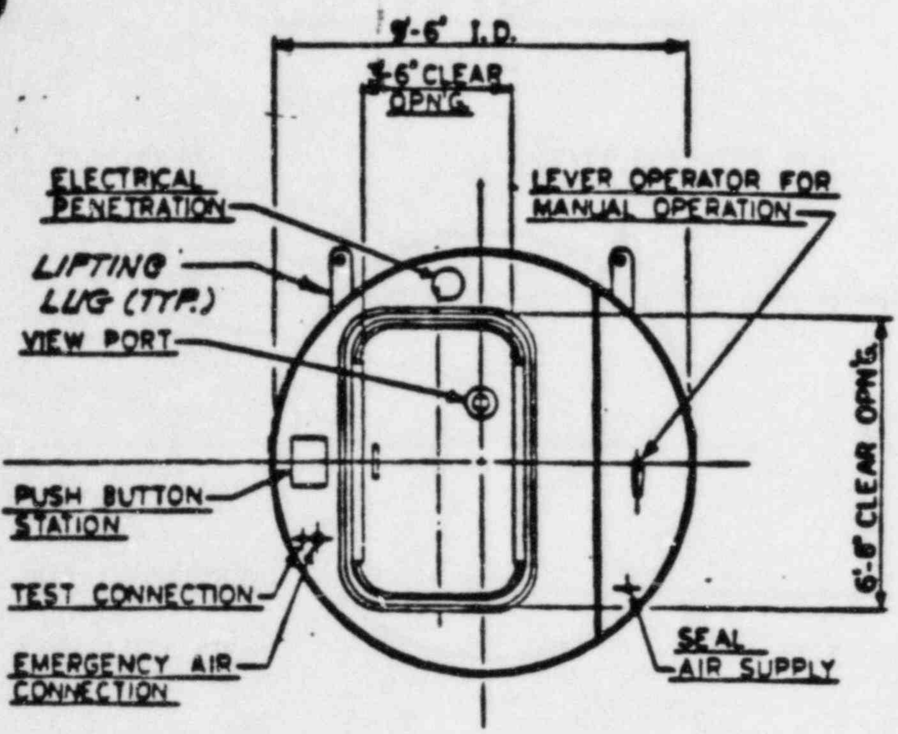
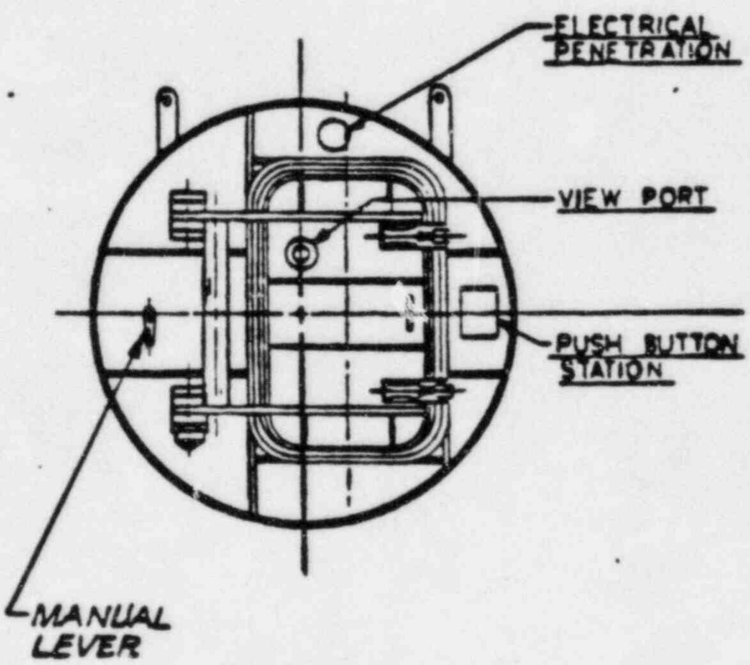


FIGURE 5 ELEVATION VIEW OF DRYWELL STRUCTURE PERSONNEL AIRLOCK



CONTAINMENT SIDE



DRYWELL SIDE

FIGURE 6 END VIEWS OF DRYWELL STRUCTURE PERSONNEL AIRLOCK

EFFECT OF MOLTEN CORE ON DRYWELL STRUCTURES

0 MOLTEN CORE ASSUMED TO BURN 6 FEET DEEP HOLE IN BASEMAT.

0 TEMPERATURE OF MOLTEN CORE ASSUMED TO BE 4000°F.

0 HEAT CONDUCTION ANALYSIS AND LINEAR ELASTIC STRESS ANALYSIS PERFORMED.

0 RESULTS

- 0 TEMPERATURE IN DRYWELL WALL = $\sim$150°F
- 0 DEFLECTION OF DRYWELL WALL = $\sim$0.5 INCHES
- 0 STRESSES IN DRYWELL WALL AT BASEMAT
  - 0 $\sim$3500 PSI IN CONCRETE (F'C=4000 PSI)
  - 0 $\sim$40 KSI IN STEEL (FY = 46 KSI)

NO DANGER EXPECTED TO DRYWELL WALL OVERALL STABILITY

Tp. 7 - Reich

STRUCTURAL VERIFICATION STUDIES:

- TORISPHERICAL STEEL CONTAINMENT
- DRYWELL (STEEL) HEAD
- REINFORCED CONCRETE DRYWELL ROOF SLAB
- RELIABILITY EVALUATION OF STEEL CONTAINMENT

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC. bnl

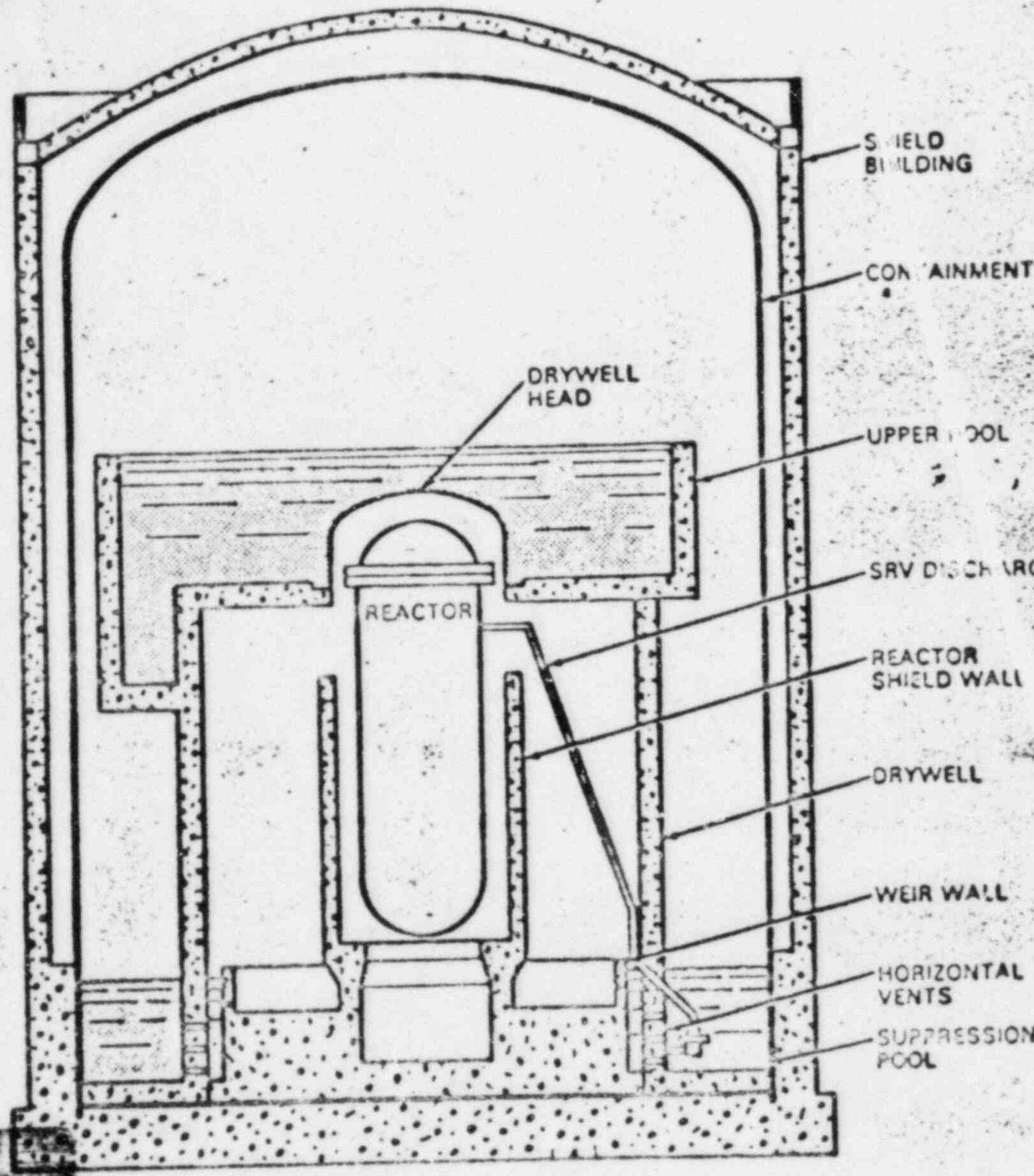


Figure 15.1 Principal features of MARK III containment

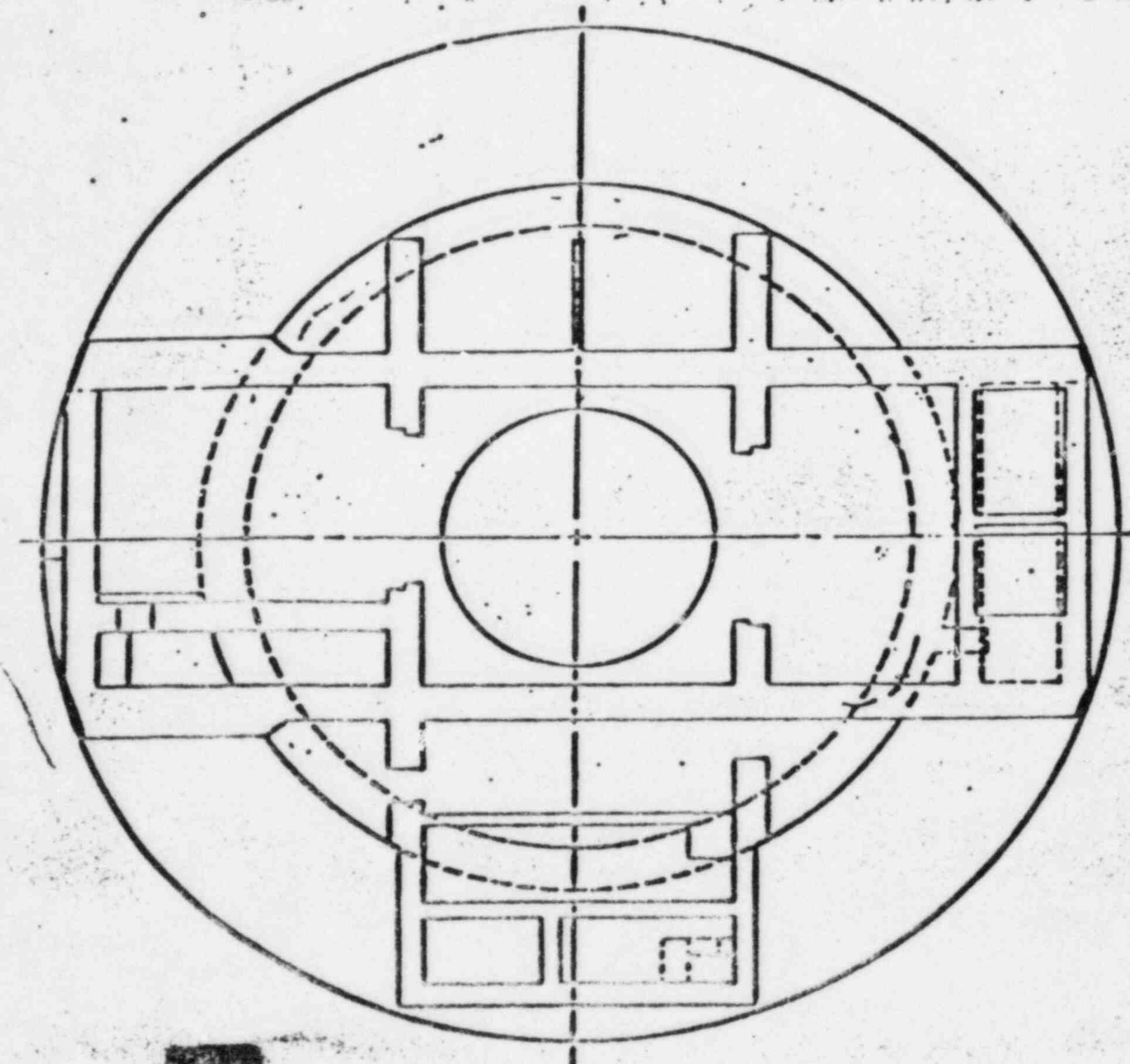


Fig. 12 - Plan View of Drywell Roof Slab

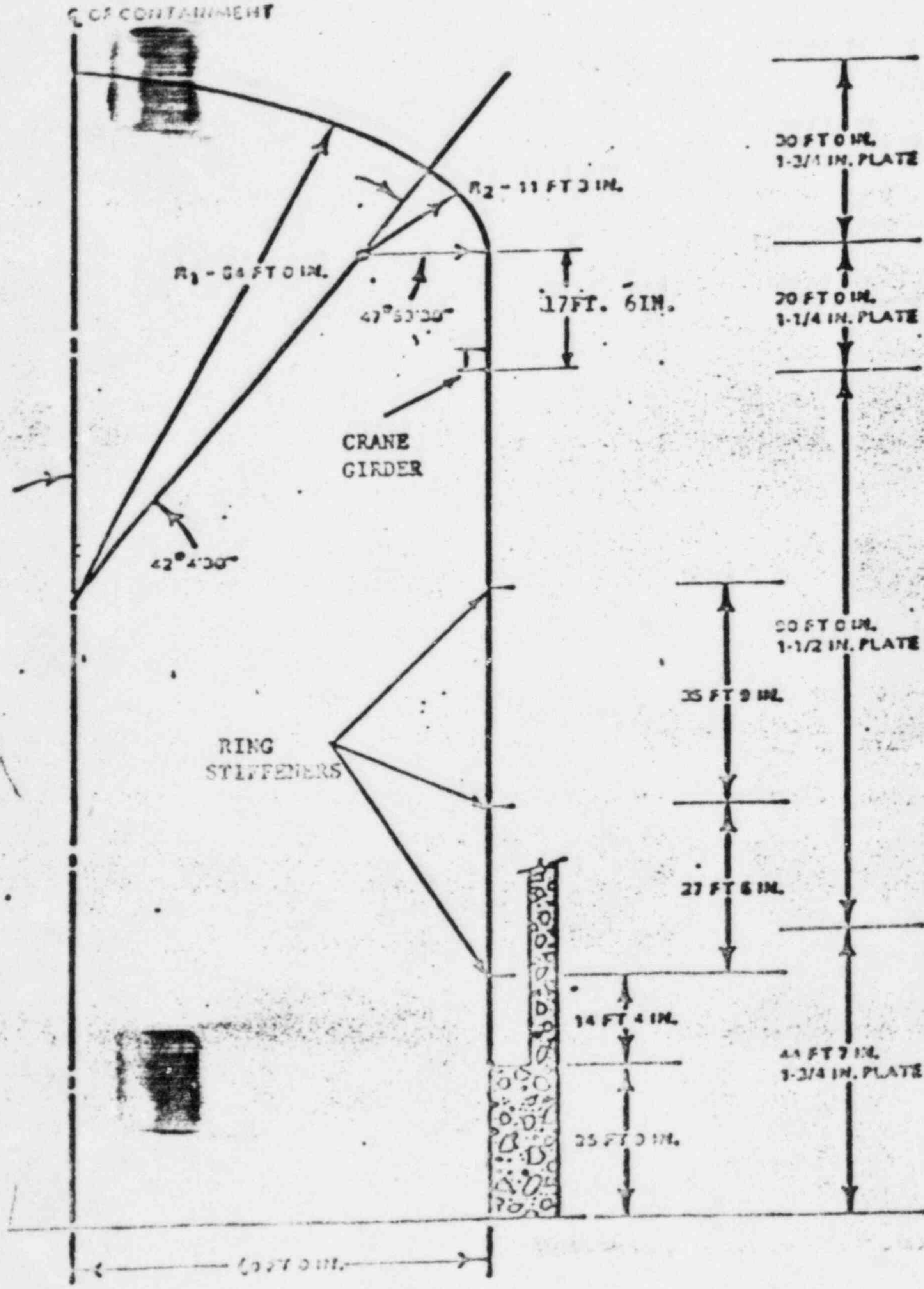


Fig. 1 - Geometrical Details of Torispherical Steel Containment

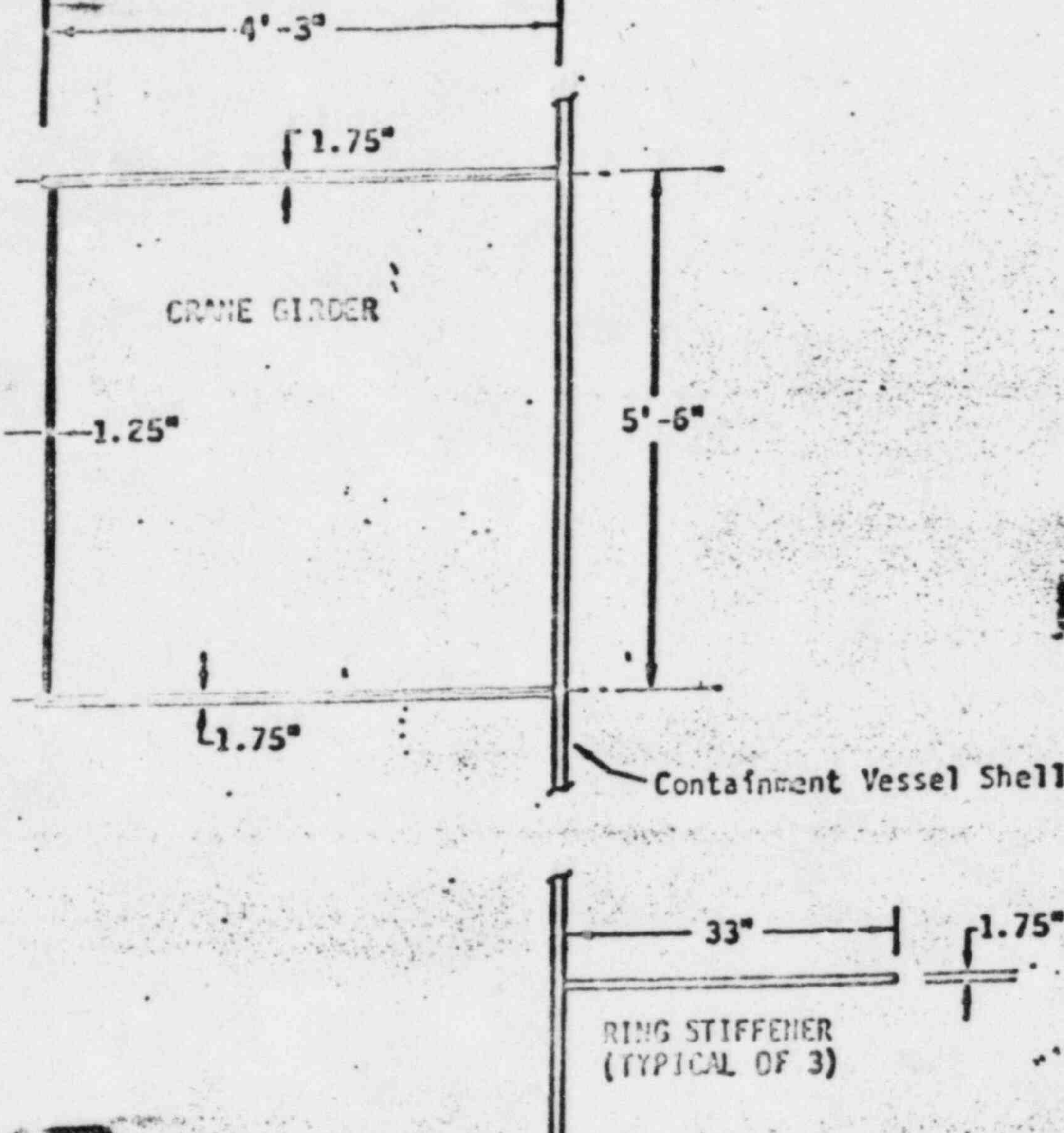


Fig. 2 - Detailed Geometry of Ring Stiffeners and Crane Girder

ANALYSIS PERFORMED FOR TORISPHERICAL CONTAINMENT

A) PLASTIC LIMIT ANALYSIS

B) SMALL DEFORMATION ELASTIC-PLASTIC FINITE-ELEMENT ANALYSIS

C) LARGE DEFORMATION ELASTIC-PLASTIC FINITE-ELEMENT ANALYSIS

D) BUCKLING EVALUATION

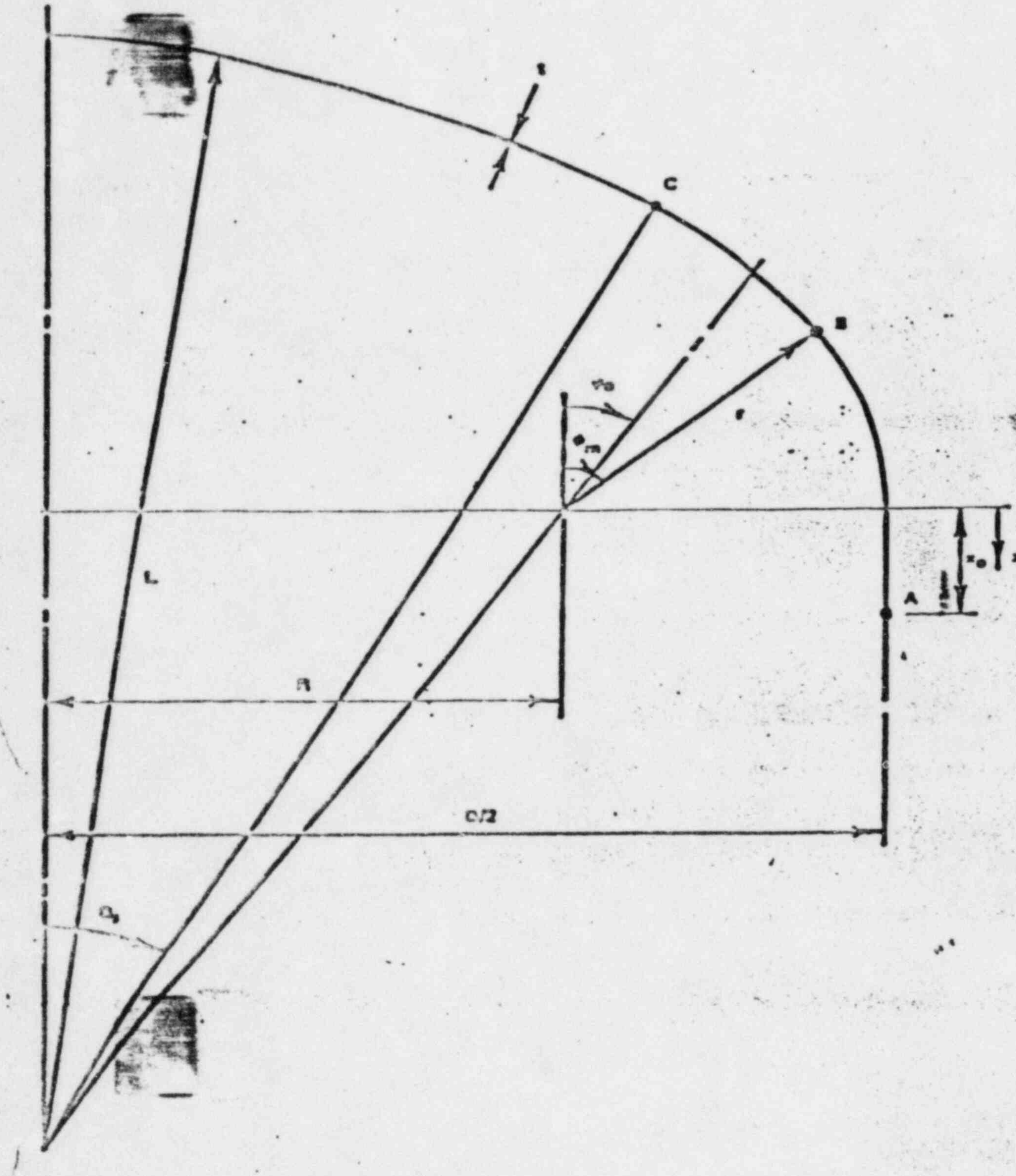


Fig. 3 - Location of Plastic Hinge Circles and Nomenclatures.

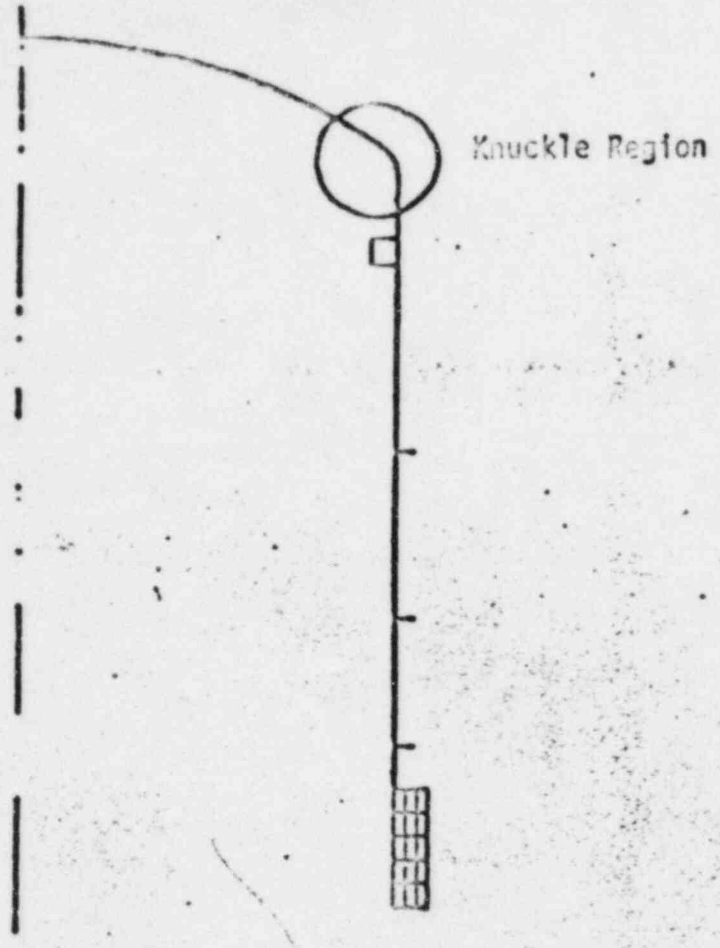


Fig. 4 - Finite Element Model for the Torispherical Containment Steel

Fig. 5 - Finite Element Grid in the Knuckle Region

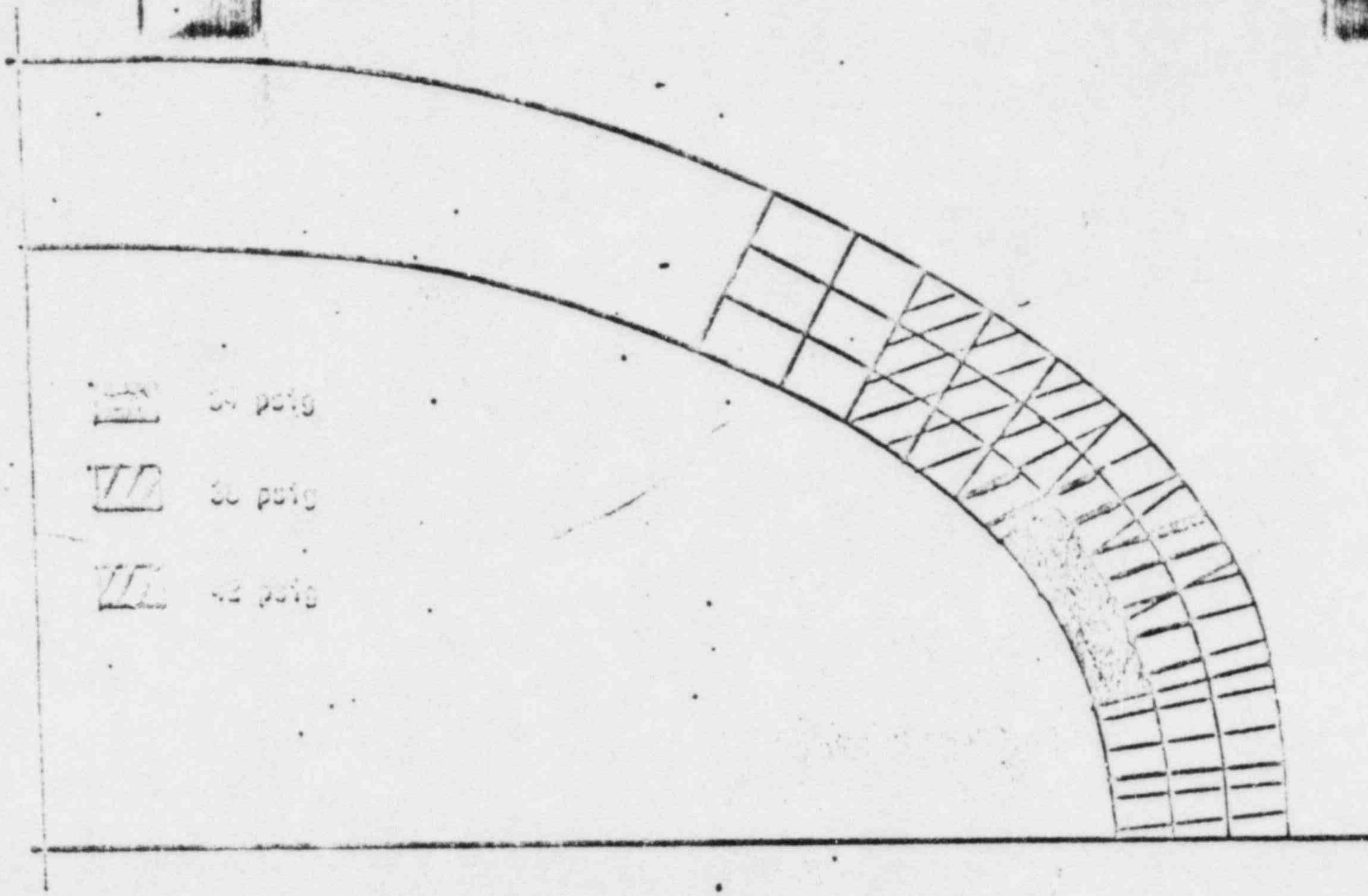


Fig. 7 - Plastic Elements in the Knuckle Region of the Containment Shell

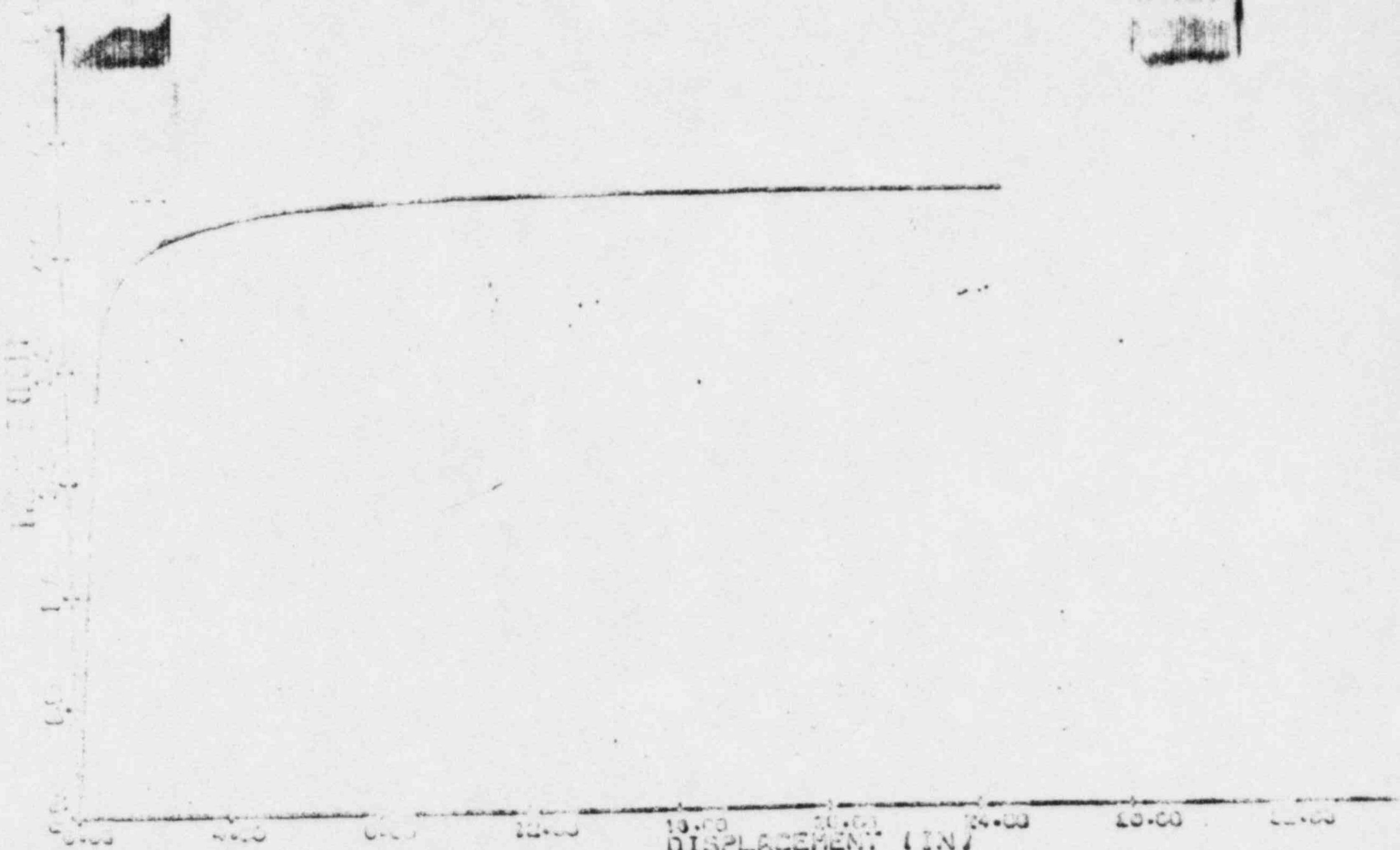


Fig. 6 - Inward Radial Displacement of Node 661 vs. Internal Pressure for Small Deformation Analy.

661 vs. Internal Pressure for Large Deformation Analysis

[illegible] ctric [illegible]

[illegible] Versus Internal Pressure

[illegible]

( ) [illegible] STIC [illegible] - [illegible]3 PSIG

( ) [illegible] STIC [illegible] STIC P.E. - [illegible]2.7 PSIG

( ) [illegible] STIC [illegible] P.E. - [illegible] AT [illegible] PSIG, SLOPE NOT YET FLAT.

( ) [illegible] - [illegible]1.3 PSIG

[illegible]

( ) [illegible] P.E. - [illegible]Y [illegible] FLAT.

) [illegible]

) LEVEL A [illegible]

) LEVEL [illegible]

) [illegible]

([illegible])

DRYWELL ROOF SLAB

( ) FINITE ELEMENT [illegible]) STEEL [illegible] - 125 PSIG TOP
[illegible] D BOTTOM 9"
[illegible] CKED, [illegible] IDDLE
SECTION INTACT,
[illegible] EARS ELASTIC.

( ) S[illegible] FIED [illegible] (GE)

( ) [illegible] RE [illegible] PSIG

( ) [illegible] E [illegible] 5 [illegible] IG

MATERIAL SA 516 GR 70

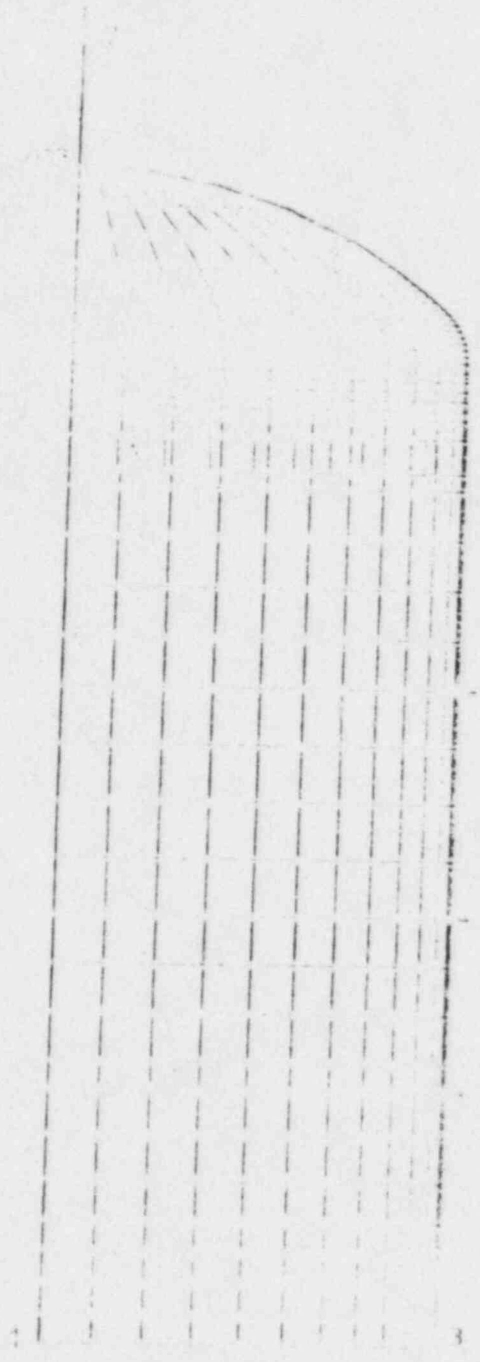
MOD. OF ELASTICITY 29×10^3 KSI (CONSTANT)

YIELD STRENGTH (MEAN)
(NORMAL DIST.) 48.3 KSI

STANDARD DEVIATION 3.12 KSI

PRESSURE
MEAN ([illegible] S [illegible] ER)
COEFFICIENT OF VARIATION [illegible] IS [illegible] OF THE MEAN
[illegible]L DISTRIBUTION

LIMIT STATE [illegible] CONNECTED STRAINS [illegible]P,
MERIDIONAL)
[illegible]



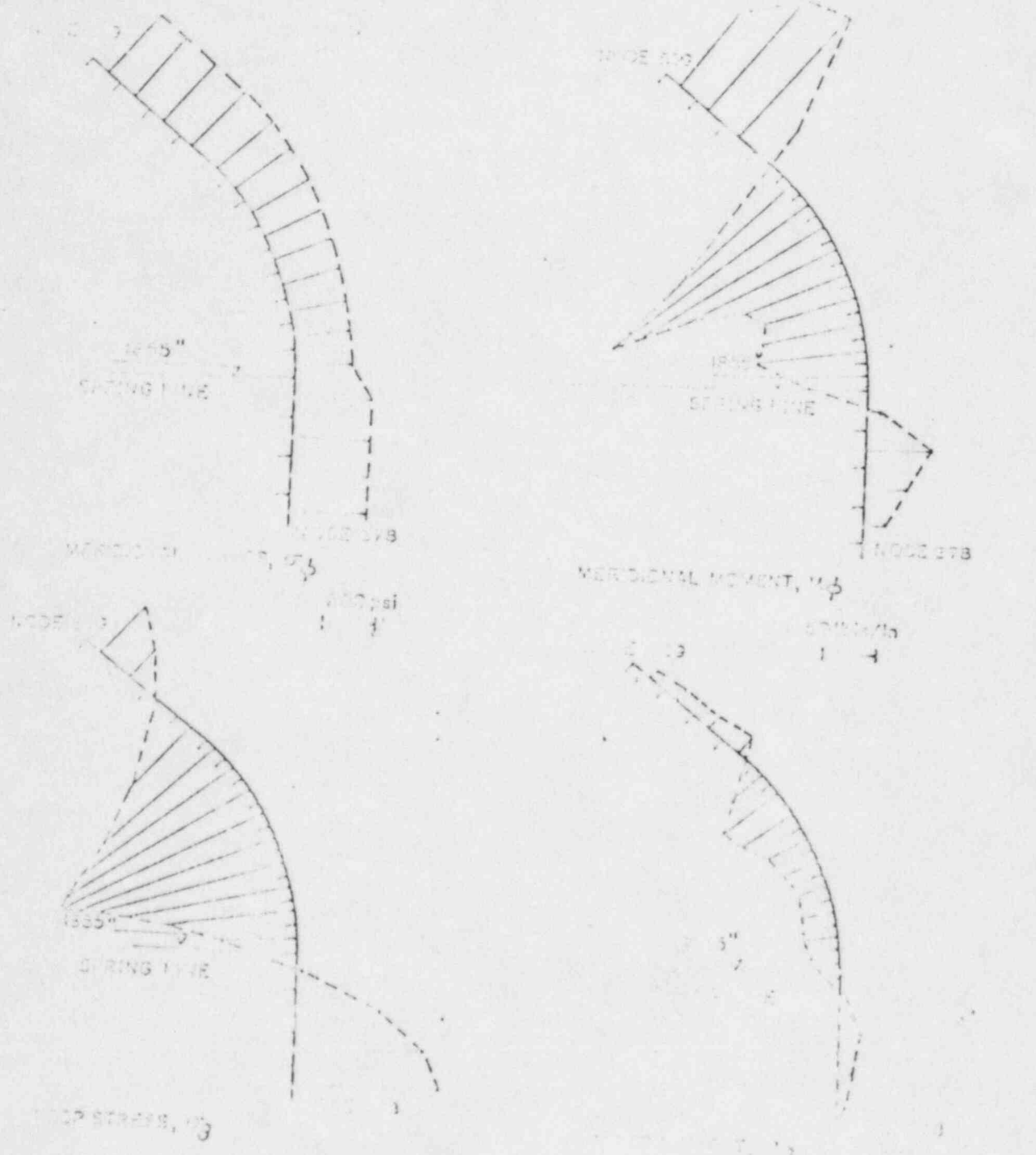


Fig. 5 - [illegible] in Knuckle Region [illegible]

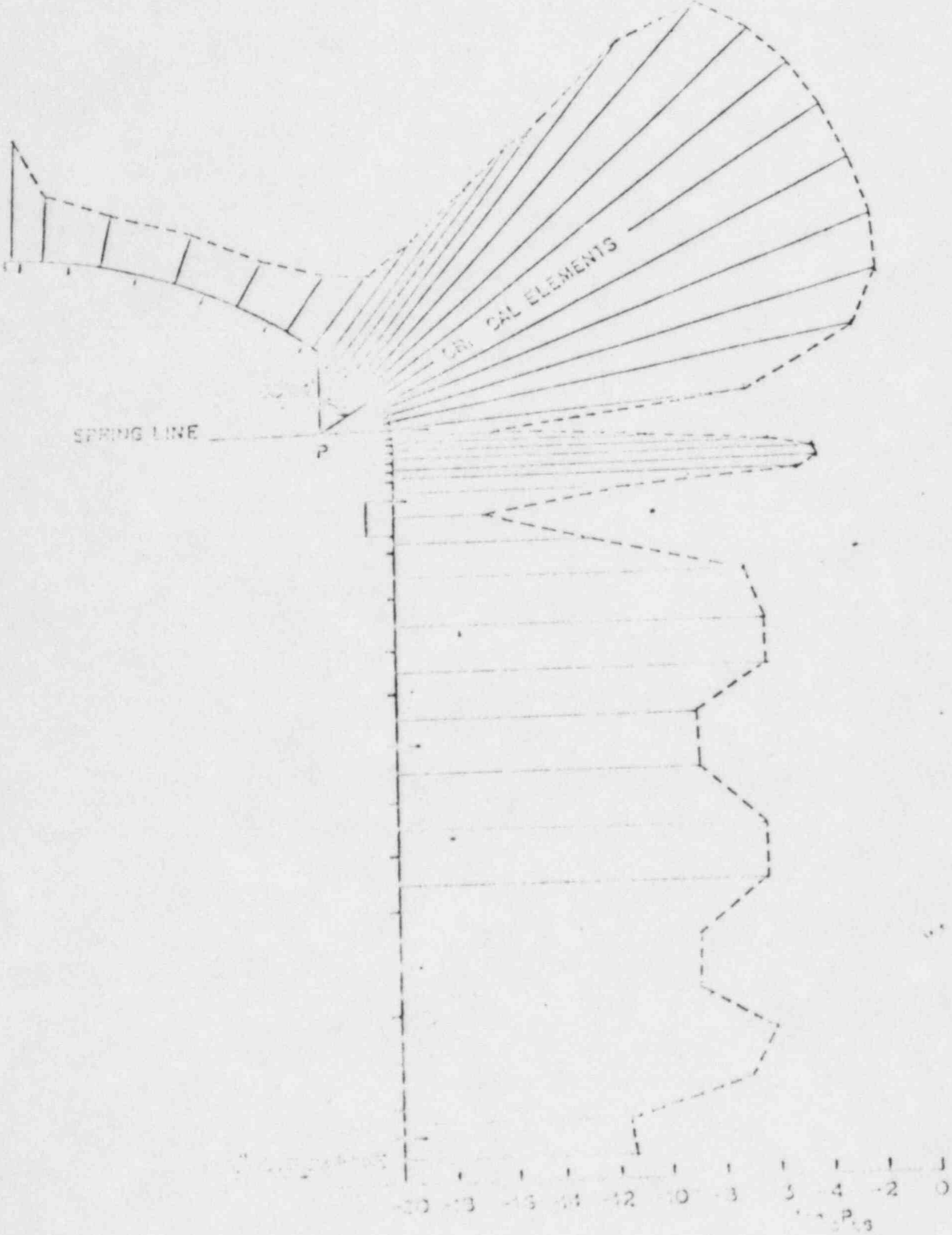


FIG. 2 [illegible]

Mean Pressure $\bar{P}_H$(*) (psi)	Limit State Probabilities
35	6.[illegible] x 10^{-2}
40	2.19 x 10^{-1}
45	4.32 x 10^{-1}
50	6.27 x 10^{-1}
55	7.69 x 10^{-1}
60	8.61 x 10^{-1}
65	9.16 x 10^{-1}
70	9.49 x 10^{-1}
75	9.68 x 10^{-1}
80	9.80 x 10^{-1}
85	9.87 x 10^{-1}
90	9.91 x 10^{-1}
95	9.94 x 10^{-1}
[illegible]	9.[illegible] x 10^{-1}

*σ_H = 0.2 $\bar{P}_H$

MEAN PRESSURE LOAD — PROBABILITIES

$\bar{p}$ (psi)	$\sigma_p = 0$	$\sigma_p = 0.15\bar{p}$	$\sigma_p = 0.20\bar{p}$
60	$5.[illegible]5 \times 10^{-45}$	1.0×10^{-8}	6.40×10^{-6}
70	9.47×10^{-31}	$2.[illegible]6 \times 10^{-5}$	9.60×10^{-4}
80	$2.[illegible]5 \times 10^{-19}$	2.72×10^{-3}	1.66×10^{-2}
90	8.33×10^{-11}	3.71×10^{-2}	8.64×10^{-2}
100	6.03×10^{-5}	0.155	0.229
110	$9.[illegible]3 \times 10^{-2}$	0.332	0.410
115	0.499	0.498	0.498
120	0.[illegible]3	0.605	0.580
130	0.[illegible]2	0.773	0.714
140	1.0	0.878	0.811
150	1.0	0.937	0.876
160	1.0	0.[illegible]8	0.[illegible]8
170	1.0	0.[illegible]83	0.946

Yield Strength: Mean, $\bar{s}_y$ = 48.3 ksi,

Standard Deviation, σ_{s_y} = 3.12 ksi

Y [illegible] [illegible]ISTIC [illegible]S)

(A) YIELD LIMIT STATE (CNL) MEAN FAILURE PRESSURE - 45 PSIG

(B) GE PLASTIC LIMIT STATE - 51 PSIG

(C) 2% MAXIMUM PRINCIPAL STRAIN LIMIT STATE (CNL) - 115 PSIG

[illegible]

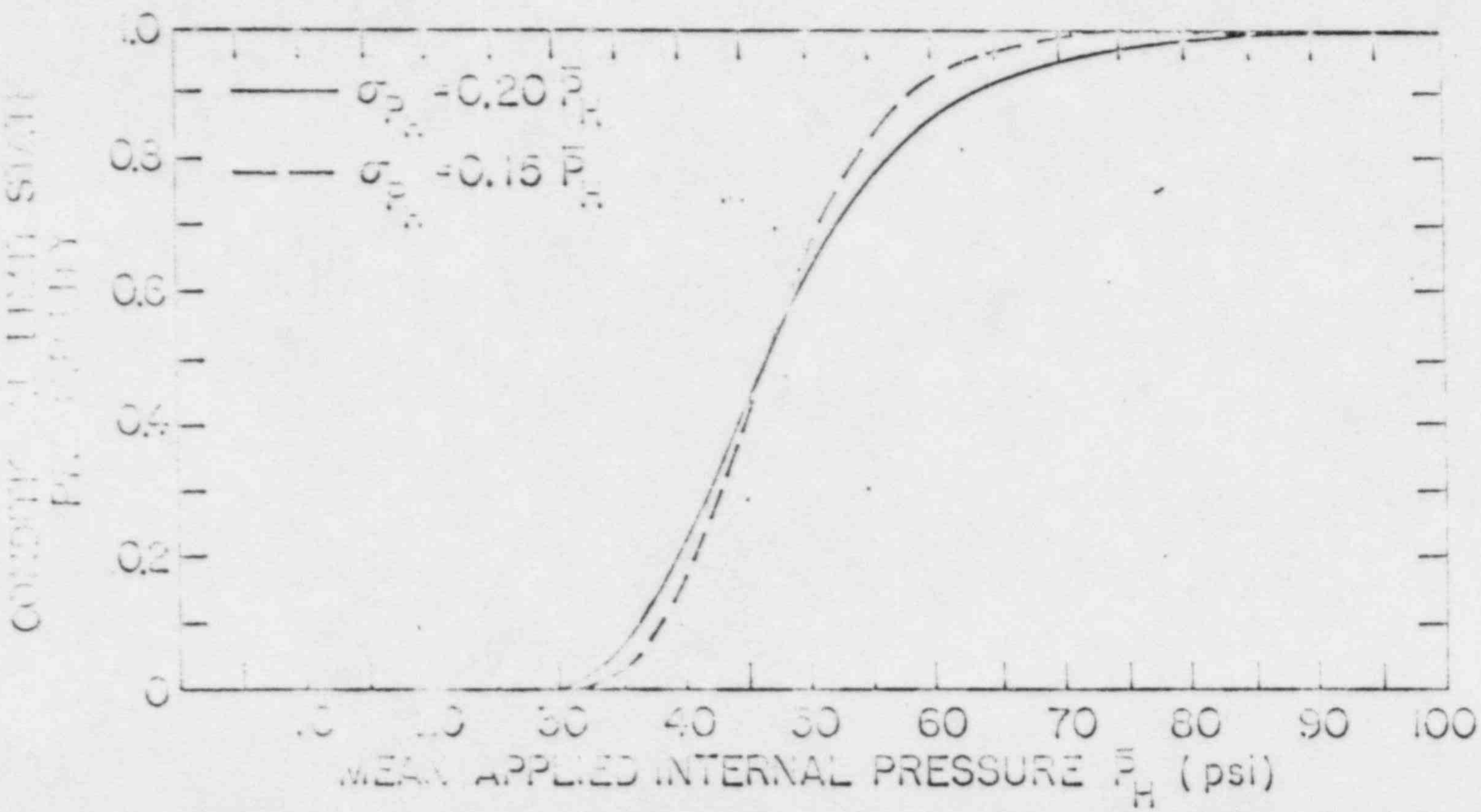


Fig. 9 Conditional Limit State Probabilities for Steel Containment.